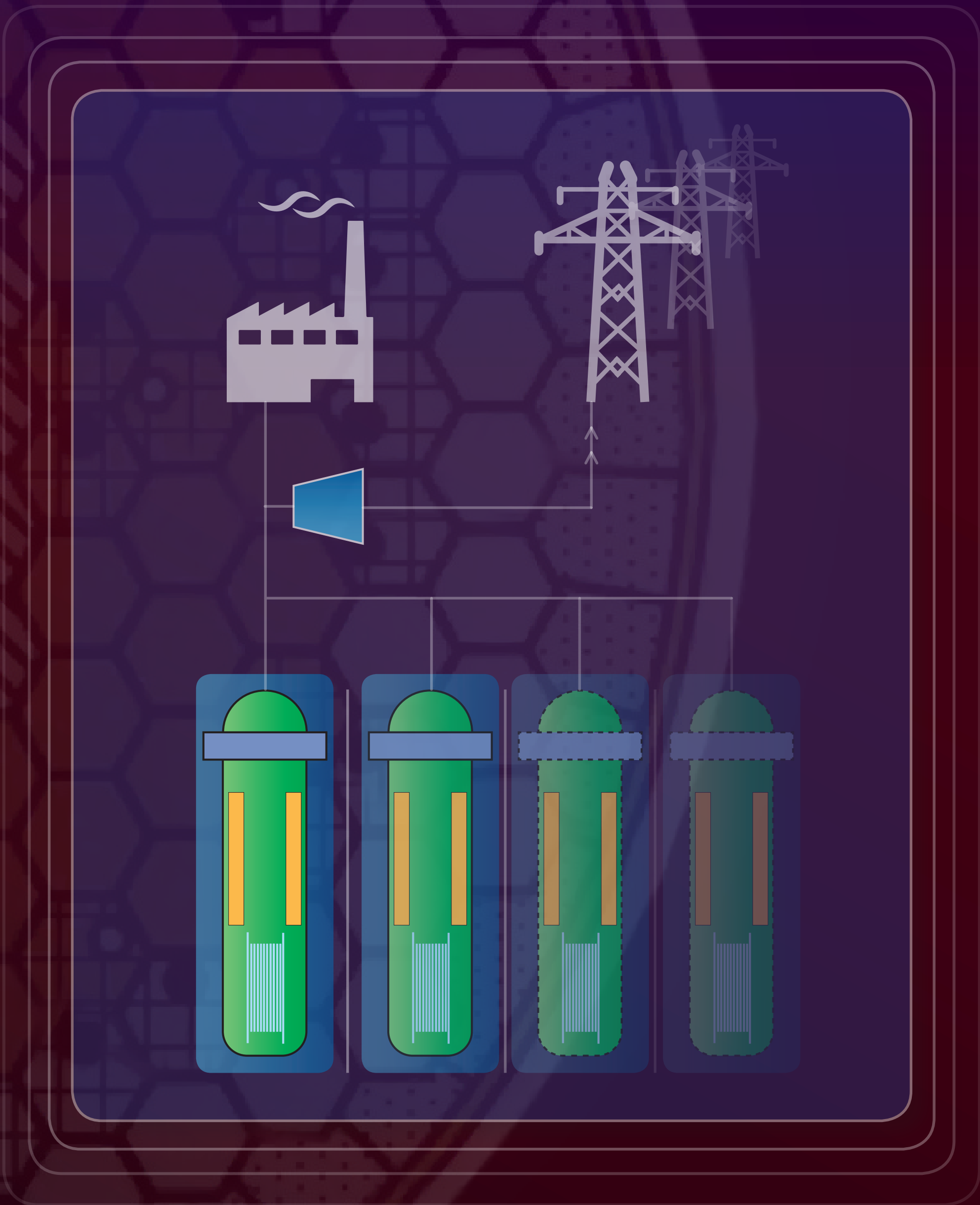
Advances in Small Modular Reactor Technology Developments

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

2018 Edition



IAEA
International Atomic Energy Agency

ADVANCES IN SMALL MODULAR REACTOR TECHNOLOGY DEVELOPMENTS

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IAEA Advanced Reactors Information System (ARIS)
http://aris.iaea.org

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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 regard, 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 published in 2012, 2014 and 2016. 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. The content on the specific SMRs is provided by the responsible institute or organization and is reproduced, with permission, in this booklet.

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 IV.

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 M. Hussain, F. Reitsma, M. H. Subki and H. Kiuchi 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 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 under discussion with 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 more than 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 People's Republic of 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 2019 and 2022. In addition, the Russian Federation have already manufactured six RITM-200 reactors (an integral PWR) with four units already installed in the Sibir and Arktika icebreakers, to be in service in 2020.

This booklet provides a brief introductory information and technical description of the key SMR designs and technologies under different stages of development and deployment. To assist the reader to easily understand the 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.

The 2018 edition comprises of six (6) parts arranged in the order of the different types of coolants, the neutron spectrum adopted, and a sixth part (a new category) on other SMRs that do not make use of the traditional coolants and/or fuel design.

Part One (Land-based water-cooled SMRs) presents the key SMR designs adopting integral light water reactor (LWR) technologies. This represents the most mature technology since it is like most of the large power plants in operation today.

Table 1: Summary of Main Design Features and Status of SMRs included this Booklet

Design	Output MW(e)	Type	Designers	Country	Status
PART 1: WATER COOLED SMALL MODULAR REACTORS (LAND BASED)					
CAREM	30	PWR	CNEA	Argentina	Under construction
ACP100	100	PWR	CNNC	China	Basic Design
CAP200	150/200	PWR	CGNPC	China	Conceptual Design
DHR400	(District Heating)	LWR(pool type)	CNNC	China	Basic Design
IRIS	335	PWR	IRIS Consortium	Multiple Countries	Conceptual Design
DMS	300	BWR	Hitachi GE	Japan	Basic Design
IMR	350	PWR	MHI	Japan	Conceptual Design
SMART	100	PWR	KAERI	Republic of Korea	Certified Design
ELENA	68 kW(e)	PWR	National Research Centre “Kurchatov Institute”	Russian Federation	Conceptual Design
KARAT-45/100	45/100	BWR	NIKIET	Russian Federation	Conceptual Design
RITM-200	50 × 2	PWR	OKBM Afrikantov	Russian Federation	Under Development
RUTA-70	70 MW(t)	PWR	NIKIET	Russian Federation	Conceptual Design
UNITHERM	6.6	PWR	NIKIET	Russian Federation	Conceptual Design
VK-300	250	BWR	NIKIET	Russian Federation	Detailed Design
UK-SMR	443	PWR	Rolls-Royce and Partners	United Kingdom	Mature Concept
mPower	195 × 2	PWR	BWX Technologies	United States of America	Under Development
NuScale	50 × 12	PWR	NuScale Power	United States of America	Under Development
SMR-160	160	PWR	Holtec International	United States of America	Preliminary Design
W-SMR	225	PWR	Westinghouse	United States of America	Conceptual Design
PART 2: WATER COOLED SMALL MODULAR REACTORS (MARINE BASED)					
ACPR50S	60	PWR	CGNPC	China	Preliminary Design
ABV-6E	6-9	Floating PWR	OKBM Afrikantov	Russian Federation	Final design
KLT-40S	70	Floating PWR	OKBM Afrikantov	Russian Federation	Under construction
RITM-200M	50 × 2	Floating PWR	OKBM Afrikantov	Russian Federation	Under Development
SHELF	6.4	Immersed NPP	NIKIET	Russian Federation	Detailed Design
VBER-300	325	Floating NPP	OKBM Afrikantov	Russian Federation	Licensing Stage
PART 3: HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS					
HTR-PM	210	HTGR	INET, Tsinghua University	China	Under Construction
GTHTR300	300	HTGR	JAEA	Japan	Basic Design
GT-MHR	285	HTGR	OKBM Afrikantov	Russian Federation	Preliminary Design
MHR-T	205.5x4	HTGR	OKBM Afrikantov	Russian Federation	Conceptual Design
MHR-100	25 – 87	HTGR	OKBM Afrikantov	Russian Federation	Conceptual Design
A-HTR-100	50	HTGR	Eskom Holdings SOC Ltd.	South Africa	Conceptual Design
HTMR-100	35	HTGR	Steenkampskraal Thorium Limited	South Africa	Conceptual Design
PBMR-400	165	HTGR	PBMR SOC Ltd	South Africa	Preliminary Design
SC-HTGR	272	HTGR	AREVA	United States of America	Conceptual Design
Xe-100	35	HTGR	X-energy LLC	United States of America	Conceptual Design

PART 4: FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS					
4S	10	LMFR	Toshiba Corporation	Japan	Detailed Design
LFR-AS-200	200	LMFR	Hydromine Nuclear Energy	Luxembourg	Preliminary Design
LFR-TL-X	5~20	LMFR	Hydromine Nuclear Energy	Luxembourg	Conceptual Design
BREST-OD-300	300	LMFR	NIKIET	Russian Federation	Detailed Design
SVBR-100	100	LMFR	JSC AKME Engineering	Russian Federation	Detailed Design
SEALER	3	Small Lead Cooled	LeadCold	Sweden	Conceptual Design
EM²	265	GMFR	General Atomics	United States of America	Conceptual Design
SUPERSTAR	120	LMFR	Argonne National Laboratory	United States of America	Conceptual Design
WLFR	450	LFR	Westinghouse	United States of America	Conceptual Design
PART 5: MOLTEN SALT SMALL MODULAR REACTORS					
IMSR	190	MSR	Terrestrial Energy	Canada	Basic Design
CMSR	100-115	MSR	Seaborg Technologies	Denmark	Conceptual Design
CA Waste Burner	20	MSR	Copenhagen Atomics	Denmark	Conceptual Design
ThorCon	250	MSR	Martingale	International Consortium	Basic Design
FUJI	200	MSR	International Thorium Molten-Salt Forum: ITMSF	Japan	Experimental Phase
Stable Salt Reactor	37.5×8	MSR	Moltex Energy	United Kingdom	Conceptual Design
Stable Salt Reactor	300~900	MSR	Moltex Energy	United Kingdom	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
MCSFR	50	MSR	Elysium Industries	USA and Canada	Conceptual Design
PART 6: OTHER SMALL MODULAR REACTORS					
eVinci	0.2~15	Small Heat Pipe	Westinghouse	United States of America	Under Development

Part Two (Marine-based water-cooled SMRs) presents concepts that can be deployed in a marine environment, either under water or on a barge. This unique application provides many and more flexible deployment options, but also face many challenges if is to be deployed internationally, i.e. such as permission to cross national and international waters.

Part Three (High Temperature Gas Cooled SMRs) provides information on the modular type HTGRs under development and under construction. HTGRs provide high temperature heat (≥750°C) that can be utilized for more efficient electricity generation, a variety of industrial applications as well as for cogeneration.

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 IV) the four major fast reactor options were described. They are sodium cooled fast reactor (SFR), the heavy liquid metal-cooled (HLMC, i.e. lead or lead-bismuth) fast reactor, the gas-cooled fast reactor (GFR) and molten salt fast reactor (MSFR). In this booklet fast reactor designs with only the first three types of coolant are included, since none of the molten salt fast reactors are SMRs. The MSR designs included all have thermal neutron spectra and are included in part five.

Part Five (Molten Salt SMRs) presents the SMRs that utilize molten salt fuelled (and cooled) advanced reactor technology.

Part Six (Other SMRs) presents the SMRs that cannot be classified into any of the above categories.

In this booklet, effort has been made to present all SMR designs within the above categories. Each description includes a general design description and philosophy, target applications, development milestone, nuclear steam supply system, a table of the major design parameters, and then descriptions of the reactor core, engineered safety features, plant arrangement, design and licensing status. 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 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.

Annex I provides a summary as a function of different power ranges while Annex II 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 III for different non-electric applications.

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).

Other recent booklets published in support of ARIS are listed in Annex IV. Annex V contains a list of commonly used acronyms.

This booklet is a supplement to the IAEA Advanced Reactor Information System (ARIS, http://aris.iaea.org).

WATER COOLED
SMALL MODULAR REACTORS
(LAND BASED)

CAREM (CNEA, Argentina)

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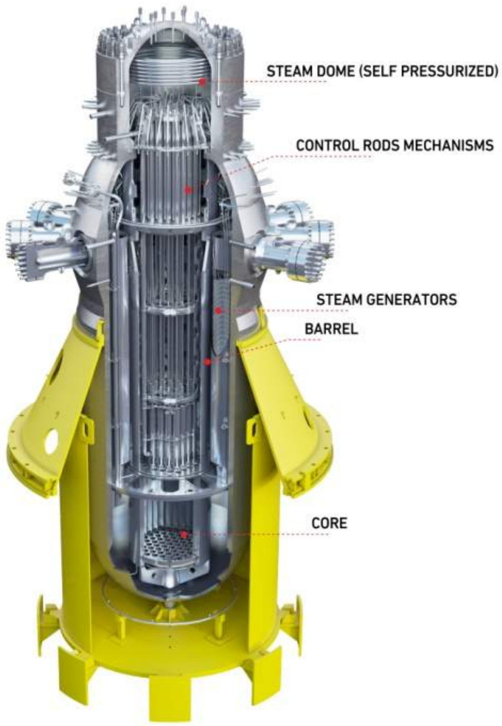


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNEA, Argentina
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	100/~30
Primary circulation	Natural circulation
System pressure (MPa)	12.25
Core inlet/exit temperatures (ºC)	284/326
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies	61
Fuel enrichment (%)	3.1% (prototype)
Fuel burnup (GWd/ton)	24 (prototype)
Fuel cycle (months)	14 (prototype)
Main reactivity control mechanism	Control rod driving mechanism (CRDM) only
Approach to engineered safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	Not available
RPV height/diameter (m)	11/3.2
RPV, internals and SGs weight (metric ton)	267
Seismic design	0.25
Distinguishing features	Core heat removal by natural circulation, pressure suppression containment
Design status	Under construction (as prototype)

1. Introduction

CAREM is a national SMR development project, based on LWR technology, coordinated by 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 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.

2. Target Application

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. Specific Design Features

(a) Design Philosophy

CAREM is a natural circulation based indirect-cycle reactor with features that simplify the design and improve safety performance. Its primary circuit is fully contained in the reactor vessel and it does not need any primary recirculation pumps. The self-pressurization is achieved by balancing vapour production and condensation in the vessel, without a separate pressurizer 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;

(b) Nuclear Steam Supply System

CAREM is an integral 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. 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.

(c) Reactor Core

The reactor core of CAREM-25 has fuel assemblies of hexagonal cross section. There are 61 fuel assemblies with 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. 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.

(d) Reactivity Control

Core reactivity is controlled using 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 for shut-down to produce a sudden interruption of the nuclear chain reaction when required.

(e) Reactor Pressure Vessel and Internals

Reactor pressure vessel (RPV) of CAREM-25 has a height of 11 meters and is 3.4 meters in diameter, having a variable thickness of 13 cm to 20 cm. The RPV is made of forged steel with an internal stainless steel liner.

(f) 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 consists of a system of 6 coiled piping layers, 52 parallel pipes of 26-m active 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. To achieve a nearly uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalized. The steam generators are designed to withstand the primary pressure without pressure in the secondary side and the entire secondary side is designed to withstand primary pressure up to isolation valves (including the steam outlet/water inlet headers) in case of SG tube breakage.

(g) 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 any pressure perturbations. Due to self-pressurization, the bulk temperature at core outlet corresponds to saturation temperature at primary pressure. In this way, typical heaters present in conventional PWR pressurizers are eliminated.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

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 a 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 when needed. Second Shutdown System consists of a gravity assisted high-pressure injection of borated water from 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.

(b) 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 extreme external events.

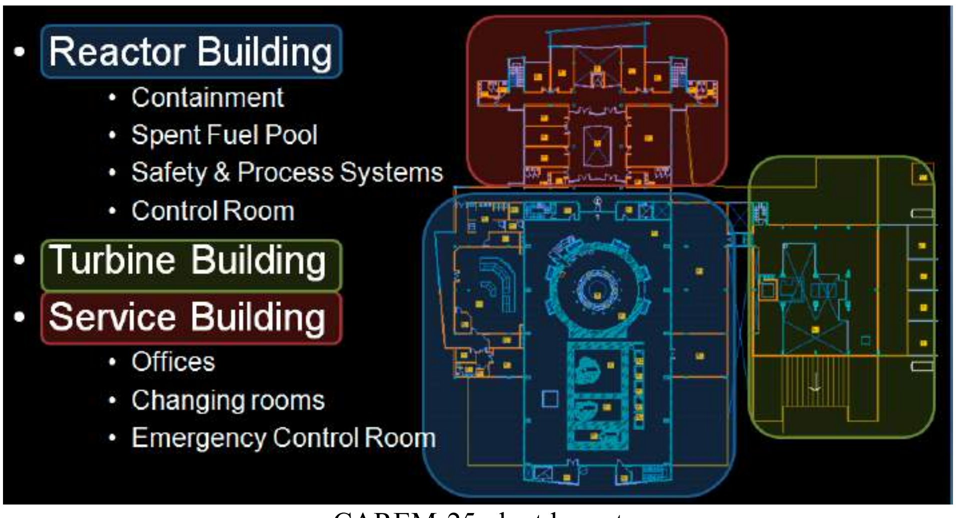
5. Plant Safety and Operational Performances

The natural circulation of coolant produces different flow rates in the primary system according to the power generated and 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.

6. Instrumentation and Control 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. 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.

7. Plant Arrangement



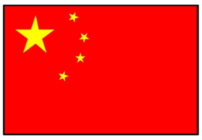
CAREM-25 plant layout

8. Design and Licensing Status

After obtaining the construction license, construction of the prototype started. Contracts with different Argentinean stakeholders and suppliers for manufacturing of components have already been signed. Environmental impact study was approved by the local authority. Non-nuclear buildings first concrete was poured in February 2014.

9. 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 (CAREM)
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
2022	First criticality



ACP100 (CNNC, China)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNNC(NPIC/CNPE) People's Republic of China
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	385/125
Primary circulation	Forced circulation
System pressure (MPa)	15
Core inlet/exit temperatures (^oC)	286.5/319.5
Fuel type/assembly array	UO_2/17x17 square pitch arrangement
Number of fuel assemblies	57
Fuel enrichment (%)	<4.95
Fuel burnup (GWd/ton)	<52000
Fuel cycle (months)	24
Main reactivity control mechanism	Control rod drive mechanism (CRDM), Gd_2O_3 solid burnable poison and soluble boron acid
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	200000
RPV height/diameter (m)	10/3.35
Module weight (metric ton)	300
Seismic design	0.3
Distinguishing features	Integrated reactor with tube-in-tube once through steam generator, nuclear island underground
Design status	Basic design finished

1. Introduction

The ACP100 is an integrated PWR design developed by China National Nuclear Corporation (CNNC) to generate an electric power of 125 MW(e). The ACP100 is based on existing PWR technology adapting verified passive safety systems to cope with the consequences of accident events; in case of transients and postulated design basis accidents the natural convection cools down the reactor. The ACP100 integrated design of its reactor coolant system (RCS) enables the installation of the major primary circuit's components within the reactor pressure vessel (RPV).

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. Specific Design Features

(a) Design Philosophy

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

(b) Nuclear Steam Supply System

The integrated nuclear steam supply system (NSSS) design consists of the reactor core, and sixteen (16) once-through steam generators (OTSG). The four (4) canned motor pumps are installed nozzle to nozzle to the RPV.

(c) Reactor Core

The 57 fuel assemblies (FAs) of ACP100 core with total length of 2.15 m core have a squared 17x17 configuration. The fuel ^{235}U enrichment is about 1.9–4.95%. The reactor will be able to operate 24 months at balance fuel cycle.

(d) Reactivity Control

The reactivity is controlled by means of control rods, solid burnable poison and soluble boron dissolved in the primary coolant. There are 21 control rods, with a magnetic force type control rod driving mechanism (CRDM).

(e) Reactor Pressure Vessel and Internals

The RPV and equipment layout are designed to enable the natural circulation between reactor core and steam generators. The RPV is protected by safety relief valves against over-pressurization in the case of strong difference between core power and the heat removed from the RPV. The internals not only support and fasten the core but also form the flow path of coolant inside RPV.

(f) Reactor Coolant System

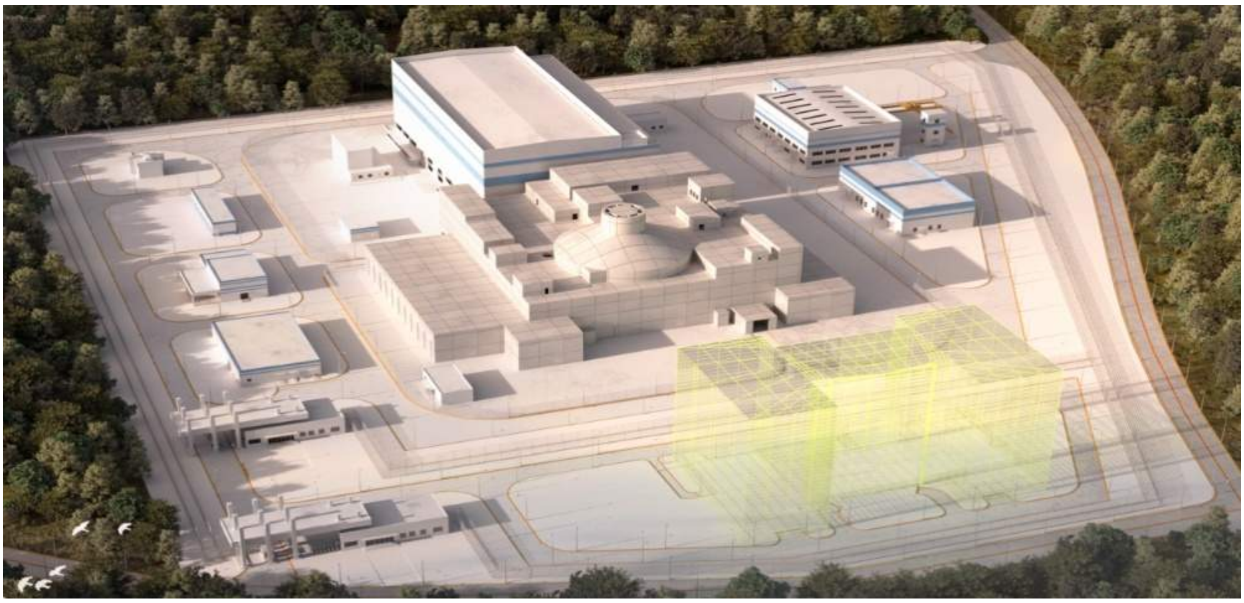
The ACP100 primary cooling mechanism under normal operating condition and shutdown condition is done by forced circulation. The RCS has been designed to ensure adequate cooling of reactor core under all operational states, during and following all postulated off normal conditions. The integral design of RCS significantly reduces the flow area of postulated small break LOCA.

(g) Steam Generator

There are 16 OTSGs, which are mounted within the RPV. All 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 steam cavity.

(h) 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.



ACP100 demonstration NPP aerial view

4. Safety Features

The ACP100 is designed with inherent and passive safety features, eliminating large bore primary coolant piping which in turns eliminates large break LOCA. The passive safety system mainly consists of the passive decay heat removal system (PDHRS), passive emergency core cooling system (ECCS), passive containment air cooling system (PAS) and reactor automatic depressurization system (RDP).

(a) 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 arranging the nuclear steam supply system and spent fuel pool underground. When the spent fuel pool is filled with spent fuel of 10 years, the pool cooling water can 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 preventing RPV melt, passive hydrogen recombination system preventing containment hydrogen explosion and maintaining the containment integrity after severe accidents, automatic pressure relief system and RPV off-gas system to remove non-condensable gas gathered at RPV head after accidents.

(b) 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 ACP100 consists of one emergency cooler and associated valves, piping, and instrumentation. The emergency cooler is located in the containment in-refuelling water storage tank, which provides the heat sink for the emergency cooler. The decay heat is removed from the core by natural circulation. The PDHRS provides core cooling for seven (7) days without operator intervention or long term with IRWST makeup water collected by gravity force from the steam condensed in containment.

(c) Emergency Core Cooling System

The emergency core cooling system (ECCS) consists of two coolant storage tanks (CST), two safety injection tanks (SIT), an in-refuelling water storage tank (IRWST) 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. After LOCA accidents, the steam in containment is condensed continuously at containment internal face thus the heat is conducted to containment which is cooled by PAS, thus ensuring the containment integrity.

(d) In-Refuelling Water Storage Tank (IRWST)

The IRWST is a passive water tank, resting on the internal structure base slab. During refuelling operations, it provides water for refuelling cavity, internals storage compartment and refuelling transfer canal to complete the refuelling operation. Under the condition of LOCA and the steam pipe rupture, it provides water for emergency reactor core cooling. In the severe accidents, water in it floods the internal structure under the balanced water level due to gravity. During the operation of reactor automatic depressurization system, it absorbs the sprayed steam from the RCS. During the operation of the passive residual heat removal cooler, it works as the heat sink of the passive residual heat removal system.

(e) Reactor Pool

The reactor pool is used during refuelling operation or inspection of reactor internals. The reactor pool consists of two compartments which can be separated by bulkhead: reactor cavity and internals storage compartment adjacent to the reactor.

(f) Containment System

The ACP100 containment houses the RCS, the passive safety systems and the auxiliary systems. ACP100 adopts small steel containment cooled by air with no need of drive signal.

5. Plant Safety and Operational Performances

Nuclear safety is always the first priority. 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 it as low as reasonably achievable. Accident prevention measures ensure that radioactive consequences are lower than limited dose in terms of all of the considered accident sequences and even in the unlikely severe accidents, mitigation of accidents' induced influences can be ensured by implementing emergency plan. The design of ACP100 incorporates operational experience of the state of the art design. Proven technology and equipment are adopted as much as reasonably possible.

6. Instrumentation and Control Systems

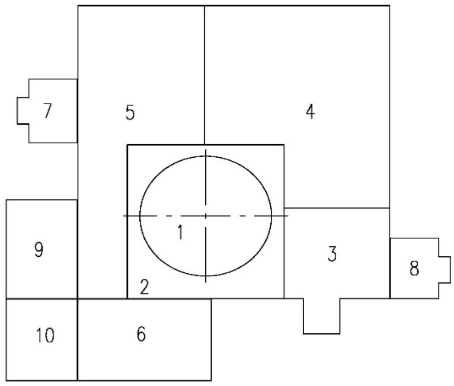
The Instrumentation and Control (I&C) system designed for ACP100 is based on defence in depth concept, compliance with the single failure criterion and diversity. The diversity in the design of I&C system is achieved through: (1) different hardware and software platforms for 1E and N1E I&C, (2) reactor 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.

7. Plant Arrangement

The ACP100 adopts compact single-unit plant layout, including nuclear island building (NI) and turbine generator building (CI). The fuel building, electrical building and the nuclear auxiliary building are arranged around the reactor building, allowing the NI building working well with a smaller size. This layout can adjust itself well to various kinds of plant sites. The operation platform of the reactor, the operation platform of fuel and the transportation platform of the radioactive waste are arranged around the ground of the power plant, which simplify the transportation of the fuel, radioactive waste and big equipment in the NI, to lower the using frequency of NI hoists as well as the cost of construction.

The turbine generator building (CI)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.

1. Reactor Building
2. Connecting Building
3. Fuel Building
4. Electrical Building
5. Nuclear Auxiliary Building
6. Access Building
7. Emergency Diesel Generator Building
8. Auxiliary Diesel Generator Building
9. Fire Protection Pump Station for NI
10. Emergency Compressor House



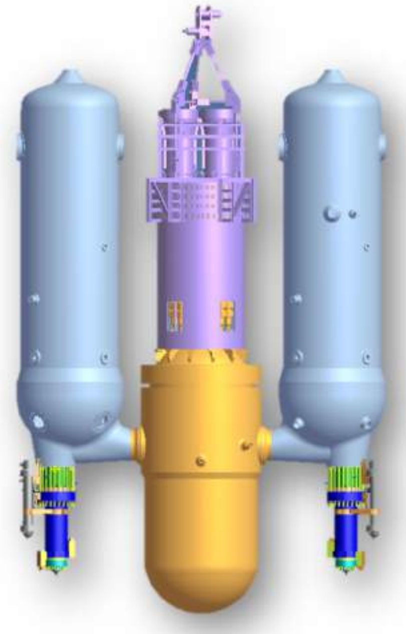
8. Design and Licensing Status

The ACP100 engineering basic design is close to completion and a preliminary safety assessment report (PSAR) is finished. Passive emergency core cooling system, control rod drive system, and critical heat flux have been tested. Passive containment air 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 site stage environmental impact assessment report to China’s regulator are supposed to be submitted for review. In April 2016, an agreement to conduct a generic safety review for ACP100 was signed between the IAEA and CNNC. An industrial demonstration plant with one 385 MW(t) unit is planned in Hainan Province, China.

9. Development Milestones

2016	Generic reactor safety review for ACP100 by IAEA finished.
2017	CNNC signed an agreement with the Changjiang municipal government in Hainan Province to host the first of a kind (FOAK) ACP100 demonstration unit.
2018	Preliminary safety assessment report (PSAR) finished.

CAP200 (SNERDI/SNPTC, China)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	SNERDI/SNPTC, People's Republic of China
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	660/>200
Primary circulation	Forced circulation
System pressure (MPa)	15.5
Core inlet/exit temperatures (^{o}C)	289/313
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies	89
Fuel enrichment (%)	4.2 (average)
Fuel burnup (GWd/ton)	37 (average)
Fuel cycle (months)	24
Main reactivity control mechanism	Control rod drive mechanism and soluble boron
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	--
RPV height/diameter (m)	8.845/3.280
Seismic design	0.30 (g)
Distinguishing features	Compact layout; passive safety; underground containment
Design status	Conceptual design finished

1. Introduction

The 200 MW(e) China advanced passive pressurized water reactor (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 of 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.

2. Target Application

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

3. Specific Design Features

(a) Design Philosophy

CAP200 is a small PWR designed with improved safety, flexibility and environmental friendliness, and is comparable with other SMRs on economy. Compared with large PWRs, CAP200 has several 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 probability 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 result in an 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 to withstand natural disasters.

(b) Nuclear Steam Supply System

The NSSS of CAP200 consists of reactor pressure vessel (including IHP), steam generator, reactor coolant pump, pressurizer and auxiliary systems. Due to the lack of RCS piping, the radioactivity containment capability of CAP200 is better than traditional PWR.

(c) Reactor Core

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 to execute load-follow operation.

(d) Reactivity Control

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.

(e) Reactor Pressure Vessel and Internals

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 CAP200, to connect the RPV and SGs replacing main pipes. The vessel is cylindrical with a removable flanged hemispherical upper head and hemispherical lower head. The area with highest neutron fluence in the active core region of the vessel is completely forged and free of welding to enhances the confidence in the 60 years’ design life and simplifies 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. 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.

(f) Steam Generator

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 the SG is 6.06 MPa and 183.6 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.

(g) Pressurizer

The pressurizer of CAP200 is a typical steam-type one with an 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 to provide 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, will not be needed because of the improved transient response by using large volume pressurizer.

(h) Reactor Coolant Pump

Either a leak-tight canned motor pump or a wet coil pump are possible choices for the RCP of CAP200. There is rich experience in 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 65 m at design flow rate of 12000 m^3/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.

(i) Electric Power Systems

Electric power system has three levels to satisfy the defence-in-depth requirement. The first level is the AC power supply for plant operation load which is supplied by main generator and off-site power sources. The second level is the AC distribution system and the non-Class 1E DC and UPS system for the permanent non-safety loads. The third level is the Class 1E DC and UPS system, which is used to supply safety related loads essential to emergency reactor shutdown, containment isolation and other essential function during design basis accident.

(j) Conventional Island Systems

The system configuration and general layout of CAP200 are similar to traditional NPP. Considering the CAP200 is close to the user, the turbine is designed to be capable of different types of extraction, and enables CAP200 to meet different levels of heating and industrial steam supply requirements.

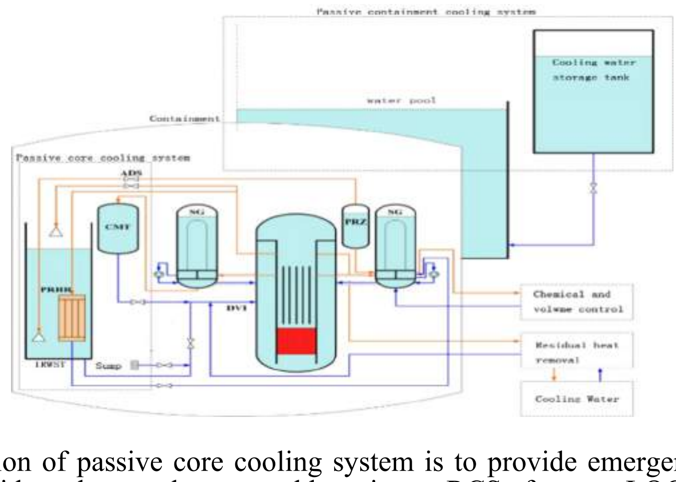
4. Safety Features

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, a passive core cooling system and passive containment cooling system are adopted.

(a) 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.

(b) 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 to as the passive core cooling system.

(c) 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. 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.

5. 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 centres 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.

6. 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.

7. Plant Arrangement

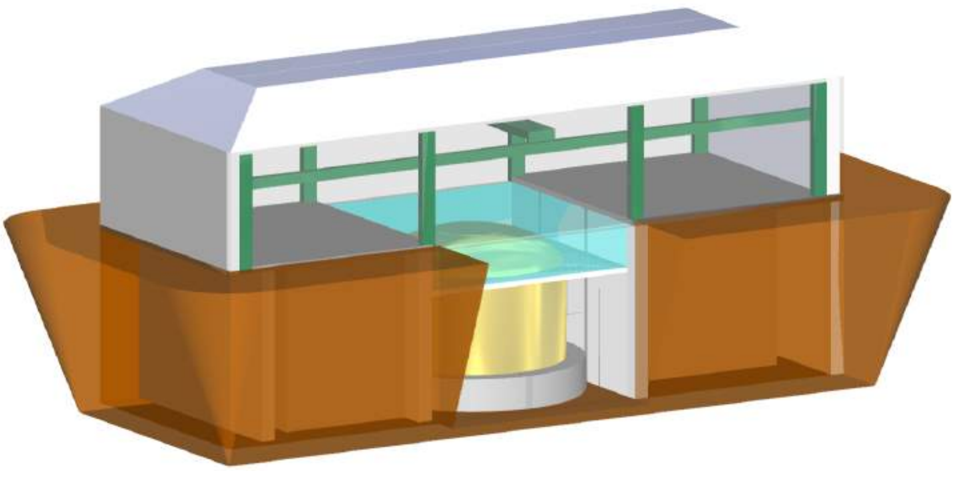
(a) Reactor Building

The reactor building is placed underground, which is suitable for various geological conditions. It can adapt to the site conditions of most potential sites and islands.

(b) Nuclear Island Layout

The size of nuclear island is minimized by both system simplification and adoption of passive design. Compact RCS especially eliminates main pipes and allows reactor building to be smaller.

The number of system component is reduced by adoption of large capacity equipment, common use of single equipment for different systems, for example, replacing the polar crane by a travelling crane to use both for reactor building an auxiliary building, etc.



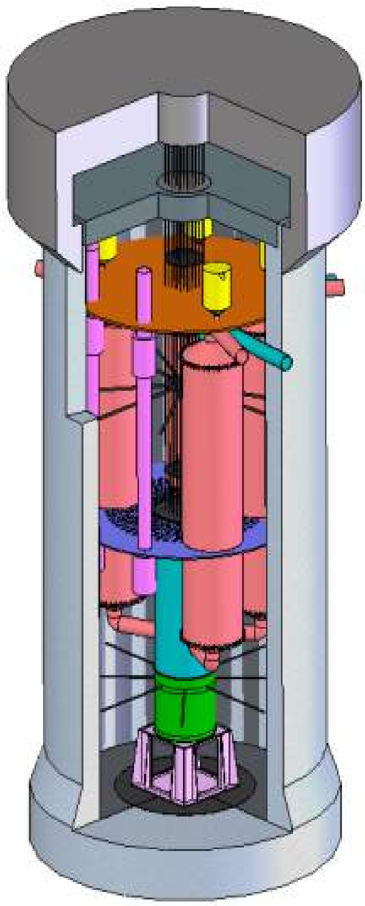
8. Design and Licensing Status

Conceptual design finished.

9. Development Milestones

2014	Conceptual design started
2015	Conceptual design finished

DHR (CNNC, China)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	China National Nuclear Corporation (CNNC), People's Republic of China
Reactor type	Pool type reactor
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	400/none
Primary circulation	Forced circulation
Core inlet pressure (MPa)	0.3
Core inlet/outlet temperatures (°C)	68/98
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies	69
Fuel enrichment (%)	<5.0
Fuel burnup (GWd/t)	30
Fuel cycle (months)	10
Main reactivity control mechanism	Control rod driving mechanisms
Design life (years)	60
Plant footprint (m^2)	40000
Pool depth/diameter (m)	26/10
Seismic design	0.3g
Distinguishing features	Coupling with desalination and radioisotope production
Design status	Basic design

1. Introduction

The district heating reactor (DHR400) is a pool type reactor with a thermal power of 400 MW. The DHR400 operates under low temperature and atmospheric pressure. Established light water reactor technologies are used in the design of the DHR400. Inherent safety features are incorporated to enhance its safety and reliability. Significant design features include: Large water inventory in the pool that provides great thermal inertia and a long response time, thus enhances the resistance to system transients and postulated accidents. Very low possibility of core meltdown, which eliminates the probability of large radioactivity release and simplifies the off-site emergency response. Simplification in designs and convenience in maintenance lead to the improvement in economics. Due to the high reliability and inherent safety features, DHR400 can be located in the immediate vicinity of the targeted heating supply area.

2. Target Application

DHR400 is a multi-purpose application reactor for district heating, sea water desalination and radioisotope production. The reactor is designed to replace traditional small regional heating plants using coal to minimize waste and reduce air pollution in northern parts in China.

3. Specific Design Features

(a) Design Philosophy

DHR400 is a pool type reactor, designed based on the pool type research reactor. It operates under low temperature and with atmospheric pressure above the water surface. With the large of the depth of the pool, appropriate core outlet water temperature is achieved by the static pressure of the water layer. The design configurations preclude the loss of coolant accident (LOCA), control rod ejection accident and the loss of decay heat removal capability. DHR400 is moderated by low pressure water, which via the negative temperature reactivity coefficient, assures inherent reactor shutdown at abnormal situations. The DHR400 design also adopts proven pool type reactor technology, design simplification, convenient maintenance, highly automation and reduced operators to obtain enhanced safety and improved economics.

(b) Reactor Core

The core of the DHR400 consists of 69 fuel assemblies. Each fuel assembly is 2.1 m long and its design is modified from a standard 17 × 17 PWR fuel assembly with 264 fuel rods, and 9 supporting tubes. The fuel is UO_2 with Gd as a burnable absorber. The ^{235}U enrichment is below 5 percent. The reactor operates 150 days per year (typical 5-month winter period in northern China) and a three-batch refuelling is conducted off power on a 10-month refuelling cycle.

(c) Reactivity Control

Reactivity control during normal operations is achieved through control rods. Two separate shutdown systems are used for reactor shutdown and scram events. Fuel rods containing different mass ratios of gadolinium oxide are introduced in all fuel assemblies to maintain reactivity during burnup and for flattening the axial power profile.

(d) Reactor Pool and Internals

The reactor pool is a cylinder with an inside diameter of 10 m and an overall height of 26 m, containing the core structure, core shroud, four attenuation barrels, four inertial tanks, the residual heat removal system, the core supporting foundation and the seismic stabilizer brackets inside its 25 m depth of water. The pool is buried underground with an elevation of its bottom of -26 m. The pool is made of reinforced concrete with an inner layer of 5 mm stainless steel and an outer layer of 10 mm carbon steel. The thickness of the surrounding concrete layer is 1.0 m and the bottom plate is 2 m thick. The upper head includes a carbon steel truss and a stainless steel plate, connected to the concrete wall of the pool and provides support for the control rod driven mechanism and the control rod guide tubes. One meter below the upper head there is a gaseous space, which is connected to an engineered venting system to exhaust vapour and other gases. Above the reactor pool there is a 2 m thick movable reinforced concrete plate. The overall structure of the reactor pool provides great resistance to external events including airplanes. The large water inventory in the pool water provides large thermal inertia and a long response time, thus enhances the resistance to system transients and accidents. These features ensure that the core will not melt down under any accident.

(e) Primary Coolant System

During normal operations, the 68^0C inlet water enters the core from the bottom, and the water is then heated to a temperature of 98^0C. The outlet water enters the 4 attenuation barrels through the rising barrel right above the core. The water rises slowly in the attenuation barrels and then enters four separate primary pump room through the pool wall. The water entering the primary pump room flows into the primary heat exchanger through two electric isolating valves. Inside the primary heat exchanger, the primary water is cooled to 68^0C by the secondary side water and then pumped back to the upper part of the reactor pool through another two electric isolating valves. The water then flows down to the bottom and enters the core again. No boiling will occur during normal operations.

(f) Secondary Coolant System

The secondary coolant system is an isolated sealed intermediate loop that separates the primary loop and the third loop while transfer heat from the primary to the third. The pressure of the secondary loop is designed to be higher than that of the primary loop, so there is no chance that the radioactive primary side water will contaminate the third loop.

(g) Primary Heat Exchanger

DHR400 uses 8 plate heat exchangers in its primary coolant system to transfer heat to the secondary loop. Plate heat exchanger is suitable for low temperature difference water to water heat exchange for its small resistance and high efficiency. The leak tightness of the plate heat exchanger is considered to be highly reliable. Even under the circumstances of leakage, the coolant leaks outwards to the pump room. This feature provides great advantages to radioactivity isolation.

(h) Residual Heat Cooling System

The residual heat cooling system of DHR400 is consists of two parts, a 2.4 MW in-pool natural circulation cooling system and a 4 MW out-pool forced circulation cooling system. The temperature of the reactor pool water is kept below boiling point after shutdown and a temperature of 40^0C can be achieved with the residual heat cooling system.

4. Safety Features

The DHR400 is designed with inherent safety features. These include a large volume of water in the reactor pool, two sets of reactor shutdown systems, pool water cooling system and a decay heat removal system. With these designs stable long-term core cooling under all conditions can be achieved.

(a) *Inherent Safety Features*

Instead of augmenting additional engineered safety systems the DHR400 emphasise on inherent safety features. The great heat capacity of the 1800 tons of water inside the reactor pool ensures that the reactor core will be kept submerged in all circumstances, thus no core melt down could occur. It has negative temperature and void reactivity feedback, therefore the power increase can be effectively restrained. In the event of severe accident, the reactor can automatically shutdown by the inherent negative reactivity feedback, and the reactor core will be kept submerged for as long as 26 days even with no further intervention.

(b) *Containment System*

There are four barriers precluding a radioactive release to the environment in DHR, including the fuel coating, the reactor pool, the earth around the pool and the reactor building on top of the pool. Due to the low operating temperature and atmospheric pressure on the top of the reactor pool, there are no high-pressure events and instead of a containment, a confinement building is sufficient for protection. The location of the reactor assembly below ground and submerged in 1800 tons of water makes DHR400 highly resistant to external events including aircraft crashes. Additional protection is provided by the reactor building above the pool.

(c) *Waste Gas Treatment System*

A waste gas treatment system is introduced to store and decay short life gaseous fission products and to absorb krypton and xenon isotopes.

5. Plant Safety and Operational Performances

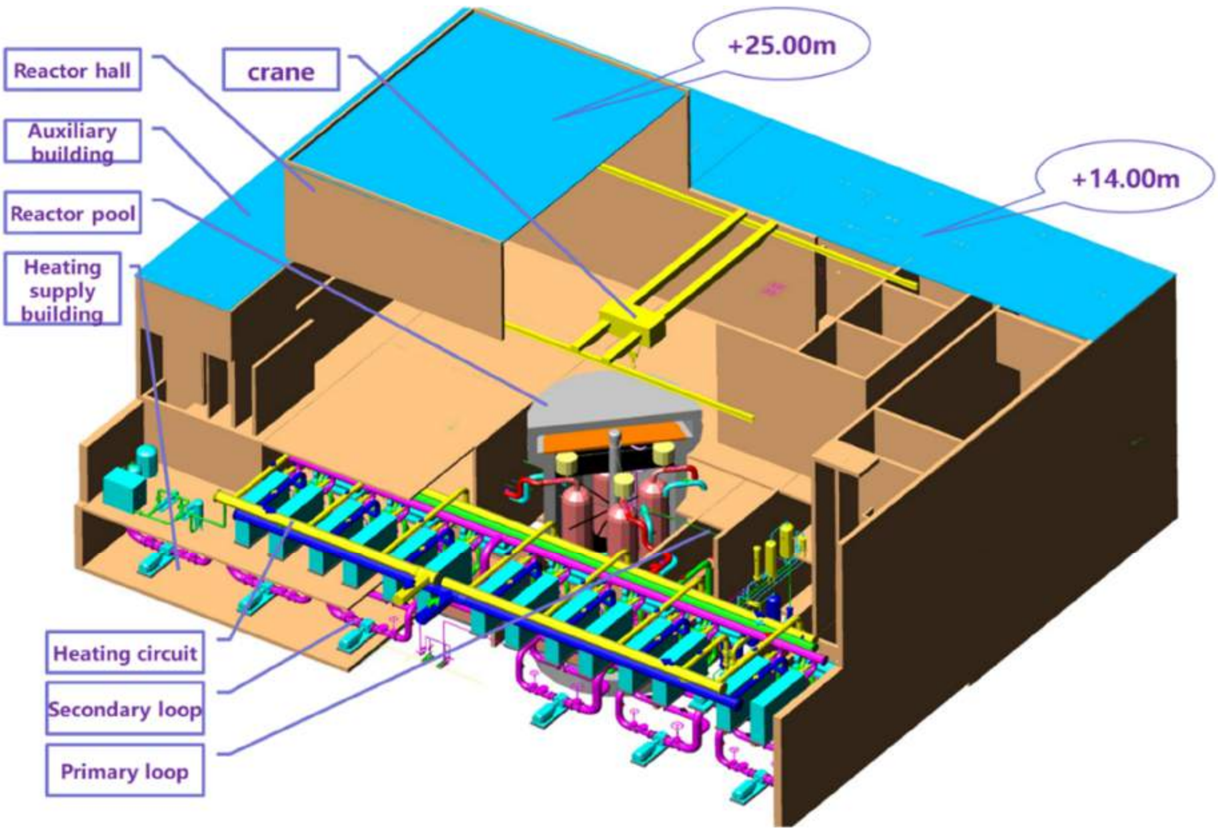
Two independent systems are provided for reactor power control and to ensure safe reactor shutdown. Reactor cold start-up and rapid start-up can be achieved safely due to the negative temperature reactivity coefficient.

6. Plant Arrangement

The layout of the DHR400 is illustrated below. The principle structures are the reactor main building, the cooling water pump room, the exhaust tower, the desalination building, the treated water room, the chlorine production room, non-radioactive oily sewage treatment room and the auxiliary building. The reactor is seismically and tsunami qualified in accordance with site conditions.

(a) *Reactor Building*

The reactor building consists of a sealed hall and auxiliary buildings including heating supply room, radioactive waste storage and treatment room and others. The layout is illustrated in the following figure. The reactor building is equipped with closed circuit monitoring system to oversee and protect the area.



(b) Balance of Plant

i. Heating Supply Building

The heating supply system is located in the heating supply building. The secondary system transfers heat to the heating supply system.

ii. Desalination Systems

Sea water desalination systems are used in conjunction with the secondary system.

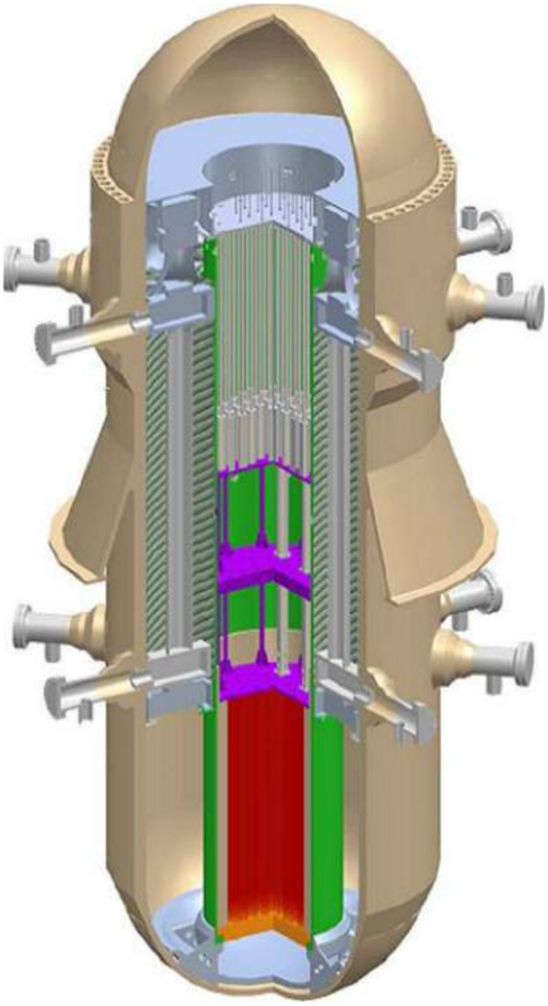
7. Design and Licensing Status

DHR400 is finalizing the preliminary design (some parameters might change with the optimization of DHR400 design) and seeking for construction license in early 2019. DHR400 has a target commercial operation date of 2021 for the first plant that is expected to be built in Xudapu, Liaoning, China.

8. Development Milestones

2019	Construction licence
2021	Commercial operation

IRIS (IRIS, International Consortium)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	IRIS
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	1000/335
Primary circulation	Forced circulation
System pressure (MPa)	15.5
Core inlet/exit temperatures (^{o}C)	292/330
Fuel type/assembly array	UO_2/MOX/17x17 square
Number of fuel assemblies	89
Fuel enrichment (%)	4.95
Fuel burnup (GWd/ton)	65 (max)
Fuel cycle (months)	48 (max)
Main reactivity control mechanism	ICRDM (Internally driven control rods)
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	14000 (four units layout)
RPV height/diameter (m)	21.3/6.2
Seismic design	0.30 (g)
Distinguishing features	Integral primary system configuration
Design status	Basic design

1. Introduction

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

2. Target Application

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

3. Specific Design Features

(a) Design Philosophy

IRIS is designed to provide enhanced safety, improved economics, proliferation resistance and waste minimization.

(b) Nuclear Steam Supply System

All of 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.

(c) 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 control rods. Low-power density is achieved by employing a core configuration consisting of 89 fuel assemblies with a 14-ft (4.267 m) 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 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.

(d) 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 nozzle welds and seals.

(e) Reactor Pressure Vessel and Internals

The IRIS reactor vessel (RV) 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 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.

(f) 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 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, 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.

(g) 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, where 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 as well as the emergency heat removal system (EHRS) are designed for the full primary pressure of 15.5 MPa. The EHRS does not inject water, but only removes heat from the reactor via the SGs.

(h) Pressurizer

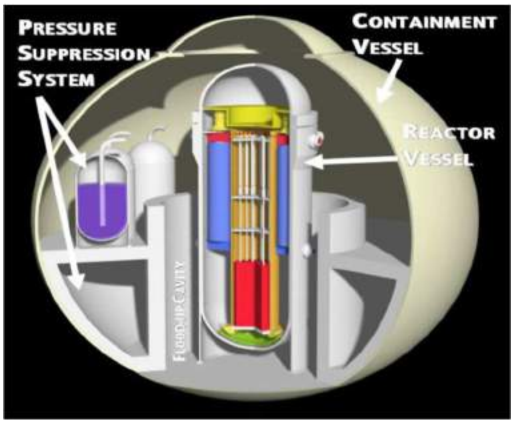
The IRIS pressurizer is integrated into the upper head of the reactor vessel. The pressurizer region is defined by an insulated, inverted top-hat structure that divides the circulating reactor coolant flow path from the saturated pressurizer water. This structure includes a closed cell insulation to 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 m3, which includes a steam volume of 49 m3.

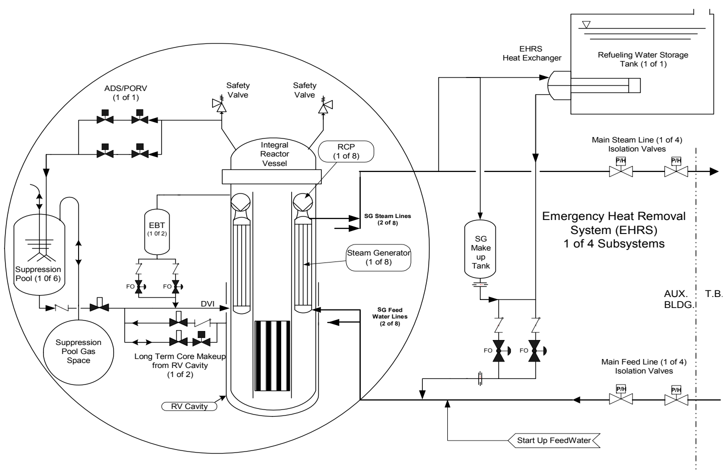
4. Safety Features

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. The auxiliary building is fully seismically isolated. This provides a high level of defence in depth (DID) 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 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.

(a) 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 refuelling 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.





(b) 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. 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 refuelling cavity above the containment and RV is then flooded, and the RV internals are removed and stored in the refuelling cavity. Fuel assemblies are vertically lifted from the RV directly into a fuel handling and storage area, using a refuelling machine located directly above the CV. Thus, no refuelling equipment is required inside containment and the single refuelling machine is used for all fuel movement activities. 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.

5. Plant Safety and Operational Performances

The IRIS design provides for multiple levels of defence 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 refuelling 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.

6. Plant Arrangement

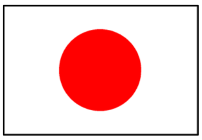
Almost half of the IRIS containment vessel is located below ground, thus leaving only about 15 m above the ground (i.e., several times less than the containment of a large LWR). This very low profile makes IRIS an extremely difficult target for aircraft flying terrorists; in addition, the IRIS containment 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.

7. Design and Licensing Status

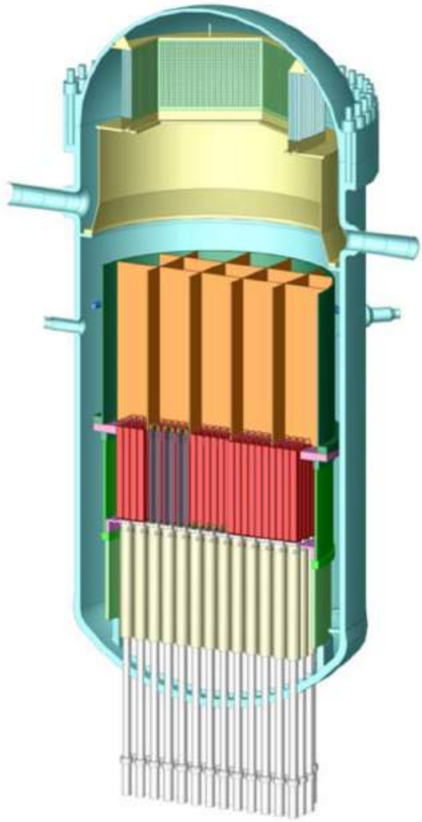
The IRIS team has completed the design of the large scale test facility, currently under construction, to prepare for future design certification. R&D activities in the field of design economics, financial risk and SMR competitiveness are under way.

8. Development Milestones

2001	Conceptual design completion
2001	Preliminary design start-up
2002	Pre-licensing process activities
2009	Integral testing facility construction



DMS (Hitachi-GE Nuclear Energy, Japan)



MAJOR TECHNICAL PARAMETERS

Parameter	**Value**
Technology developer, country of origin	Hitachi-GE Nuclear Energy Japan
Reactor type	Boiling water reactor
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	840/about 300
Primary circulation	Natural circulation
System pressure (MPa)	7.17
Core inlet/exit temperatures (^oC)	186/287
Fuel type/assembly array	UO_2 pellet/10x10 square configuration with channel box
Number of fuel assemblies	400 (short length fuel assembly)
Fuel enrichment (%)	< 5
Fuel burnup (GWd/ton)	< 60
Fuel cycle (months)	24
Main reactivity control mechanism	Control rod drive
Approach to engineered safety systems	Hybrid (passive + active) system
Design life (years)	60
Plant footprint (m^2)	--
RPV height/diameter (m)	15/4.8
Seismic design	0.45 (g)
Distinguishing features	Simple reactor design, natural circulation system, hybrid safety system, multipurpose energy use
Design status	Basic design

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(t) 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 enables the modules to be transported to the site.

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. Specific Design Features

(a) 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.

(b) 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. This system feature contributes to remove a potential of loss of coolant accident (LOCA). Also, all the large sized piping connected to the RPV are 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 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.

(c) 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, 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.

(d) 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 kinds of independent operation mode, one is fine motion control by electric motor, and the other is rapid motion (scram) by hydraulic pressure. In the normal operation, the reactivity in the core is controlled by the fine motion feature of FMCRD.

(e) 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 greatly influences construction cost of the reactor building. The small size RPV in the DMS reactor is achieved by simplifying core internals through small size 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 44 MW/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.

(f) Reactor Coolant System

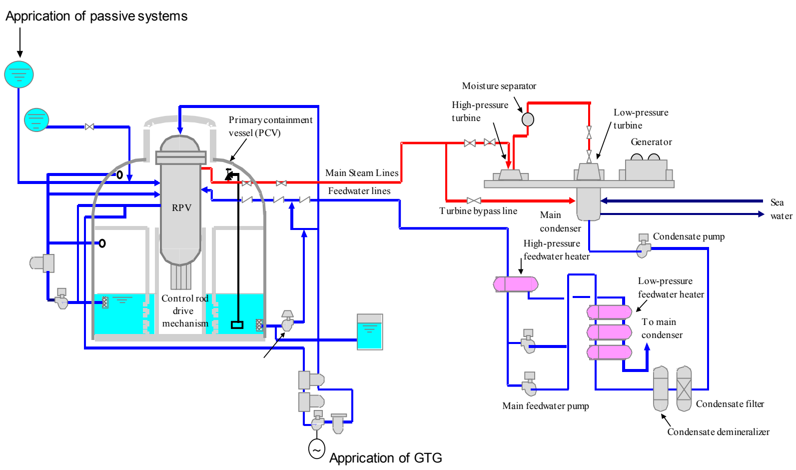
The DMS reactor primary cooling mechanism under normal operating condition and shutdown condition is by natural circulation of coolant. In order to enhance natural circulation driving force, divided chimney of about 3 m height is placed above the core. The reactor coolant system (RCS) is designed to ensure adequate cooling of reactor core under all operational states, and during and following all of the 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.

4. 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.

(a) 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 1200^0C 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.

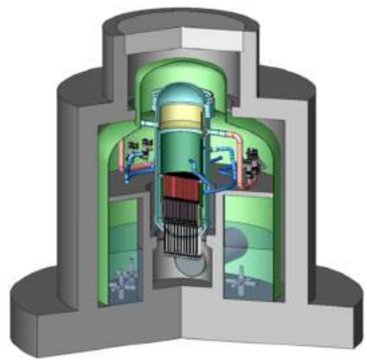


(b) Decay Heat Removal System

The residual heat removal system (RHR) includes 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.

(c) 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 m compared with 3.7 m 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 passive autocatalytic re-combiners (PARs) are planned in the PCV to react hydrogen and oxygen generated due to water radiolysis.



5. 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 the 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.

6. Instrumentation and Control Systems

The digital I&C systems (e.g. microprocessors, field programmable gate arrays (FPGAs)) are mainly adopted to the DMS, which makes 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.

7. Plant Arrangement

(a) 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. are designed as flexible areas. By this approach of rationalized layout, it is possible to realize that the building volume per unit output power is equivalent to ABWR.

(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.

i. Turbine Generator Building

The structure of turbine systems is 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.

ii 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.

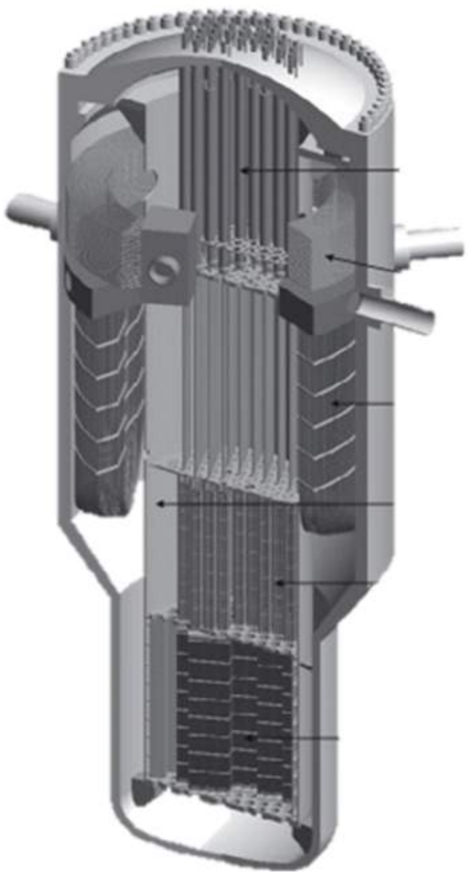
8. Design and Licensing Status

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. Some SMR designs 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).

9. 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

IMR (Mitsubishi Heavy Industries, Japan)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Mitsubishi Heavy Industries, Japan
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	1000/350
Primary circulation	Natural circulation
System pressure (MPa)	15.51
Core inlet/exit temperatures (°C)	329/345
Fuel type/assembly array	UO_2 pellet/21x21 square
Number of fuel assemblies	97
Fuel enrichment (%)	4.8
Fuel burnup (GWd/ton)	> 40
Fuel cycle (months)	26
Main reactivity control mechanism	Control rods drive mechanism
Approach to engineered safety systems	Hybrid (Passive + Active) system
Design life (years)	60
Plant footprint (m²)	--
RPV height/diameter (m)	17/6
Seismic design	0.3 (g)
Distinguishing features	Integral PWR with natural circulation; employs two in-vessel circular steam generators: one in lower region of the vessel, one in upper region.
Design status	Conceptual design completed

1. Introduction

The integrated modular water reactor (IMR) is a medium sized power reactor with a reference output of 1000 MW(t) 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 the primary means of reactivity control. These design features allow the elimination of the emergency core cooling system (ECCS).

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 the capacity of grids is small. IMR can also be used for cogeneration of electricity and district heating, seawater desalination, process steam production and so forth. IMR adopts structures, systems and equipment that require no large-scale development. This is meant to facilitate in obtaining regulatory authority's approval for the construction and operation of power plant.

3. Specific Design Features

(a) Design Philosophy

The IMR is an integral type water cooled reactor where the primary system components are all installed inside the reactor pressure vessel (RPV). Main coolant piping and primary coolant pumps are eliminated by adopting integrated natural circulation system. Pressurizer is eliminated by adopting the self-pressurization system. There are two types of steam generators (SGs) inside the RPV; one is located in the vapour portion in the upper region of the RPV and the other is located in the liquid portion in the RPV's lower region. The SGs are also used as decay heat removal heat exchangers during accidents and normal startup and shutdown operations. Hence to eliminate the need of ECCS, the SGs serve as a passive safety system that do not require any external power. The IMR is a PWR with moderation ratio 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 measures 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.

(b) 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.

(c) Reactor Core

The IMR core consists of 97 fuel assemblies in 21×21 array with an average enrichment of 4.95 % and produces a thermal output of 1,000 MW(t). 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 an average discharged burnup of 46 GWd/ton can be attained, which is approximately the same as in current PWRs. The cladding material is 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 one-third to one-half of that for a conventional PWR. The design-refuelling interval is three (3) years in three (3) batches of fuel replacement. The fuel rod design is the same as that for a conventional PWR.

(d) Reactivity Control

The chemical shim reactivity control is not implemented in the IMR, rather both control rods that contain enriched ^{10}B and burnable absorbers control the whole reactivity. Control rods with 90 wt% enriched B_4C neutron absorber 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 of the fuel to reduce axial differences in 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.

(e) Reactor Pressure Vessel and Internals

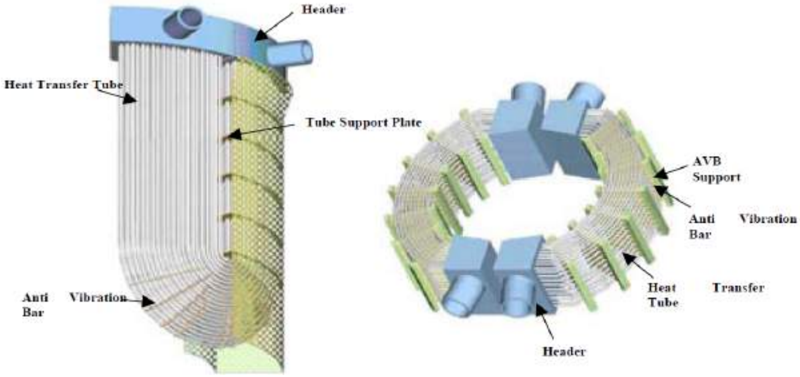
The upper part of the RPV inside diameter is about 6 m in order to accommodate the in-vessel SGs. The inside diameter of the lower part of the RPV is reduced to about 4 m in order to minimize the cold-side water volume. In order to eliminate the necessity for the consideration of LOCA, the largest diameter 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.

(f) 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 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.

(g) Steam Generator

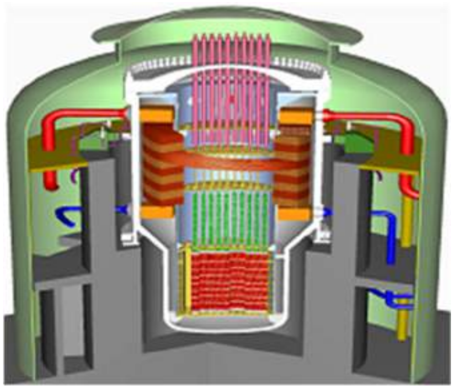
There are two types of SGs adopted in the IMR. One is the SG in vapour (SGV), which is located above the water level in the RPV. The energy transported by vapour formation generates secondary steam through SGV. In addition, since the vapour in the RPV is condensed by SGV, controlling the feedwater flow 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. 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.

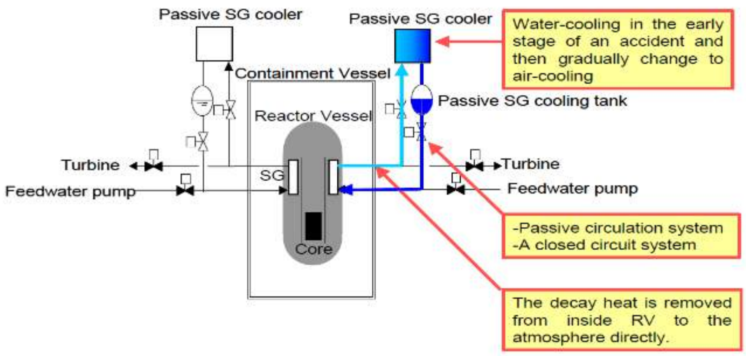


(h) Pressurizer

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

4. Safety Features





(a) 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 systems are required in IMR. 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.

(b) Decay Heat Removal System

For the mitigation system, the SDHS is activated. 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 the plant safety.

(c) Containment System

A small containment vessel (CV) is achieved due to the integrated primary system and simplified auxiliary systems. The IMR adopts reinforced concrete containment. A higher design pressure of the containment is planned 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 is expected 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.

5. 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.

6. Plant Arrangement

(a) Reactor Building

Ground level is assumed to be above sea level, and flat land is assumed. The bedrock is assumed to be less than 40 m 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 is 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.

(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.

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.

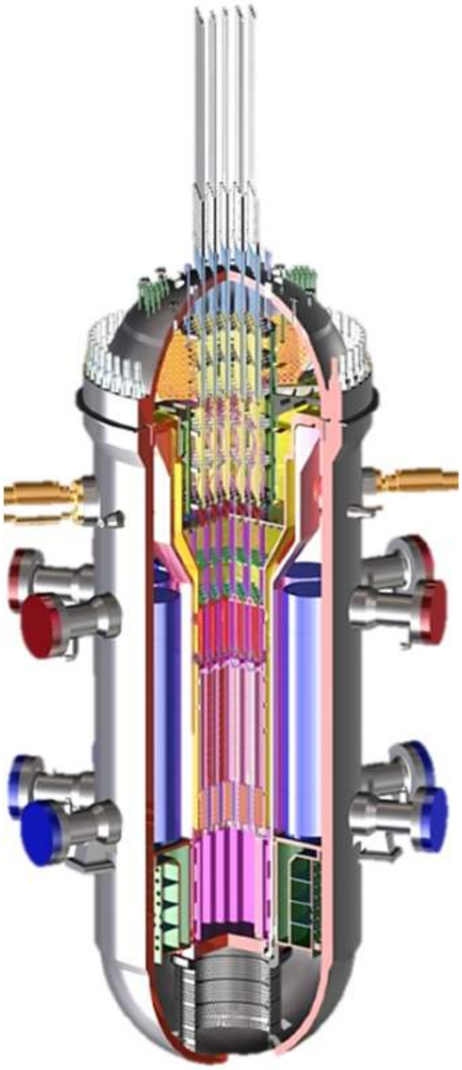
7. Design and Licensing Status

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.

8. 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 performed

SMART (KAERI, Republic of Korea)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Korea Atomic Energy Research Institute (KAERI), Republic of Korea
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	330/100
Primary circulation	Forced circulation
System pressure (MPa)	15
Core inlet/exit temperatures (^{0}C)	296/323
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies	57
Fuel enrichment (%)	< 5
Fuel burnup (GWd/ton)	< 60
Fuel cycle (months)	36
Main reactivity control mechanism	Control rod driving mechanisms and soluble boron
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	90000
RPV height/diameter (m)	18.5/6.5
Module weight (metric ton)	1070 (including coolant)
Seismic design	> 0.18 g automatic shutdown
Distinguishing features	Coupling with desalination and process heat application, integrated primary system,
Design status	Licensed/certified (standard design approval)

1. Introduction

The system-integrated modular advanced reactor (SMART) is an integral PWR with a rated electrical power of 107 MW(e) from 365 MW(t). SMART adopts advanced design features to enhance 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 economics through system simplification, component modularization, reduction of construction time and high plant availability.

2. Target Application

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 large enough to meet the demands of electricity and fresh water for a city population of 100000.

3. Specific Design Features

(a) Design Philosophy

The SMART design adopts an integrated primary system, modularization and advanced passive safety systems to improve the safety, reliability and 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 neither operator intervention to cope with malfunctions and safety events. Improvement in economics is achieved through system simplification, in-factory fabrication, reduction of construction time and high plant availability.

(b) 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 (LB-LOCA) from the design bases events. The nuclear steam supply system (NSSS) consists of the reactor core, steam generators, reactor coolant pumps, control rod drive mechanisms, and reactor internals in the reactor pressure vessel (RPV) and the reactor closure head. The primary cooling system is based on forced circulation by the reactor coolant pumps during normal operation. The system has natural circulation capability for use in emergency conditions.

(c) 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. A two-batch refuelling scheme without reprocessing provides a cycle of 870 effective full power days for 30 months of operation.

(d) Reactivity Control

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

(e) Reactor Pressure Vessel and Internals

SMART integrated design means that the RPV contains all of the primary components. The RPV houses the reactor core, eight (8) steam generators (SGs), four (4) canned motor reactor coolant pumps, twenty five (25) control rod drive mechanisms and reactor internals such as the core support barrel assembly and the upper guide structure assembly.

(f) 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.

(g) Pressurizer

The in-vessel pressurizer is designed to control the system pressure at a nearly constant level for plant operation. The 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.

4. Engineered Safety System Approach and Configuration

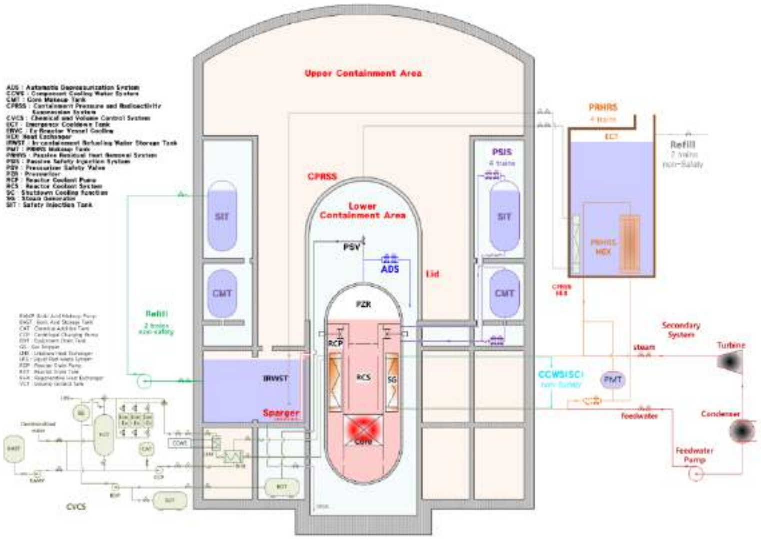
Safety systems of SMART are designed to function automatically on demand. These consist of a reactor shutdown system, a passive safety injection system, a passive residual heat removal system, and containment pressure and radiation suppression system. Additional safety systems include reactor overpressure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

(a) Decay Heat Removal System

After the reactor is shutdown, when the normal decay heat removal mechanism utilizing the secondary system is not operable for any reason, the PRHRS brings the RCS to a 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 operators for the postulated design basis accidents. The safety function of PRHRS is maintained continuously for a long term period when the emergency cooldown tank (ECT) is replenished periodically by a refilling system designed according to regulatory treatment of non-safety system (RTNSS) requirements.

(b) Emergency Core Cooling System

A passive safety injection system (PSIS) 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 the early stage of a SBLOCA or non-LOCA event. The top and bottom of CMT are connected to the RCS through the pressure balance line and safety injection line, respectively. The safety injection function of the PSIS is maintained long term as the SITs are replenished periodically.Engineered safety system of SMART



(c) Containment System

The containment system is designed to contain radioactive fission products within the containment building and to protect the environment against primary coolant leakage. This safety function is realized by the containment pressure and radiation suppression system (CPRSS) as a passive safety system. The containment system is composed of the lower containment area (LCA), the upper containment area (UCA), the in-containment refuelling water storage tank (IRWST), and the CPRSS, besides these, it includes the ECT heat exchange system (ECTHS). In case of main steam line break (MSLB) or LOCA, some of the released energy is absorbed into the IRWST and the rest is removed to environment by the ECTHS. Fission products are scrubbed in the IRWST water. For combustible gas control, passive autocatalytic hydrogen recombiners are equipped inside LCA and UCA.

5. Plant Safety and Operational Performances

The load follow operation of SMART is simpler than that of large PWR because only a 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. The daily load follow performance of SMART core shows that radial peaking factor, 3-dimensional peaking factor and the axial offset were satisfied within design limit.

6. 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. The 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 also used by the reactivity control system. The in–core instrumentation system consists of 29 detector assemblies which are developed as mini type for SMART with four stacked rhodium self–powered detectors.

7. Plant Arrangement

SMART NPP has been designed to be located in coastal area. Therefore, SMART NPP has a seawater intake structure and other buildings including chlorination building in the yard. Power block accommodates reactor containment and auxiliary buildings, turbine generator buildings for two units and one compound building for two units. The compound building consists of an access control area, a radwaste treatment area, and a hot machine shop. Reactor containment building and auxiliary building are integrated into one building named as the rector containment and auxiliary building (RCAB). RCAB houses reactor containment area, auxiliary area and fuel handling area. For efficient radiation management, the RCAB is sub-divided into two zones; the duty zone and the clean zone.

(a) Reactor Containment and Auxiliary Building

Reactor containment and auxiliary building (RCAB) is seismic category-I reinforced concrete structure. The RCAB has been developed with the integration of the auxiliary and reactor containment building to adapt the small and modular plant concept to combine the related equipment and system. Reactor containment area consists of lower containment area (LCA) and upper containment area (UCA). Lower containment area houses reactor pressure vessel, core makeup tanks (CMT), and safety injection tanks (SIT). Ample laydown space is provided at the refueling deck level for equipment dismantling and tool handling. Auxiliary area houses emergency cooldown tanks (ECT), main control room (MCR), electrical and control facilities, main steam isolation valves in separate areas. The auxiliary area houses safety-related equipment required to provide safe shutdown capability. Redundant systems essential for safe shutdown shall be physically separated from one another to prevent a common failure of both systems.

(b) Main Control Room

The SMART compact control room is designed for one reactor operator operation with shift supervisor 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 for standard design approval. 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 including minimum inventory control, main monitoring and control workstation, auxiliary monitoring and control workstation, and a large display panel are installed in the MCR.

(c) Balance of Plant

i. 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.

ii. Electric Power Systems

The electrical power system provides a reliable power to all electrical auxiliary loads, and provides the power plant output to the transmission system. The offsite power consists of the switchyard system (SWYD) and transmission system, and onsite power consists of plant main power system (MP), plant auxiliary power system (AP) and DC distribution system and instrumentation and control power system (DC/IP). The switchyard supplies the generator power to the transmission grid and also provide the preferred power circuit for the auxiliary power system of SMART. The MP system converts mechanical energy into electrical energy and transforms it to supply the power grid and station electrical auxiliaries. The auxiliary power (AP) system provides a reliable source of power to all electrical AC loads in the station. The DC/IP system provides the plant normal control power and backup power to plant control system during normal operation and abnormal operation. The DC system also provides power to DC emergency lighting system and motor operated valves.

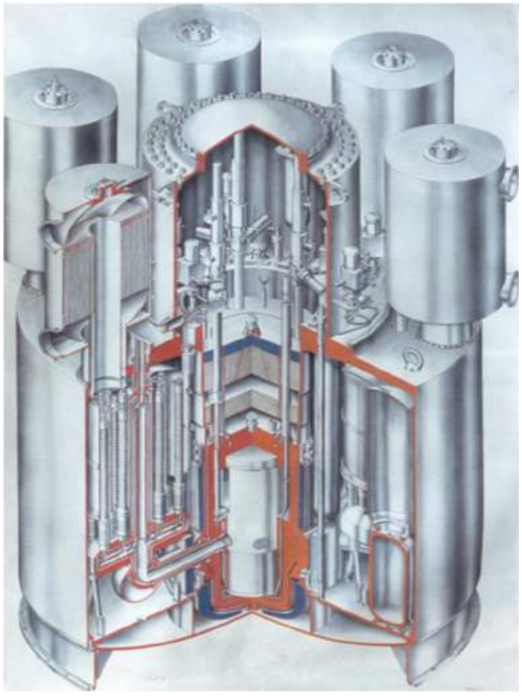
8. Design and Licensing Status

Korea Atomic Energy Research Institute (KAERI) received the standard design approval from Korean Nuclear Safety and Security Commission (NSSC) in July 2012. 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. 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.

9. 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

ELENA (National Research Centre “Kurchatov Institute”, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	National Research Centre “Kurchatov Institute” (RRC KI), Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	3.3/0.068
Primary circulation	Natural circulation
System pressure (MPa)	19.6
Core inlet/exit temperatures (ºC)	311/328
Fuel type/assembly array	UO_2 pellet; MOX is an option
Number of fuel assemblies	109
Fuel enrichment (%)	15.2
Fuel burnup (GWd/ton)	57600/27390
Fuel cycle (months)	300
Main reactivity control mechanism	Control rods and absorber rods
Approach to engineered safety systems	Passive
Design life (years)	--
Plant footprint (m^2)	--
RPV height/diameter (m)	3.7/1.25
Seismic design	Target 85% of contiguous USA
Distinguishing features	Twenty-five (25) years plant life without refuelling, passive reactivity regulation and control systems and a long period of unattended operation
Design status	Conceptual design

1. Introduction

The ELENA nuclear thermoelectric plant (NTEP) is a direct conversion water-cooled reactor without on-site refuelling capable of supplying 68 kW(e) of electricity and 3.3 MW(t) of heating capacity for 25 years without refuelling. The technology and techniques were developed incorporating 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 was 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 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 method 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 and the use of thermoelectric energy conversion to produce electricity.

2. Target Application

The unattended ELENA NTEP plant is designed to produce heat for towns with a population of 1500–2000 and 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. Specific Design Features

(a) 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 areas, as well as in uninhabited or underwater stations, e.g., robotized systems for investigation and extraction of ocean resources or hydrology research laboratories.

(b) Nuclear Steam Supply System

The nuclear steam supply system (NSSS) consists of a reactor core internals and steam generators. The design is based on an integral reactor located in a large volume of secondary water. The NSSS is enclosed in a cylindrical vessel that is embedded in a 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.

(c) Reactor Core

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

(d) Reactivity Control

A reliable operation and reactivity control are 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 is to provide a small total reactivity margin in a hot core so 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.

(e) Reactor Pressure Vessel and Internals

The cylindrical core with a height of 850 mm and an equivalent diameter of 833 mm is installed in a steel shell with a diameter of 920 mm and 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.

(f) 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 is capable of unattended operation without on-site refuelling for up to 25 years. The temperature of water within the third loop is ~100^0C. 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.

(g) Pressurizer

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 shield;
- 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 strengthened area of the 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 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 a protective plate.

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

(a) 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 the defence-in-depth approach based on six safety barriers prevents the depressurization of the primary circuit from depressurization and secure activity confinement inside the reactor during accidents. Though 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 a 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 consumers. The active components of the protection system are scram actuators for the six compensation groups of control rods.

(b) Decay Heat Removal System

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

(c) 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.

(d) Containment System

The reactor is installed in a caisson forming a heat-insulating gas cavity in the strengthened area of the 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 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 beginning of 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 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

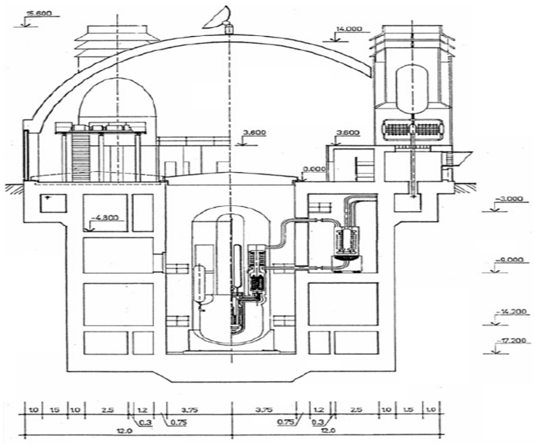
The instrumentation and control (I&C) system of the ELENA reactor is specially arranged to register parameter deviations at early stages of the accidental conditions to predict their further progression.

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.

(a) 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.



(b) 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 information to be transmitted to a monitoring centre.

(c) Balance of Plant

i. 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 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.

ii. Electric Power Systems

The ELENA-NTEP CSS has three independent power supply systems, consisting of 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 a low electrical conversion efficiency, and the waste heat is used for district heating.

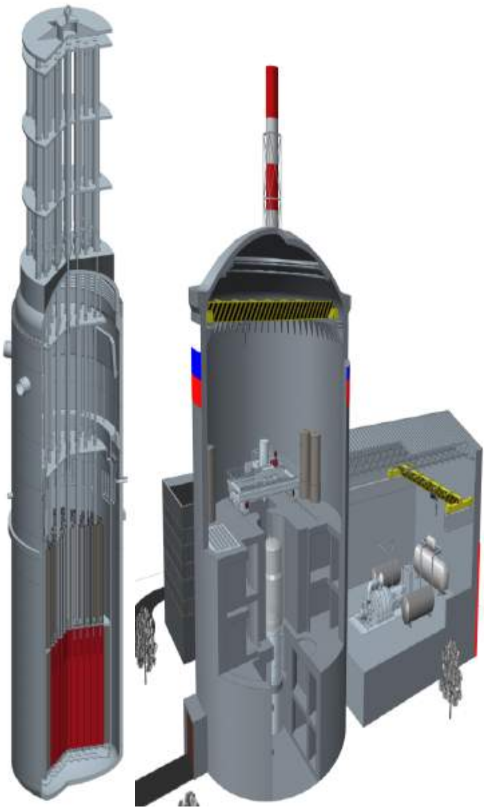
8. Design and Licensing Status

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

9. Development Milestones

ELENA is a conceptual design. Milestones are to be developed.

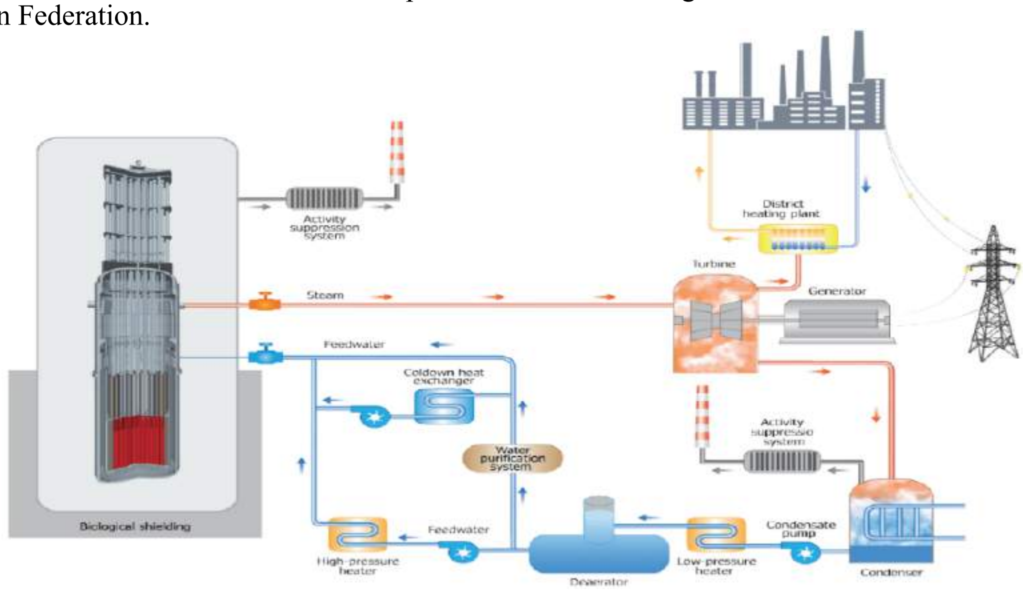
KARAT-45 (NIKIET, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	BWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	180/45-50
Primary circulation	Natural circulation
System pressure (MPa)	7.0
Core inlet/exit temperatures (^{o}C)	180/286
Fuel type/Assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies	109
Fuel enrichment (%)	4.5
Fuel burnup (GWd/ton)	45.9
Fuel cycle (months)	84
Main reactivity control mechanism	Control rods drive mechanism
Approach to engineered safety systems	Passive
Design life (years)	80
Plant footprint (m^2)	9000
RPV height/diameter (m)	11.15/3.10
Seismic design	0.3 (g)
Distinguishing features	Designed for extreme arctic and northern area conditions
Design status	Conceptual design

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.



2. Target Application

KARAT-45 power unit has a high load follow capability to cope with daily power variation from 20 % to 100 % of nominal capacity.

3. Specific Design Features

(a) Design Philosophy

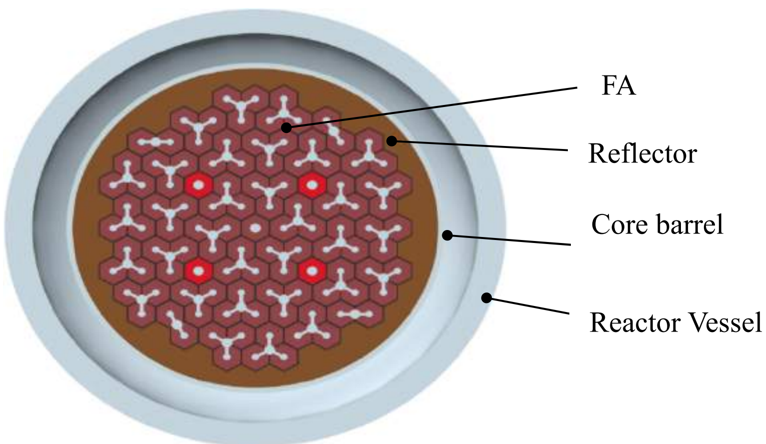
The BWR technology was selected as a basis for the design and technology development of KARAT-45 due to the following rationales: BWR employs single circuit removal of heat so capital cost for construction can be minimized; lower system pressure poses fewer challenges to the reactor vessel; BWR has inherent 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.

(b) 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.

(c) Reactor Core

The reactor core 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.



(d) 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.

(e) 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.

(f) 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.

(g) 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.

(h) Pressurizer

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

4. 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 reactivity control, core cooling and confinement of radioactive materials in the required limits.

(a) 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.

(b) 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.

(c) 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.

(d) 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.

(e) 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.

5. 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.

6. 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.

7. 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.

(a) 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.

(b) Control Building

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

(c) Balance of Plant

i. 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.

ii. 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.

8. Design and Licensing Status

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

9. 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

KARAT-100 (NIKIET, Russian Federation)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	BWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	360/100
Primary circulation	Natural circulation
System pressure (MPa)	7.0
Core inlet/exit temperatures (°C)	104/286
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies	199
Fuel enrichment (%)	4
Fuel burnup (GWd/ton U)	45.9
Fuel cycle (months)	90
Main reactivity control mechanism	Control rods drive mechanism
Approach to engineered safety systems	Passive
Design life (years)	80
Plant footprint (m^2)	22500
RPV height/diameter (m)	13.25/4.00
Seismic design	0.3 (g)
Distinguishing features	Multi-purpose reactor: electricity generation, heat production and nuclear cogeneration plant
Design status	Conceptual design

1. Introduction

KARAT-100 is an integral type multi-purpose boiling water reactor (BWR) with a power output of 360 MW(t) 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.

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.

3. Specific Design Features

(a) 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.

(b) 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^{0}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

(c) Reactor Core

The reactor core 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.

(d) 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.

(e) 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.

(f) 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.

(g) Pressurizer

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

4. 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.

(a) 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.

(b) 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.

(c) 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.

(d) 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.

5. 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:

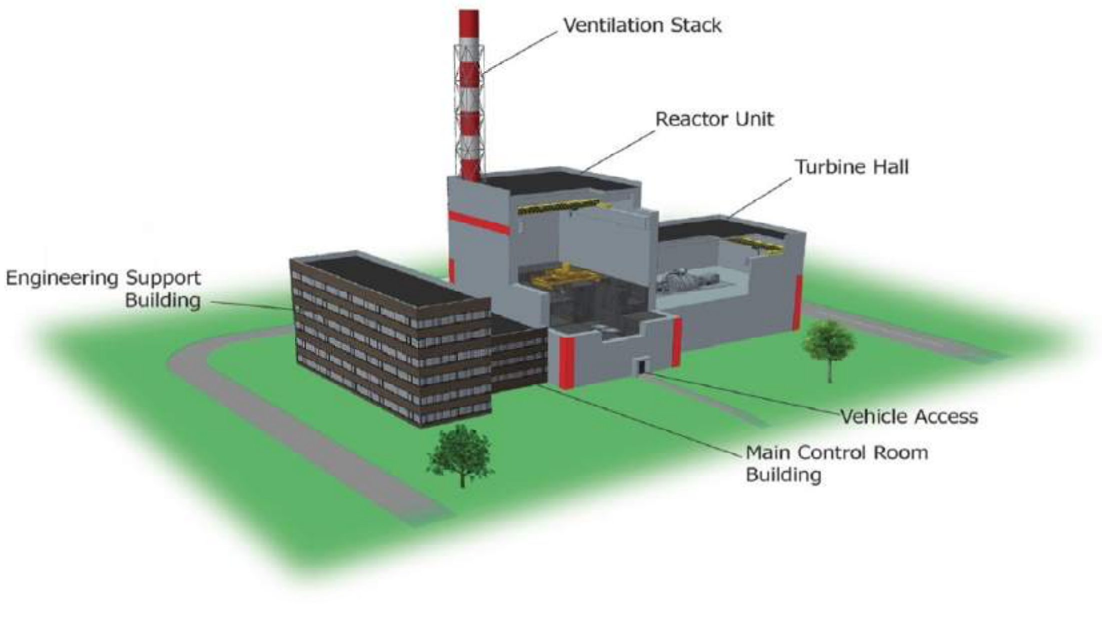
- A high inter-phase barrier (water-steam) to the spreading of nongaseous radionuclides prevents these from entering the steam-condensate line;
- Continuous degassing of coolant and removal of gaseous fission products from the circuit limit their accumulation in the circuit.

6. 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.

7. 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.



(a) 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.

(b) 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.

(c) Balance of Plant

i. 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.

ii. 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.

8. Design and Licensing Status

At the present time, the KARAT-100 is developed to the level of conceptual design and its further development is expected to be continued upon the receipt of the commercial request.

9. 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

RITM-200 (Afrikantov OKBM, Russian Federation)

RITM-200 Reactor

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Afrikantov OKBM, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	175/50
Primary circulation	Forced circulation
System pressure (MPa)	15.7
Core inlet/exit temperatures (^{o}C)	277/313
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies	199
Fuel enrichment (%)	<20
Fuel burnup (GWd/ton)	-
Fuel cycle (months)	60
Main reactivity control mechanism	Control rod driving mechanism
Approach to engineered safety systems	Combined active and passive
Design life (years)	60
Plant footprint (m^2)	10 300 – NPP with 2 reactors 16 500 – NPP with 4 reactors
RPV height/diameter (m)	8.5 / 3.3
Seismic design	0.3 g
Distinguishing features	Integral reactor, in-vessel corium retention, double containment.
Design status	Land-based NPP under development

1. Introduction

RITM series reactor RITM-200 is the latest development in III+ generation SMR line designed by the OKBM Afrikantov and has incorporated all the best features from its predecessors. RITM-200 based NPP are a referenced technology and currently are at a rather advanced development stage and will be available for commercial implementation in short/medium term. RITM-200 based NPP modularity enables to provide power both for commercial/local needs (for example 100 MW) and regional needs (300 MW).

Being initially designed for icebreakers, RITM series reactors are already responding to the key market requirements, such as small size, long refuelling cycle and flexible load following capabilities (100-30-100 % N_{nom}). Six RITM-200 reactors have been already manufactured for nuclear icebreakers. Four reactors are already installed on «Sibir» and «Arktika» icebreakers. Regarding these factors RITM series reactors are the most optimized solution in terms of its size and other technical and economic parameters.

2. Target Application

RITM series reactors are multi-purpose application reactors for electrical power generation, water desalination, and district or industrial heating. RITM-200 based NPPs is suitable for small, isolated, or distributed grids.

3. Specific Design Features

(a) Design Philosophy

RITM series reactors are the evolutionary development of the reactors (OK-150, OK-900, KLT-40 series) for Russian nuclear icebreakers with a total operating experience of more than 50 years (more than 400 reactor-years). With incorporation of the steam generators into the reactor pressure vessel, the reactor system and containment is very compact compared to the KLT-40. RITM design makes it possible to increase electric output (40% more) and reduces the dimensions (45% less) and the mass (35% less) in compare with KLT-40S. While integral reactor configuration almost eliminates the classic large loss-of-coolant accident (LOCA) and the other inherent features and active and passive safety systems apply concepts of diversity, redundancy, physical separation, and functional independence to achieve the necessary safety level and reliability.

(b) Nuclear Steam Supply System

RITM nuclear steam supply system consists of the reactor core, four steam generators integrated in the reactor pressure vessel, four canned main circulation pumps (MCP), and pressurizer. The primary cooling system is based on forced circulation during normal operation and allows natural circulation for emergency condition.

(c) Reactor Core

RITM adopts a low enriched cassette core similar to KLT-40S that ensures long time operation without refuelling and meets international non-proliferation requirements. The core consists of 199 fuel assemblies with fuel enrichment up to 20%. The core has the assigned service life of 7 TWh.

(d) Reactivity Control

Control rods are used for reactivity control. A group of control rods drive mechanisms is intended to compensate for the excessive reactivity at start up, power operation and reactor trip. A group of shutdown rods is designed for fast reactor shut down and to maintain it in the subcritical condition in case of accident. The design of control and shutdown rods is based on the drives used in KLT-40S reactor.

(e) Reactor Coolant System

The reactor pressure vessel (RPV) is thick-walled cylindrical pressure vessel with an integrally welded bottom head and a removable top head. The reactor is designed as an integral vessel with the main circulation pumps (MCP) 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 MCPs are installed in the colds leg of the primary circulation path and separated into four independent loops. The SGs generate steam of 295°C at 3.82 MPa flowing at 261 t/h. The conventional MCPs are used. The pump is vane, single step, and has a sealed asynchronous electric motor with one winding.

(f) Steam Generator

The RITM uses once through (straight tube) SGs. The configuration of the steam generating cassettes makes possible to compactly install them in the RPV.

(g) 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 requires no electric power. 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.

4. Safety Features

The safety concept of the RITM-200 is based on the defence-in-depth principle combined with the inherent safety features and use of passive systems. Properties of inherent safety features 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 systems have been included (system reliability is confirmed by test bench);
- 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 MCP failures.

The exposure dose for the staff in normal operation and design basis accidents does not exceed 0.01% of the natural radiation limit. The public exposure dose in case of severe accident is below the value requiring protective measures.

(a) Engineered Safety System Approach and Configuration

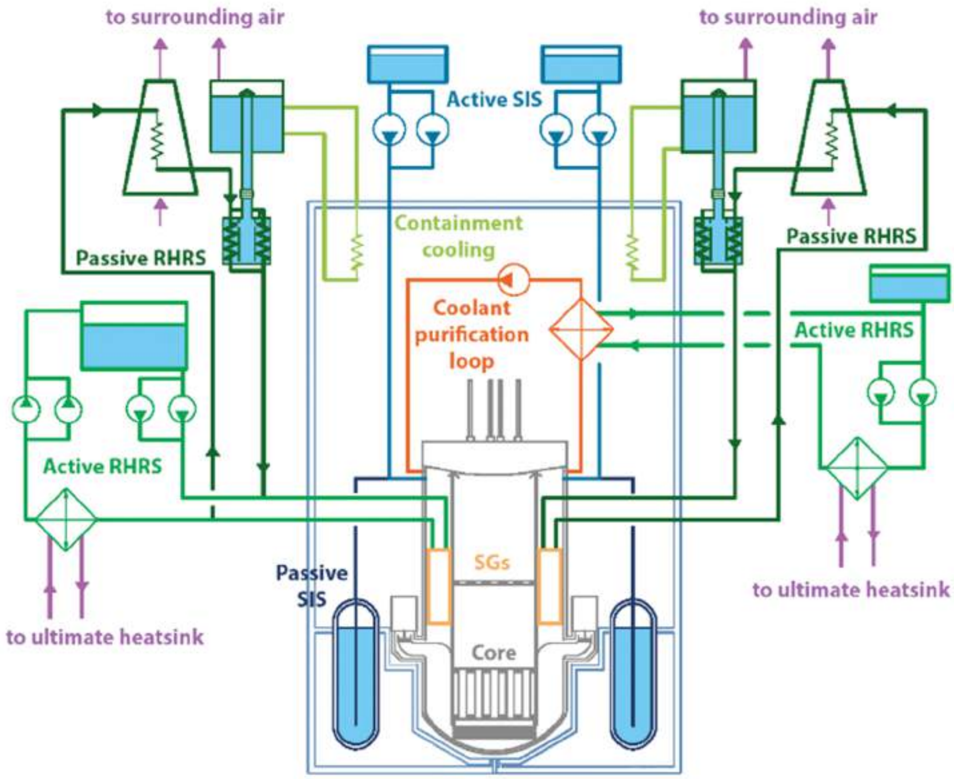
The high safety level of RITM series reactors is achieved both by inherent safety features and a combination of passive and active safety systems. Moreover, redundancy of safety system equipment and channels and their functional and/or physical separation are provided to ensure high reliability. Safety 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 pressure to ensure reactor trip and initiate the safety systems. Shutdown rods drop into the core by gravity with spring assist when power is removed from electromechanical clutches consequently ensure reactor shutdown even in case of total station black out.

(b) Residual Heat Removal System

The residual heat removal system (RHRS) consists of four safety trains:

- Active safety loop with forced circulation through steam generator.
- Active safety loop with forced circulation through primary-third circuit heat exchanger of primary circuit coolant purification loop.
- Two passive safety loops with natural coolant circulation from water tanks through steam generators. Evaporated is steam generators water condenses in air cooled heat exchangers and flow back to tanks with water heat exchangers. After complete water evaporation from the tanks, the air-cooled exchangers continue provide cooling for unlimited time. Combination of air and water heat exchangers allows to minimize dimensions of the heat exchangers and water tanks.

All safety train are connected to different steam generators and provide residual heat removal in compliance with the single failure criterion. Active safety trains consist of water tank, pumps, and heat exchanger to ultimate heatsink.



RITM series safety systems

(c) Emergency Core Cooling System

The emergency core cooling system consists of safety injection system (SIS) for water injection in primary circuit to mitigate the consequences of a break loss-of-coolant accident. The system is based on active and passive principles with redundancy of active elements in each channel and consists of:

- Two passive pressurized hydraulic accumulators;
- Two active channels with water tanks and two make-up pumps in each channel.

In combination with the residual heat removal system the passive safety trains anticipate a post-accident grace period of 72 hours without operator action or power in case of combination of LOCA and total station blackout.

(d) Containment System

Land-based NPP with RITM-200 containment system consists of three levels:

- Hermetically sealed envelope with dimensions of 6m×6m×15.5m around the reactor vessel to localize possible radioactive releases. In case of severe accident thick wall of the reactor vessel keeps molten corium within the reactor. Water filled caisson under the reactor provides the reactor vessel cooling. The envelope integrity ensured by overpressure relief valve, containment cooling system, and passive autocatalytic recombiners.
- Solid building core made of thick reinforced concrete walls.
- Collapsible building structure of thin reinforced concrete walls.

The design of solid core and collapsible structures takes into consideration the maximum potential external impacts including large commercial aircraft crash**.**

5. Instrumentation and Control Systems

An automated control system is provided in the RITM-200 based 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 control of the power plant.

6. Plant Arrangement

The basic architectural design for the land-based NPP includes nuclear island, spent fuel pool, turbine island, auxiliary buildings for water treatment, maintenance, and switchyard connections are similar to conventional large NPP however housed within a relatively small footprint.

Land-based NPP with RITM reactors

7. Design and Licensing Status

The RITM-200 design was developed in conformity with Russian laws, norms and rules for nuclear power plants and safety principles developed by the world community and IAEA recommendations. The RITM-200 design adopts optimal combination of passive and active safety systems. Reactors are manufactured and installed in nuclear icebreakers, with the land-based NPP design under development.

8. Development Milestones

2012	Detailed design of RITM-200
2018	Land-based NPP conceptual design
2020	Land-based NPP detailed design
2022	First concrete
2025	NPP commissioned

RUTA-70 (NIKIET, Russian Federation)

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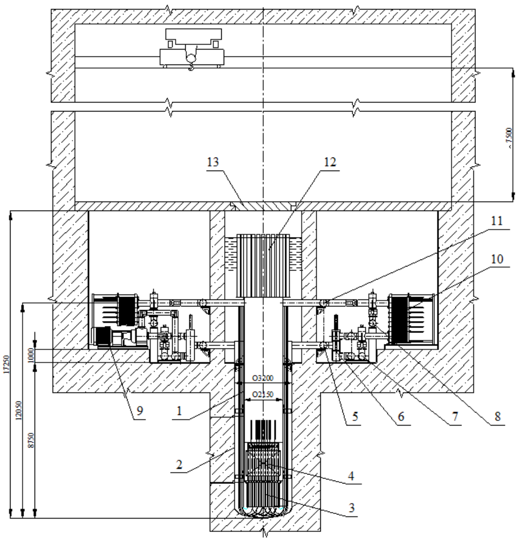


1. Riser shroud
2. Pool metallic liner
3. Core supporting plate with control lead tubes
4. Reactor core
5. Plenum
6. Check valve
7. Secondary water inlet
8. Secondary water outlet
9. Primary pump
10. Primary HX
11. Upper header
12. Control rod drives
13. Isolation plate

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Pool-type
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	70/NA
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
Fuel type/assembly array	Cermet (0.6 UO_2+ 0.4 Al alloy) / hexagonal
Number of fuel assemblies	91
Fuel enrichment (%)	3.0
Fuel burnup (GWd/ton)	25-30
Fuel cycle (months)	36
Main reactivity control mechanism	Control rods
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	100000
Reactor pool height/diameter (m)	17.25/3.20
Seismic design	>0.8g (automatic shutdown)
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
Design status	Conceptual design

1. Introduction

RUTA-70 is a multi-purpose water-cooled water-moderated integral pool-type reactor (NHP) of 70 MW(t) thermal capacity for district heating, desalination and radioisotopes production for medical and industrial purposes. It has no energy conversion system. In the primary cooling circuit, the heat from the core is transferred to the primary heat exchanger (HX) by forced convection at full power and by natural convection at power 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 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 RUTA reactors can be located in the immediate vicinity of the heat users.

2. Target Application

The conceptual design of RUTA is primarily developed to provide district heating in remotely isolated areas. 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. Specific Design Features

(a) 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 based on a simple design and as a result has a low cost of plant construction and operation, high level of safety achieved through 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 pool type research reactors.

(b) 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.

(c) 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. 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.

(d) 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.

(e) 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. 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.

(f) 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 increases 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. 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.

4. Safety Features

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.

(a) 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. 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.

(b) 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 by passive condensing facilities. 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.

(c) 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.

(d) 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. The big amount of water in the reactor pool ensures slow changing of coolant parameters and reliable heat transfer from the fuel, even if controlled heat transfer from the reactor is not available. Fuel temperatures are moderate.

(e) Containment System

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

5. 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.

6. Instrumentation and Control System

RCP actuators based on two diverse principles of action have been chosen for the RUTA-70:

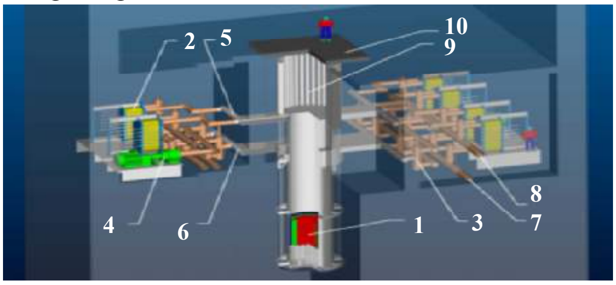
- Multi-position mechanical RCP actuator for automatic (ACR) and manual control rods (MCR);
- Two-position hydrodynamic RCP actuator for scram rods (SR).

In the core there are 42 reactor control and protection system (RCP) rods composing two shutdown systems with diverse actuators. One of these systems intended specifically for core emergency protection (EP) includes 12 rods. The second shutdown system performing the concurrent functions of shutdown and control includes a group of six (6) automatic regulators (ACRs) and four (4) groups of a total of 24 control rods, for remote manual reactivity control (manual rods - MCRs). In response to the scram signal, all control rods of the second shutdown system also perform the functions of emergency protection. MCRs are used to compensate for relatively fast reactivity changes such as heat up and xenon poisoning of the reactor therefore, most of MCRs will be withdrawn under nominal operating parameters. MCRs and scram rods may take the intermediate position in the core performing the functions of power control and forming the radial power profile. The slow transients of reactivity change (such as burn-up of fuel and burnable poison) are also controlled by the group of ACRs plus the required groups of MCRs.

7. Plant Arrangement

(a) 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.



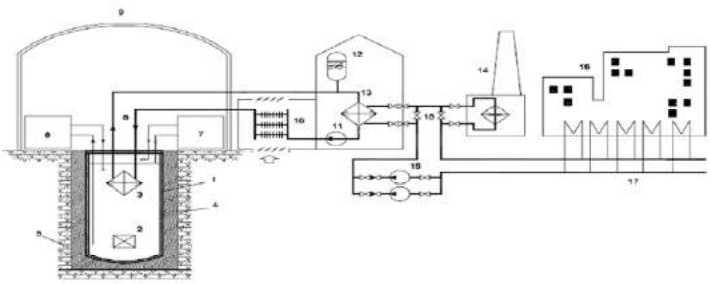
1. Core, 2. Primary heat exchanger
3. Check valve, 4. Pump
5. Primary Circuit distributing header
6. Secondary circuit inlet pipeline
7. Secondary circuit outlet pipeline
8. SCS drives, 9. Upper slab

(b) 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 start up and refuelling periods. The total personnel number including regular engineers, technicians and administrative staff may reach up to 40 persons.

(c) Balance of Plant

The turbine and associated systems are not used in the NHP RUTA-70. Such scheme, despite some reduction of autonomy of the 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.



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

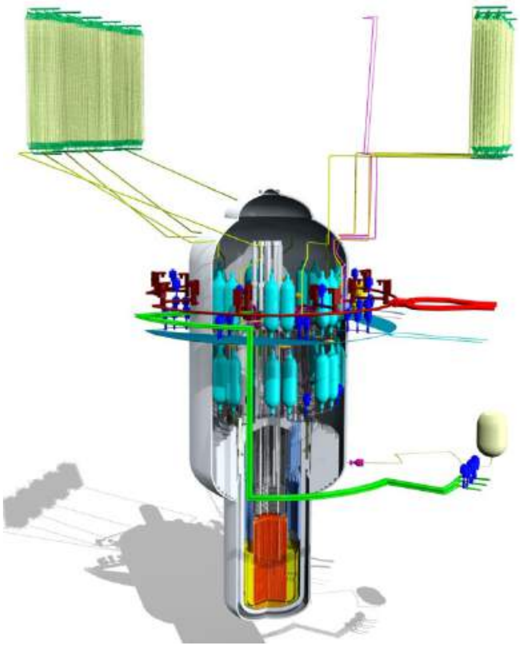
8. Design and Licensing Status

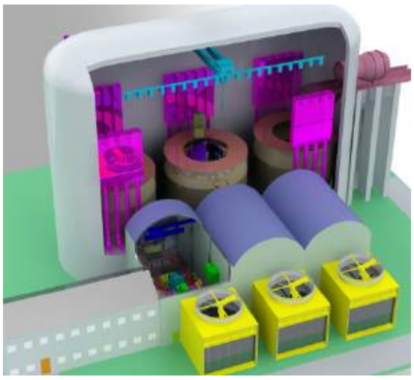
To provide an operating reference for the reactor, in 2004, the feasibility study was carried out jointly by NIKIET, IPPE, and Atomenergoproekt (Moscow). This study showed that RUTA-70 could be deployed along with the non-nuclear sources of power operating in peak and off-peak mode.

9. Development Milestones

1990	Conceptual design of the 20 MW(t) RUTA heating plant
1994	Feasibility study 'Underground NHP with 4 × 55 MW(t) RUTA reactors for district heating in Apatity-city, Murmansk region'
2003	Technical and economic assessments for regional use of the 70 MW(t) RUTA reactor to improve the district heating system.

UNITHERM (NIKIET, Russian Federation)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	PWR
Coolant/moderator	High purity water
Thermal/electrical capacity, MW(t)/MW(e)	30/6.6
Primary circulation	Natural circulation
System pressure (MPa)	16.5
Core inlet/exit temperatures (ºC)	249/330
Fuel type/assembly array	UO_2 particles in a metallic (silumin or zirconium) matrix, metal-ceramic/ 54-55
Number of fuel assemblies	265
Fuel enrichment (%)	19.75
Fuel burnup (GWd/ton)	1.15
Fuel cycle (months)	200
Main reactivity control mechanism	Soluble boron and control rod insertion
Approach to engineered safety systems	Hybrid (passive + active) system
Design life (years)	25
Plant footprint (m^2)	~10000
RPV height/diameter (m)	9.8/2.9
Seismic design	VIII-IX-MSK 64
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
Design status	Conceptual design

1. Introduction

The UNITHERM is a small transportable nuclear power plant (NPP) with a capacity of 30 MW(t) 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.

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. Specific Design Features

(a) 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.

(b) 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.

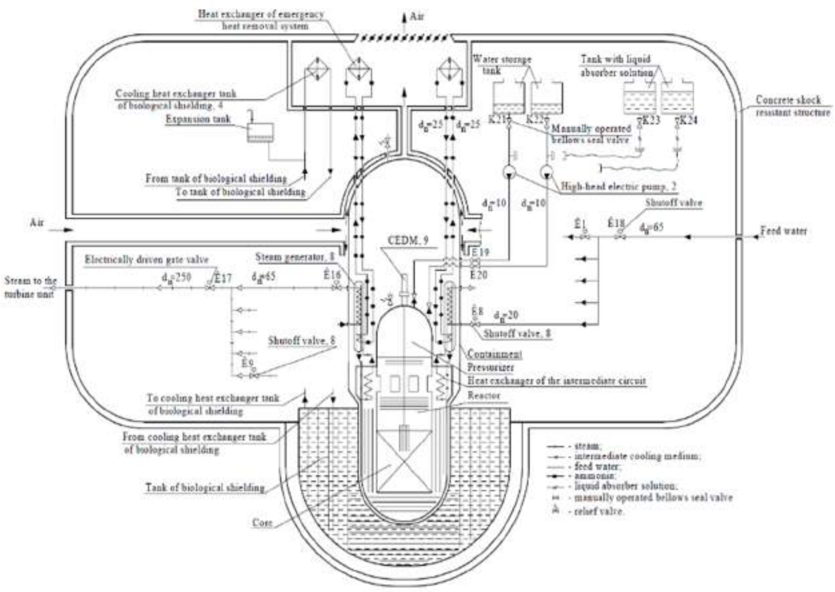
The secondary system (consumer’s circuit) is intended to generate a superheated steam from the feedwater (supplied by NPP turbogenerator pumps) by means of heat transfer from the intermediate circuit coolant inside the steam generator. Secondary coolant circulates by means of natural convection.

(c) 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.

(d) 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.



(e) 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).

(f) 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).

(g) 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.

(h) Pressurizer

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

4. 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.

(a) 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.

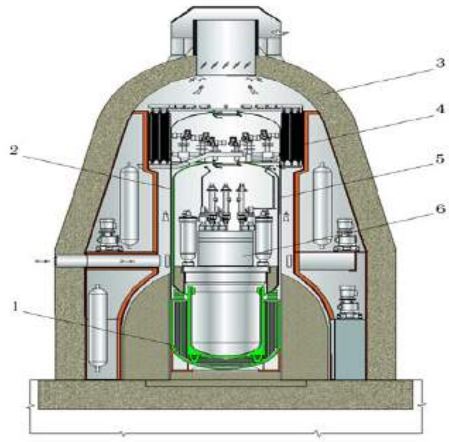
(b) Emergency Core Cooling System

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 shielding 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.

(c) 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.

(1) Iron-water shielding tank;
(2) Containment;
(3) Shock-proof casing;
(4) Cooldown system heat exchanger;
(5) Safeguard vessel
(6) The reactor



5. Plant Safety and Operational Performances

Electrical output of the NPP with UNITERM-30 reactor equals to N(e) – 6,6 MW(e). Electrical voltage provided to the user grid – alternate 3-phase 10,5 кV ± 10 %, frequency 50 ± 1Hz.

Basic regime of NPP operation lies within the power range from 20 to 100 % N(e) providing a daily and annual load following. The speed of power augment and drop – 0,1 % от N (e)/sec.

Upon a customer request there may be foreseen additional provision of thermal power. Maximum rate of it could be up to 28 Gcal/hour.

6. Instrumentation and Control Systems

Automatic System of control for technological processes of the NPP allows for:

- Safe operation of the NPP and electrical generation; protection from the violations of safe operation limits and conditions; prevention of accidents; mitigation of accident consequences; bringing the NPP back to the controlled and safe condition during accidents and after them.
- The Automatic Control System consists of functionally completed systems developed on the basis of programmatic-technical systems and instruments that were trialled n the NPP conditions or other analogous objects.
- Technical appliances for the ACS are manufactured at the enterprises according to approbated technology and methods of testing and control while strictly observing the requirements of quality control.

7. Plant Arrangement

(a) Reactor Building

The NPP site is limited by perimeter of the protected zone that does not exceed a square of 2 hectares.

The site hosts reactor building for housing reactor(s) which possesses special transport locks for delivery of the reactor plant for mounting and other equipment necessary during outages and removal of the reactor facility; building to house turbine-generator(s) and some other auxiliary buildings. 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.

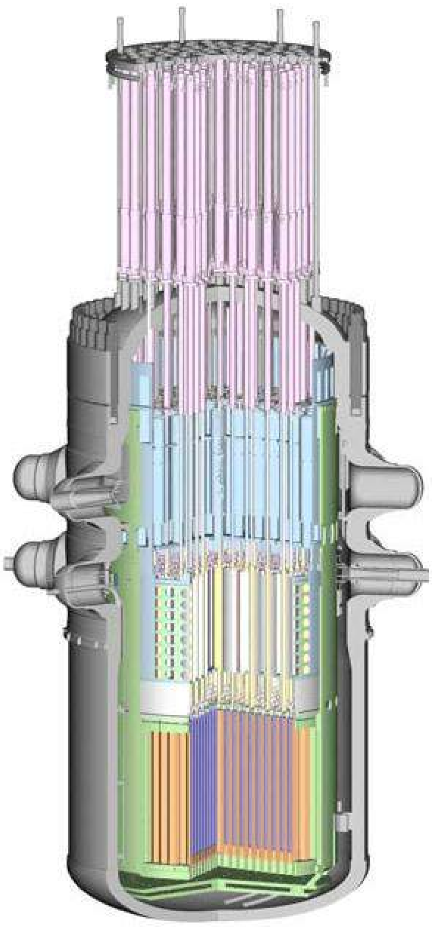
8. Design and Licensing Status

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.

9. 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

VK-300 (NIKIET, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Simplified passive BWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	750/250
Primary circulation	Natural circulation
System pressure (MPa)	6.9
Core inlet/exit temperatures (°C)	190/285
Fuel type/assembly array	UO_2 pellet/hexahedron
Number of fuel assemblies	313
Fuel enrichment (%)	4
Fuel burnup (GWd/ton)	41.4
Fuel cycle (months)	72
Main reactivity control mechanism	Rod insertions
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	40000
RPV height/diameter (m)	13.1/4.535
Seismic design	Max 8 point of MSK-64
Distinguishing features	Innovative passive BWR based on operating prototype and well-developed equipment
Design status	Detailed design of reactor and cogeneration plant standard design

1. Introduction

The VK-300 is an integral simplified passive boiling water reactor (BWR) with a rated output of 750 MW(t) or 250 MW(e), adopting natural circulated primary coolant system. The design and operation of the VK-50 simplified BWR reactor in the Russian Federation for 50 years is used as a basis for the design of the VK-300 reactor. The design is based on a proven technology, utilizing the components developed and manufactured for other reactor types. The VK-300 uses the reactor pressure vessel and fuel elements of the WWER-1000 reactor. The design configuration incorporates inherent and passive safety systems to enhance the safety and reliability. The design aims to achieve improved economics through system simplification. The reactor core is cooled by natural circulation of coolant during normal operation and in emergency condition. The design reduces the mass flow rate of coolant by initially extracting moisture from the flow and returning it to the core inlet, ensuring a lower hydraulic resistance of the circuit and raising the natural circulation rate. The VK-300 reactor has a reactivity margin for nuclear fuel burnup due to the partial overloading and use of burnable absorbers. The integral arrangement of reactor components and availability of preliminary and secondary containments are non-proliferation features of VK-300.

2. Target Application

VK-300 reactor facility is specially oriented to the effective co-generation electricity and heat for district heating and for sea water desalination having excellent characteristics of safety and economics.

3. Specific Design Features

(a) Design Philosophy

Design of the VK-300 is based on the proven WWER technologies and takes over the operating experience of the reactor of smaller size namely VK-50 that successfully operates in Russian Federation over the last 50 years. Therefore, by the use of some proven modified structures and components in the design the enhanced reliability and economics are achieved.

(b) Nuclear Steam Supply System

In a cogeneration plant with VK-300 reactor steam goes directly from reactor to a turbine. After passing several stages, some steam is extracted from the turbine and sent to the primary circuit of the district heat supply or to the sea water desalination facility. Heat from the secondary circuit of the district heat facility is supplied to consumers. The circuit pressures are chosen so as to exclude possibility of radioactivity transport to the consumer circuit.

(c) Reactor Core

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

(d) Reactivity Control

The reactor is provided with two independent reactivity control systems that use different principles of action. The first system is a traditional rod system including 90 drives of the CPS. Each of the drives simultaneously moves control rods installed in three adjoining fuel assemblies of the core. The second reactivity control system is a liquid system intended for injection of boric acid solution to the reactor coolant at failures of the rod reactivity control system. The system consists of pressurized hydraulic accumulators with a boric acid solution. A lifting tube unit provides a guiding structure for the reactor control rods, which is very important at the upper location of the CPS drives. The VK-300 reactor has a small reactivity margin for fuel burnup that creates pre-conditions for designing a simpler CPS system with light rods, which mitigates the consequences of accidents with the CPS rod withdrawal.

(e) Reactor Coolant System

The VK-300 primary cooling mechanism under normal operating condition and shutdown condition is by natural circulation of coolant. The VK-300 design adopts an advanced coolant circulation system and a multistage separation in the reactor. A lifting tube (chimney) unit forms the raising and downstream coolant flows, preliminary separates moisture and build-up the water inventory (between lifting tubes) that immediately goes back to the reactor core in the event of the reactor shutdown or during accidents.

(f) Steam Generator

The VK-300 reactor employs in-vessel cyclone separators that are designed and experimentally optimized for being used in vertical steam generators of the WWER-1000.

4. Safety Features

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

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

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

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

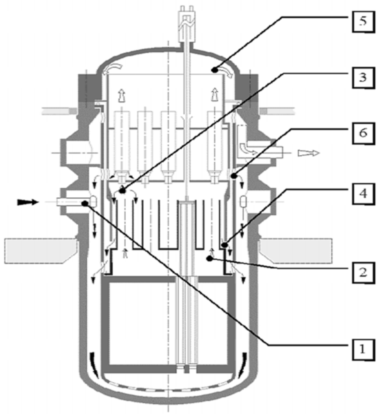
(a) Decay Heat Removal System

The primary task following a scram actuation is to remove residual heat from the shutdown reactor and ensure its normal cool down. This function is performed by the residual heat removal system (RHRS) that passively removes heat from the reactor in special heat condensers located inside the PC. The condensers are connected to the reactor by pipelines that are filled with water during normal operation of the reactor. As the water level decreases in the reactor, the upper pipeline opens for the steam passage from the reactor to the condensers and the resultant condensate goes back to the reactor. The RHRS condensers are cooled with water from the emergency cool down tanks. The system is fully based on passive principles of action and ensures natural heat transport from the reactor to the emergency cool down tanks. The heat capacity of the tanks as such is enough for independent operation throughout the day (i.e. without personnel interference). This interval may be prolonged for an infinite period of time thanks to the operation of the heat removal system from the tanks to the ultimate heat sink. This is a simple and reliable system consisting of two heat exchangers connected with pipelines. One of the heat exchangers is plunged into the emergency cooldown tank water and the other is installed in the atmospheric air flow outside the reactor hall. The coolant in the system is water circulating in the circuit naturally without circulation boosters.

(b) Emergency Core Cooling System

The emergency cooldown tanks contain the water inventory for emergency reactor flooding and core cooling during steam or water line ruptures within the PCS. The emergency cooling tanks (ECTs) performs the functions of: (a) accumulating the reactor energy with the potential of transferring it to the end absorber for an unlimited time period; (b) compensating the cooling water inventory in the reactor during accidents by returning the condensed coolant to the reactor; and (c) receiving steam or steam-water mixture (e.g., the exhaust of the reactor safety valves installed inside the PC). During a LOCA (rupture of a steam line or feedwater pipeline adjoining the reactor within the containment), pressure increases inside the PCS which serves as a signal for actuation of the reactor scram and passive closure of shutoff devices (valves) cutting the reactor off the external steam-water lines. A pressure reduction in the reactor as a result of coolant leak through the rupture creates conditions for the water delivery from the ECTs to the reactor via a special pipeline under the action of hydrostatic pressure. The steam-air mixture goes via discharge pipelines from the containment to the ECTs where it is condensed. As a result, a circulation circuit of the ECT – reactor – PCS – ECT is formed and its function ensures long-term passive cooling of the reactor.

1-Feedwater
2-Out-core mixing chamber
3-Preliminary separation chamber
4-Pre-separated water outlet
5-Steam
6-Major separated water stream



(c) Containment System

The VK-300 reactor adopts a metal-lined primary containment system (PCS) of reinforced concrete. The PCS helps to solve the safety assurance problem economically and reliably using structurally simple passive safety systems. The PCS is rather small, with about 2000 cubic meters. The PCS of the VK-300 performs the functions of: (a) a safeguard reactor vessel; (b) a protective safety barrier limiting the release of radioactive substances during accidents with ruptures of steam, feedwater and other pipelines immediately near the reactor; and (c) providing the possibility of the emergency core cooling by the reactor cooling water making additional water inventory unnecessary.

5. Plant Safety and Operational Performances

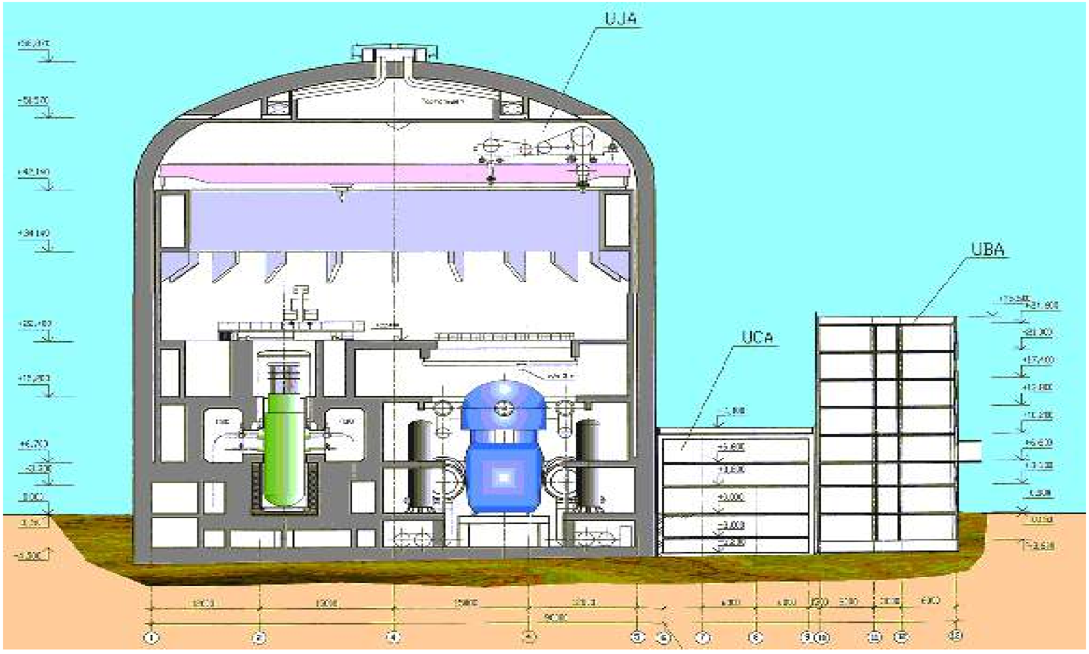
A set of reactor facility safety features and the concept of defence-in-depth against radioactivity escape allow plant location in the vicinity of a residential district limiting the control area around the VK-300 cogeneration plant by the dimensions of the cogeneration nuclear power plant (CNPP) site.

6. Instrumentation and Control Systems

Instrumentation and Control Systems based on proven technologies, ensure cogeneration NPP with effective operation and provide safety assurance.

7. Plant Arrangement

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



(a) Reactor and Turbine Building

The necessity of deploying cogeneration NPP within the city limits, with regard for the single-circuit layout and the necessity of raising the reliability of the environmental protection during accidents, the power unit design stipulates that all of the power unit will be within a leak-tight enclosure (the containment). The containment accommodates the PC with the VK-300 reactor, emergency cooldown tanks, turbine, spent fuel storage pools, refuelling machine and central hall crane. The electric generator is installed in a separate annex outside the containment using a shaft that passes through the containment wall to beyond the containment. The containment is an attended room whose primary function is to protect the reactor from external impacts such as aircraft fall, terrorist acts, etc. Thanks to new layout concepts for the main equipment of the VK-300 power unit, the containment dimensions do not exceed the dimensions of the WWER-1000 reactor.

(b) Electric Power System

Electric Power System of VK-300 cogeneration power unit based on 220 MW turbogenerator.

8. Design and Licensing Status

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

9. Development Milestones

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

UK SMR (Rolls-Royce and Partners, UK)

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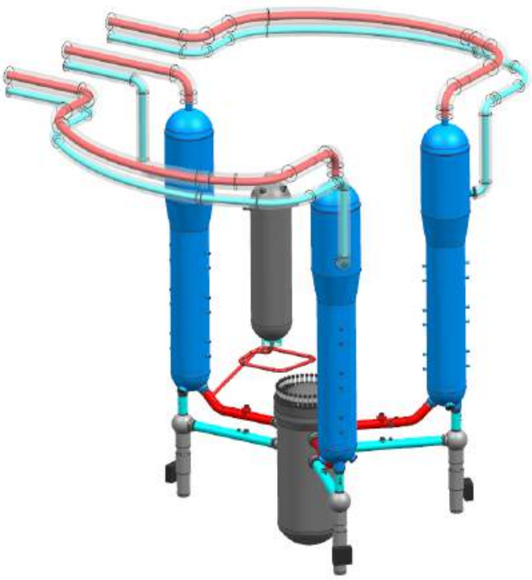
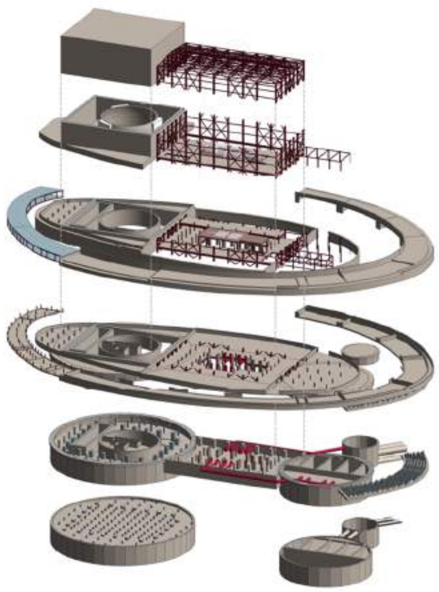




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Rolls-Royce & Partners, United Kingdom
Reactor type	PWR
Coolant/moderator	Light water
Thermal/electrical capacity, MW(t)/MW(e)	1276/443
Primary circulation	Forced circulation
System pressure (MPa)	15.5
Core inlet/exit temperatures (°C)	296/327
Fuel type/assembly array	UO_2 fuel in 17x17 array
Number of fuel assemblies	121
Fuel enrichment (%)	< 4.95
Fuel burnup (GWd/ton)	55 – 60
Fuel cycle (months)	18 – 24
Main reactivity control mechanism	Control rods
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	40000 (10 acres)
RPV height/diameter (m)	11.3/4.5
Seismic design (g, peak ground acceleration)	0.3 unmitigated, with seismic isolation system providing significant withstand beyond unmitigated level
Distinguishing features	Compact site footprint. Modular approach facilitating rapid and cost-effective build. Highly reliable passive safety systems. Aesthetic and functional exterior that is robust to hazards.
Design status	Mature concept

1. Introduction

The UK SMR has been derived to deliver a market driven, affordable, low carbon, energy generation capability. The developed design is based on optimised and enhanced use of proven technologies that presents an attractive market offering with minimum regulatory risk.

A three loop, close-coupled, pressurised water reactor (PWR) provides a power output at 443 MW(e) from 1276 MW(t) using industry standard UO_2 fuel. Coolant is circulated via three centrifugal reactor coolant pumps (RCPs) to three corresponding vertical u-tube steam generators (SGs). The design includes multiple active and passive safety systems, each with substantial internal redundancy.

Rapid, certain and repeatable reactor and overall power station build is enhanced through site layout and maximising modular build, standardisation and commoditisation. The three-loop reactor is located in Reactor Island, adjacent to turbine island, with the cooling water island following. Key facilities are protected by a hazard protection barrier. Support buildings and those containing auxiliary services are situated within a berm that sweeps around the site and provides further protection from external hazards, e.g. tsunami or aircraft impact.

2. Target Application

The UK SMR is primarily intended for electricity production; however, the design can be configured to support other heat-requiring or cogeneration applications.

3. Specific Design Features

(a) Design Philosophy

The UK SMR is designed to optimise levelized the cost of electricity against low capital cost. The power output is maximised whilst delivering robust economics for nuclear power plant investment and a power plant size that enables standardisation and modularisation across the entire power station.

To minimise the construction phase of the programme, the UK SMR is fully modularised to enable the plant to be transported by road, rail or sea. Targeting a 500 day modular build, this concept minimises the onsite time and effort required to construct and build the plant, allowing power generation to commence at the earliest possible point. The plant has been designed to optimise manufacture, inspection and maintenance with the intention to maximise through life plant availability.

Through a combination of active and passive safety measures, the UK SMR has been designed to drive radiological risk as low as reasonably practicable, surpassing internationally recognised numerical safety targets by significant margins.

(b) Reactor Core and Reactivity Control

The nuclear fuel is industry standard UO_2 enriched up to 4.95%, arranged in a 17x17 assembly. UO_2 pellets are contained in tubes made from zirconium alloys, which give good neutron economy, structural stability and corrosion resistance in the PWR environment. The core contains 121 fuel assemblies and has an active fuelled length of 2.8 m. Each fuel assembly contains 40 poisoned fuel pins, with 224 fuel pins being unpoisoned. The poison used is distributed Gd_2O_3 (containing natural gadolinium) at 8 wt%.

No concentration of soluble boron is maintained in the primary coolant for duty reactivity control. This affords a simplified design and eliminates risks associated with hazardous boric acid and environmental impact of boron discharge. It is a design goal to achieve a zero-discharge plant. Duty reactivity control is provided through movement of control rods and use of the negative moderator temperature coefficient inherent to PWRs.

(c) Reactor Pressure Vessel

The RPV assembly consists of an RPV body, a tori spherical closure head assembly and a bolting arrangement comprising studs, nuts, spherical washers and mechanical seals. The RPV diameter is constrained to be less than 4.5 m to ensure that the UK road transport height limit of 4.95 m is not exceeded.

The close-coupled loops are connected to the RPV above the fuelled region with no connections or penetrations below this so that the Loss of Coolant Accident (LOCA) risk is minimised.

(d) Steam Generator

A vertical u-tube SG design has been selected as a mature and readily deployable technology. Other designs have been considered but deemed insufficiently mature for commercial deployment in 2030 in the first of a kind plant.

(e) Pressuriser

Primary circuit pressure is controlled by use of electrical heaters located at the base of the pressuriser and spray from a nozzle located at the top of the pressuriser. Steam and water are maintained in equilibrium to provide the necessary overpressure. The pressuriser is a vertical, cylindrical vessel with top and bottom heads constructed of low alloy steel.

The UK SMR employs surge induced spray whereby primary coolant passively expands into the spray line causing spray. This provides a simple and safe configuration. The pressuriser is sized to provide passive fault response for bounding faults, with accidents causing either rapid and significant cooldown or heat-up accommodated.

4. Safety Features

The UK SMR design has been developed through a combined systems engineering and safety assessment approach. Building on application of relevant good practice and sound engineering, the safety informed design supports the process by which SMR risks are demonstrated to be tolerable and as low as reasonably practicable.

Defence in depth is provided through the provision of robust safety measures, designed against conservative conditions, which meet the guidelines from the deterministic design basis analysis. Multiple layers of fault prevention and protection are provided through diverse and independent active and passive systems, comprehensively ensuring UK SMR safety for design basis and design extension conditions, for all modes of operation and during all lifecycle stages.

In addition to heat removal via the closed loop SG steam and feed cycle, the passive decay heat removal (PDHR) system and the emergency core cooling system (ECCS) are passive, redundant, diverse and segregated protective safety measures that provide multiple means of decay heat removal in response to faults. All design basis LOCAs are protectable by ECCS, with diverse protection additionally available from the small leak injection system (SLIS) for smaller leaks. Control rods (scram) and emergency boron injection provide two diverse and highly reliable means of reactor shutdown. Three safety relief valves are provided to protect against overpressure hazards. Containment is provided to mitigate the release of fission products to the environment in the unlikely event of core damage.

The safety benefits afforded by the design of UK SMR are reflected within the probabilistic safety assessment (PSA) which calculates an overall core damage frequency from all plant hazards $<10^{-07}$ per year of power operation. PSA additionally identifies that the UK SMR presents a balanced design with no single initiating fault making a disproportionate core damage frequency contribution.

The design philosophy prioritises the use of passive safety features. Therefore, failure of active components and supporting electrical supply are not important contributors to the UK SMR risk. As such, hazards which render active systems unavailable such as complete loss of electrical supplies (station blackout) present very low risk. Similarly, failures of operator actions in delivery of safety functions are also identified to not be significantly important, with automated delivery of one-time-movement valve alignment for safety measure actuation, such that the burden on the operator in delivering safety actions is extremely low.

Internal and external hazards assessment have defined the design basis and informed the site layout from the perspective of segregation and separation of safety related equipment. Key equipment is protected by a hazard protection barrier, which is resilient against external hazards including aircraft impact and tsunami.

5. Instrumentation and Control Systems

Control and instrumentation (C&I) systems have been designed in line with recognised and endorsed good practices, including provision of adequate and reliable engineering solutions, defence in depth, minimisation of design complexity, provision of diversity and redundancy, etc.

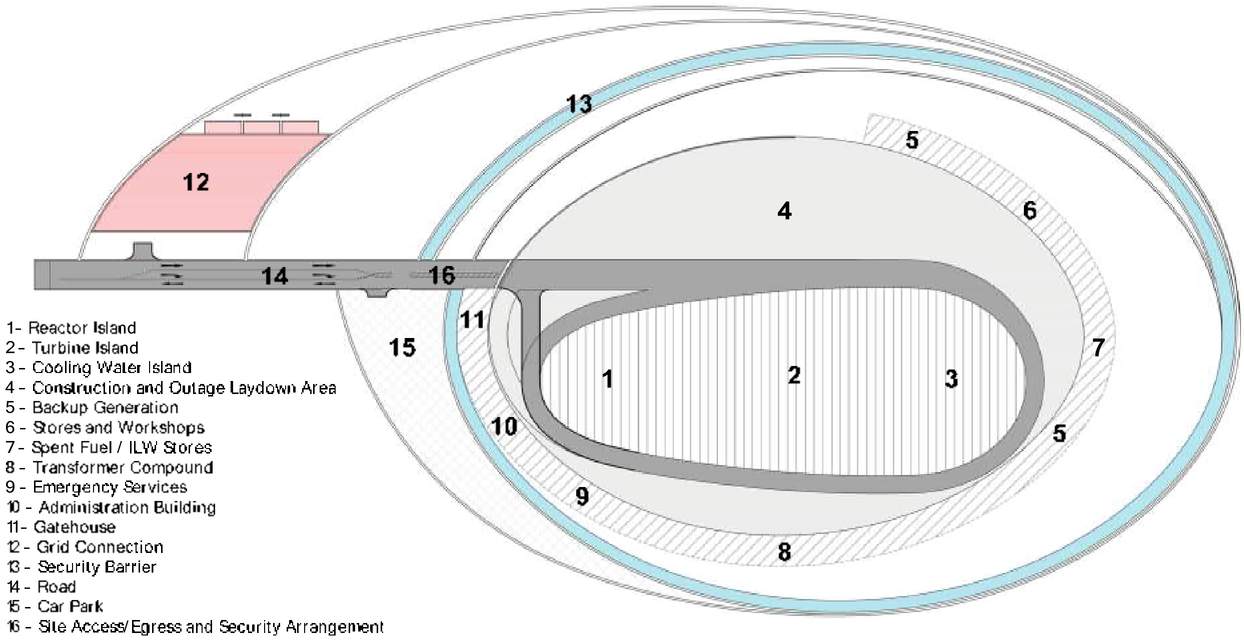
The reactor plant control system will use an available in industry programmable logic controller (PLC) or distributed control system (DCS). It uses mixed analogue and nonprogrammable digital sensors and communicates on hardwired multichannel digital electrical networks. Opportunities to use smart devices and wireless technologies are being pursued.

The reactor protection system (RPS) provides safe shutdown in response to a fault. The RPS contains priority logic, which from the range of input signals received determines whether or not to initiate a reactor shutdown. The RPS uses digital systems, designed specifically for the nuclear industry. It uses mixed analogue and nonprogrammable digital sensors and communicates on hardwired multichannel digital electrical networks by preference. The hardwired diverse protection system (HDPS) is diverse to the RPS and will therefore use nonprogrammable simple electronics.

Post-accident and severe accident management systems within the nuclear C&I system provide clear plant status displays, over the days and months following an accident. Control room human machine interface is defined, encompassing displays, alarms and manual controls. The nuclear C&I definition also covers fuel route and hazardous material monitoring system. Non-nuclear C&I features include distributed control, networked communications and both traditional and ‘smart’ instruments/actuators.

6. Plant Arrangement

An aerial view of the UK SMR and the plant layout are presented below.



7. Design and Licensing Status

Design definition is at a mature design concept stage. A Rolls-Royce design certificate has been issued reflecting the product definition. This covers:

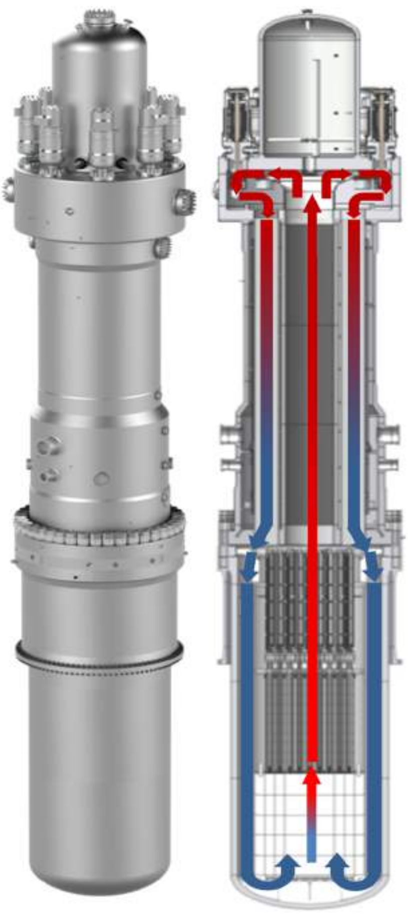
- Power station definition and principles of operation
- Reactor island systems definition
- Turbine island systems definition
- Civil engineering solution
- Site layout
- Electrical power system
- Safety management prospectus
- Preliminary safety and environmental report
- Preliminary security solution

The UK SMR project aims to deploy the first of a kind SMR in the UK by 2030. The programme includes design review and acceptance as part of the UK office for nuclear regulation (ONR) generic design assessment (GDA), with 2024 targeted for regulatory design acceptance.

8. Development Milestones

2015	Rolls-Royce development of initial reference design
2016	Formation of consortium for design of whole power station concept
2017	Mature design concept developed
2024	Planned achievement of regulatory design acceptance confirmation (DAC) and statement of design acceptability (SoDA)
2030	Planned first of a kind commercial operation

mPower (BWX Technologies, Inc., USA)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Generation mPower, LLC, United States of America
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	575/195
Primary circulation	Forced circulation
System pressure (MPa)	14.8
Core inlet/exit temperatures (°C)	290.5/318.9
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies	69
Fuel enrichment (%)	< 5
Fuel burnup (GWd/ton)	< 40
Fuel cycle (months)	24
Main reactivity control mechanism	Control rods
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	157000
RPV height/diameter (m)	27.4/4.15
Seismic design	Target 85% of contiguous USA
Distinguishing features	Internal once-through steam generator, pressurizer and control rod drives
Design status	Under development

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.

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. Specific Design Features

(a) 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 defence-in-depth principle, including an underground steel containment vessel structure and an underground spent fuel storage pool.

(b) 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.

(c) 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.

(d) 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.

(e) 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.

(f) 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.

(g) 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.

(h) Pressurizer

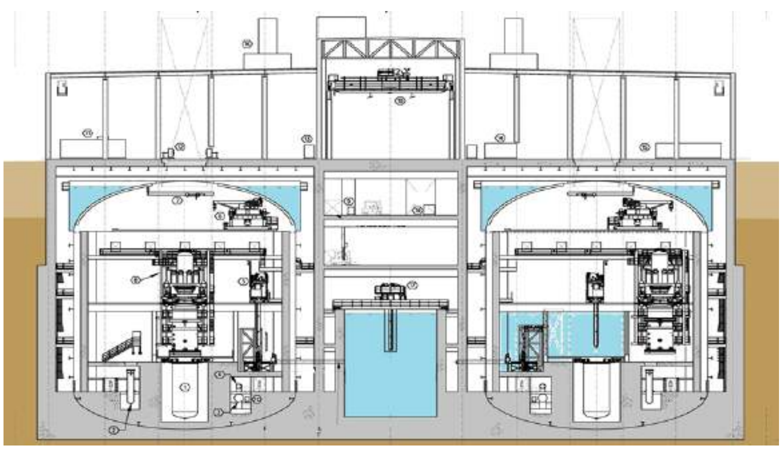
The integrated electrically heated pressurizer located at the top of the RPV maintains a nominal 14.8 MPa. Reactor coolant pressure is controlled by the heaters and steam space spray.

4. 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.

(a) Engineered Safety System Approach and Configuration

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.



(b) Decay Heat Removal System

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

(c) 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.

(d) 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.

5. 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 centres 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, defence-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.

6. 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.

7. Plant Arrangement

(a) 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.

(b) 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.

i. 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.

ii. Electric Power Systems

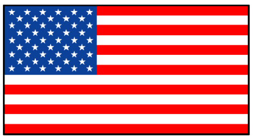
The main generator supplies power to plant auxiliaries during normal plant operation through an isolated phase bus duct and the unit auxiliary transformer. Offsite power to plant auxiliaries during startup, shutdown, and outage conditions is supplied via back-feed from the main transformer and the unit 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.

8. 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.

9. 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.



NuScale (NuScale Power Inc., USA)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NuScale Power, LLC, USA
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	160/50
Primary circulation	Natural circulation
System pressure (MPa)	12.8
Core inlet/exit temperatures (°C)	258/314
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies	37
Fuel enrichment (%)	< 4.95
Fuel burnup (GWd/ton)	> 30
Fuel cycle (months)	24
Main reactivity control mechanism	Control rod drive, boron
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	140000
RPV height/diameter (m)	17.8/3.0
Module weight (metric ton)	~700 ton
Seismic design	0.5g peak ground accelerations
Distinguishing features	Unlimited coping time for core cooling without AC or DC power, water addition, or operator action
Design status	Under regulatory review

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. All modules are managed from a single control room. Significant plant design features include: factory fabricated compact module, natural circulation coolant flow during all operational states, high design pressure containment vessel, use of established light-water reactor technology, and testing-based design development.

2. Target Application

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

3. Specific Design Features

(a) Design Philosophy

The NuScale plant design philosophy consists of: design simplification, use of proven light-water reactor technology, modular nuclear steam supply system, factory-fabricated power modules, 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.

(b) 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. Each power module is connected to a dedicated turbine-generator unit and balance-of-plant systems.

(c) Reactor Core

The core configuration for the NPM 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 ^{235}U enrichment is below the current U.S. manufacturer limit of 4.95 percent enrichment. The three-batch refuelling is conducted on a 24-month refuelling cycle.

(d) Reactivity Control

Reactivity control in each 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.

(e) Reactor Pressure Vessel and Internals

The RPV consists of an approximately cylindrical steel vessel with an inside diameter of approximately 3 m, an overall height of approximately 17.8 m, and is designed for an operating pressure of 12.8 MPa. The upper and lower heads are tori spherical and the lower portion of the vessel has flanges just above the core region 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 secondary system steam piping.

(f) Steam Generator

Each NPM uses two inter-woven 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 feedwater 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 primary coolant. The secondary side fluid is heated, boiled, and superheated to produce dry steam for the turbine-generator unit.

(g) Pressurizer

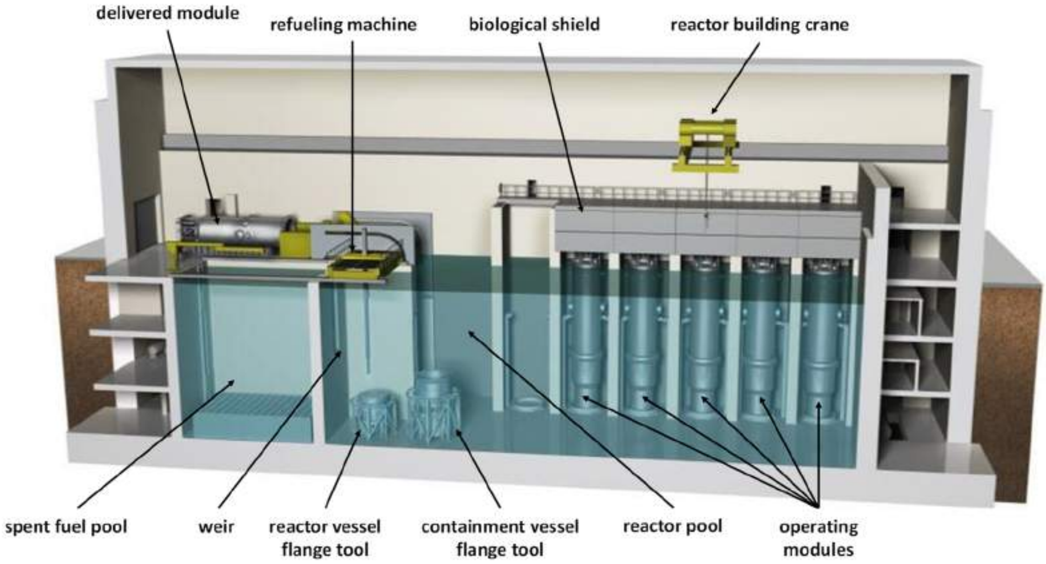
The internal pressurizer provides the primary means for controlling reactor coolant system pressure. It is designed to maintain a constant reactor coolant pressure during operation. Reactor coolant pressure is increased by applying power to a bank of heaters installed above the pressurizer baffle plate.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

4. Safety Features

The NuScale plant includes a comprehensive set of engineered safety features designed to provide stable long-term nuclear core cooling under all conditions, as well as severe accident mitigation. They include: integral primary system configuration, a high design pressure containment vessel, passive heat removal systems, and severe accident mitigation features. Probabilistic risk analyses indicate a core damage frequency that is orders of magnitude below that of currently operating plants.

(a) Engineered Safety System Approach and Configuration

Each NPM incorporates several simple, redundant, and independent safety features, which are discussed as follows:



Cut-away view of NuScale power plant

(b) Decay Heat Removal System

The decay heat removal system (DHRS) 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% of the decay heat load and cooling the primary coolant system. Each train has a passive condenser immersed in the reactor pool. During normal operations, the DHRS condensers are maintained with sufficient water inventory for stable and effective operation.

(c) Emergency Core Cooling System

Emergency core cooling system (ECCS) consists of three 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 defence-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.

(d) 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 4.5 m and an overall height of approximately 23.1 m. 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 an assured passive heat sink for containment heat removal under LOCA conditions.

5. 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 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 inspecting the module sections and refuelling the core, 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 one module is refuelled.

6. Instrumentation and Control Systems

The NuScale design includes a fully digital control system based on the use of field programmable gate array (FPGA) technology. The highly integrated protection system (HIPS) platform, which is already approved by the US Nuclear Regulatory Commission, is based on the fundamental I&C design principles of independence, redundancy, predictability, repeatability, diversity, and defence-in-depth. The HIPS platform is comprised of four module types that can be interconnected to implement multiple configurations to support various types of reactor safety systems. It also uses FPGA technology that is not vulnerable to internet cyber-attacks. The NuScale design effectively integrates human factors engineering (HFE) into the development, design, and operation of the plant.

7. Plant Arrangement

(a) Reactor Building

The NuScale plant consists primarily of a reactor building, a control room building, two turbine-generator buildings, a radiation waste treatment building, forced-draft cooling towers, a switchyard, and a dry-cast storage area for discharged fuel. The reactor building, shown in the figure above, consists of up to 12 power modules, module assembly/disassembly equipment, fuel handling equipment, and a spent fuel pool. Each NPM operates immersed within a common reactor pool in a separate bay with a concrete cover that serves as a biological shield. The reactor pool is located below grade and the reactor building is designed to seismic category 1 standards.

(b) Control Building

The main control room is housed below grade in the control building located adjacent to the reactor building. All NPMs are controlled from a single control room. The reactor operators monitor the automated control system for each reactor and common systems. 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.

(c) Balance of Plant

i. Turbine Generator Building

A NuScale plant has two separate turbine buildings, each housing up to six turbines and air-cooled generators. The turbine buildings are above-grade structures that house the turbine-generators with their auxiliaries, the condensers, condensate systems, and the feedwater systems. Each turbine-generator is associated with a single NPM and has dedicated condensate and feedwater pumps. Each condenser is located adjacent to the turbine-generator unit and is designed for 100% steam bypass of the turbine. An overhead crane is provided for the installation and maintenance in each turbine building.

ii. Electric Power Systems

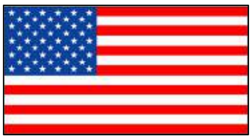
Under normal operating conditions the AC electrical power distribution system supplies reliable and continuous power to equipment required for startup, normal operation, and shutdown of the plant. The NuScale plant does not require off-site AC electrical power to cope with design-basis events. No backup 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.

8. Design and Licensing Status

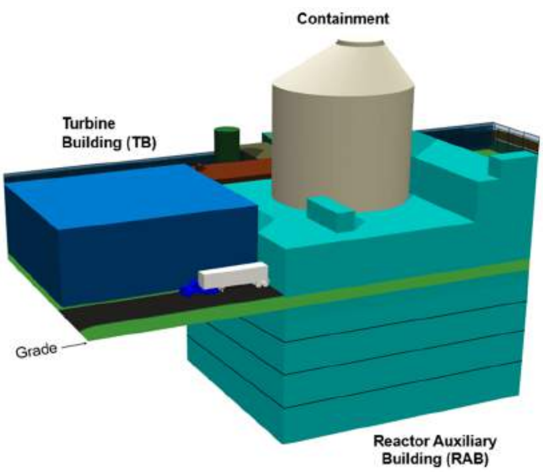
NuScale submitted a design certification application to the US Nuclear Regulatory Commission in January 2017. Phase 1 of the review was completed in April 2018 and design approval is expected in mid-2022. The first anticipated plant owner, the Utah Associated Municipal Power Systems, has a target commercial operation date of 2026 for the first plant that is expected to be built in Idaho.

9. Development Milestones

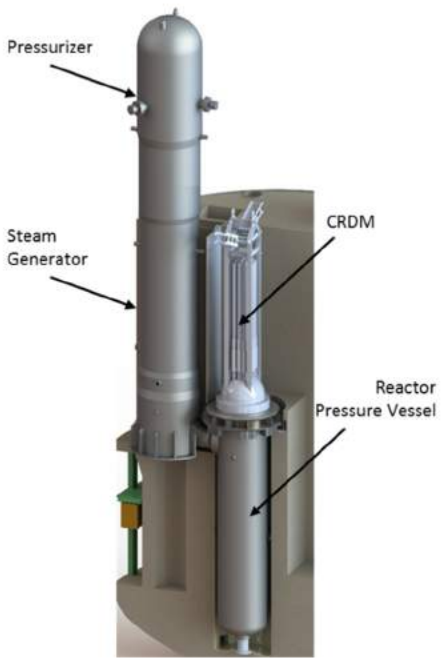
2003	Initial concept developed and integral test facility operational
2007	NuScale Power was formed
2011	Fluor Corporation became major investor and strategic partner for plant construction
2012	Twelve-reactor simulated control room was commissioned
2017	Design certification application was submitted to the US Nuclear Regulatory Commission
2018	NRC completed Phase 1 of design certification application review
2022	NuScale design certification approval expected
2026	First commercial NuScale plant targeted to be operational in Idaho



SMR-160 (Holtec International, USA)



SMR-160 nuclear island



SMR-160 reactor coolant system

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Holtec International, United States of America (Holtec)
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	525 / 160
Primary circulation	Natural circulation
System pressure (MPa)	15.5
Core inlet / exit temperatures (ºC)	209 / 321
Fuel type/assembly array	UO_2 pellet / square array
Number of fuel assemblies	112
Fuel enrichment (%)	4.95 (maximum)
Fuel burnup (GWd/ton)	45 (maximum, initial design)
Fuel cycle (months)	18 – 24 (flexible)
Main reactivity control mechanism	Fine motion control rod drive mechanism, with control rods
Approach to engineered safety systems	Numerous. Systems utilize diverse phenomena and are entirely passive.
Design life (years)	80
Plant footprint (m^2)	20500
RPV height/diameter (m)	15 / 3
Seismic design	The generic seismic design response spectrum is derived from NRC regulatory guide 1.60 spectra (modified), with a 0.3g PGA.
Distinguishing features	Active non-safety and passive safety cooling systems. Employs very robust protective structures, with critical components below grade. An extremely small source term diminishes the consequences of postulated accidents.
Design status	Preliminary design in progress, supporting commercial project development and pre-licensing engagements.

1. Introduction

The SMR-160 has been developed by Holtec International as an advanced PWR-type small modular reactor producing 525 MW thermal power or 160 MW electric power. The plant design incorporates both passive and extremely robust safety systems to enhance its safety and operation, which results in a highly reliable design that protects an owner’s investment from all manner of postulated accidents, sabotage, or inadvertent human actions. The SMR-160 design is “walk-away safe” – no operator actions are necessary to place the plant in a safe condition for design basis accidents and safely reject decay heat. The design cures LWR common failure mechanisms, eliminating fuel fretting failures with a fully turbulent low velocity primary, eliminating reactor vessel internal SCC through removal of tensile structural welds, and a simpler cleaner water chemistry without boron. The plant is greatly simplified relative to conventional plants to improve its fabricability, constructability, and maintainability, in part, facilitated by incorporating entirely passive safety systems and a natural circulation primary loop. Further simplification is achieved by using fewer valves, pumps, heat exchangers, instrumentation, and control loops than conventional plants, which reduces capital cost of the plant. Reduction of inspection, test, and maintenance requirements and functions reduces operating costs. The SMR-160 uses fuel very similar to existing commercial LWR product lines, includes no reactor coolant pumps and utilizes a large vertical once-through steam generator. A modular construction plan for the SMR-160 involves fabrication of the largest shippable components prior to arrival at a site. A 24-month construction period is envisaged for each nth-of-a-kind unit.

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). The design is readily configurable for dry siting locations though use of Holtec International's air-cooled condenser technology. The SMR-160 is capable of both "black-start" and isolated operation, rendering the plant ideal for destinations with unstable power grids or off-grid applications.

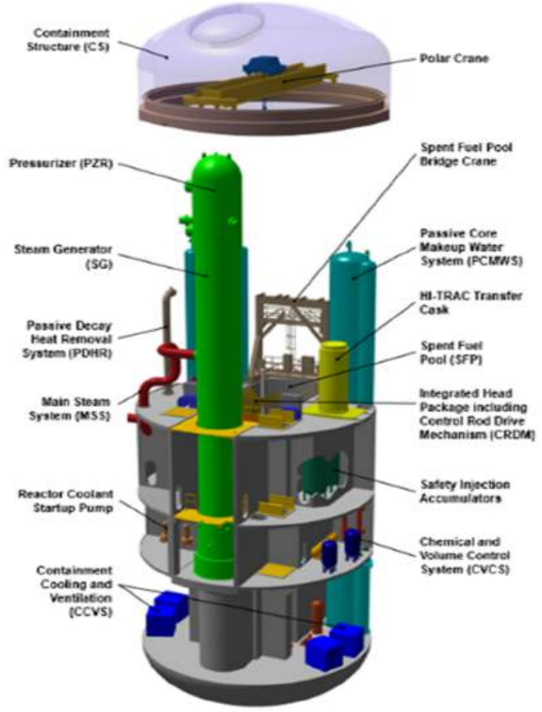
3. Specific Design Features

(a) Design Philosophy

The SMR-160 design philosophy is driven by the principal criterion of achieving unparalleled safety without reliance on active systems or operator actions, while ensuring the SMR-160 design remains inherently securable, fabricable, constructible, and economically competitive in world-wide markets.

(b) Nuclear Steam Supply System

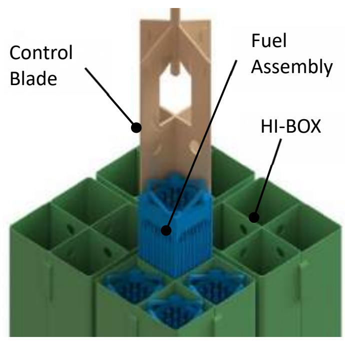
The SMR-160 is a pressurized water reactor with a reactor coolant system (RCS) with a natural circulation primary loop. The RCS is comprised of the reactor pressure vessel (RPV) and a steam generator (SG) in an offset configuration with an integrated pressurizer flanged to the top of the steam generator. The RPV and the SG are connected by a single connection which houses both the hot leg and the cold leg. Unique among integral PWRs, the offset configuration allows easy access to the core without moving the RPV or SG during refuelling. Due to the high SG superheat there is no need for a moisture separator reheater (MSR) or multiple trains of feedwater heaters. The secondary loop includes one stage of feedwater heating and eliminates high-pressure turbine stages.



SMR-160 containment internals

(c) Reactor Core

The SMR-160 employs an efficient reactor core design that uses traditional reload shuffle. The reactor core contains a matrix of fuel rods assembled into mechanically identical fuel assemblies along with control and structural elements. The reactor vessel internals support the reactor core, the control rod assemblies, and the control rod drive shafts. The fuel, reactor vessel internals, and coolant are contained within a heavy walled RPV. The core is designed for a nominal two-year cycle with flexibility for shorter cycles depending upon utility requirements. The SMR-160 core is based on proven LWR technology and operational characteristics and is designed to ensure large margin to thermal-mechanical fuel limits. Most importantly, it should be noted that while there are some innovative aspects of the core design, it largely remains in the realm of currently licensed technology using well known and available materials.



SMR-160 core

(d) Reactivity Control

Long term reactivity control is provided by burnable absorbers integral to the fuel which 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. No chemical shim (e.g. Boron) is used for reactivity control. Control blades are positioned by fine motion control rod drive mechanisms (CRDM) based on existing electro-mechanical technology. The CRDMs are located outside the reactor coolant system on the RPV upper head.

(e) Reactor Coolant System

The SMR-160 RCS operates purely by gravity, as reactor coolant circulates entirely through 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, a reactor pressure vessel (RPV), a steam generator (SG), and an integral pressurizer.

(f) Reactor Pressure Vessel and Internals

The RPV is an ASME Section 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 enables the use of traditional external control rod drive mechanism and greatly simplifies refuelling operations relative to typical integral PWR designs. There are no penetrations to the RPV below the elevation of the safety injection lines. The reactor internal structures are designed to be supported from the bottom of the vessel and are completely replaceable.

(g) Steam Generator

The SMR-160 includes a single, vertically-oriented, once through straight tube SG with the reactor coolant flowing inside thermally-treated Inconel 690 tubes. The use of straight tubes ensures easy access for in-service inspection. The SG uses sub-cooled feedwater to produce superheated steam on the shell side. The SG features a large inventory of secondary water on the shell side which provides substantial margin to dry-out.

(h) Pressurizer

The pressurizer is integral to the steam generator and uses heaters and cold-water spray nozzles to perform the functions of a typical pressurizer. Integrating the pressurizer with the steam generator eliminates significant primary piping, along with the typical supporting structures normally connecting the primary external loop of a PWR to an external pressurizer and reactor coolant pumps. The large relative size of the Pressurizer eliminates any need for Power Operated Relief Valves (PORVs).

4. Safety Features

The SMR-160 safety basis incorporates defence-in-depth via multiple and diverse pathways for rejection of decay heat. All safety systems are protected with a robust containment enclosure structure, rendering them secure and safe from external threats. A large inventory of water within a reservoir between the enclosure and the containment vessel provides long-term post-accident coping and allows the structure to transition to air cooling for decay heat removal for an unlimited coping period.

(a) Engineered Safety System Approach and Configuration

SMR-160 relies on gravity-driven passive safety systems to ensure the plant is maintained in a safe configuration for all postulated design basis and beyond design basis events. The SMR-160 approach is to employ non-credited, non-safety active systems as the first line of defence, backed by passive safety systems which will ensure safe shutdown and decay heat removal for an unlimited period without the need for power, make-up water, or operator actions. This approach ensures that the plant achieves ultimate safety while simultaneously ensuring recovery of the plant after an event.

(b) Passive Core Cooling System (PCCS)

The PCCS is designed to provide emergency core cooling during postulated accidents. The system uses passive means such as natural circulation, gravity injection, and compressed gas expansion for core makeup and cooling without the use of active components such as pumps. The PCCS is comprised of four major subsystems: (i) Primary decay heat removal system (PDHR); (ii) Secondary decay heat removal system (SDHR), (iii) Automatic depressurization system (ADS), and (iv) Passive core make-up water system (PCMWS). The PDHR directly cools the primary coolant by rejecting the heat to a second loop full of water, which rejects heat to the large annular reservoir around containment (AR). The SDHR provides an alternative and diverse passive means to reject decay heat. The SDHR is a closed loop system that relies on buoyancy driven flow to route steam from the SG to a heat exchanger in the AR, where the steam then condenses and rejects its latent heat. Condensate is then returned to the shell side of the SG. The ADS is a depressurization system designed to safely let down RCS pressure to permit staged safety injection by the PCMWS and permit long-term recirculation within the containment vessel.

(c) Containment and the Passive Containment Heat Removal System (PCHR)

The SMR-160 containment system consists of a free-standing steel containment structure (CS), enclosed within a reinforced concrete containment enclosure structure (CES). The CES provides shielding and protection from external events. The CES walls are constructed of extremely robust steel-concrete modules designed to withstand an impact from large commercial aircraft and other hazards. In addition to preventing the release of radioactive fission products to the environment, the containment system acts as a large passive heat exchanger. All engineered safety features are located within the containment system. The containment system is partially embedded, with half of the total height located below grade to maximize protection against external hazards and dampen seismic effects for critical components.

The PCHR passively cools the containment volume, without any required actuations. During a postulated high energy release, steam rejects heat to the inner wall of the containment, condensing as heat is transported to the AR. The large heat transfer area and high conductance of the metal containment wall results in near-instantaneous heat rejection to the AR. The AR then rejects heat to the environment. The large inventory of water in the AR is sufficient to extract energy from the containment for over 3 months without replenishment, enabling the PCHR to transition to air cooling and ensure safe decay heat removal indefinitely. The containment houses both the reactor and the spent fuel pool. The spent fuel pool is a large open volume of water designed to passively reject decay heat through surface evaporation to the containment volume in the event of loss of active cooling. This allows safe and indefinite cooling of the spent fuel in the same manner as the fuel within reactor, rendering the plant immune to spent fuel related events, such as those at the Fukushima nuclear power plant.

5. Plant Safety and Operational Performances

The SMR-160 natural convection-driven reactor coolant loop is coupled with an optimized simple steam cycle and is well adapted to load following. Refuelling operations take advantage of industry operational experience to limit the required number of heavy lifts, use traditional core shuffling techniques, and incorporate all necessary inspections and maintenance.

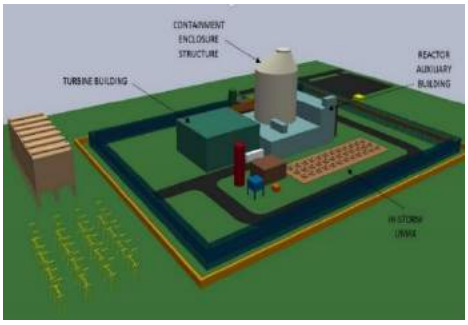
6. Instrumentation and Control Systems

SMR-160 utilizes the Mitsubishi Electric Total Advanced Controller (MELTAC) platform for the plant's I&C/HSI design. MELTAC is a proven technology, with over 300 years of operating reactor experience. MELTAC provides unique nuclear specific I/O and configuration flexibility to perform all nuclear safety and non-safety functions using the same digital platform. 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), controllers, sensors, processors, field cables, operators, and plant staff by eliminating most periodic safety system surveillance and including on-line condition monitoring. SMR-160 additionally employs an independent diverse actuation system design to make a common cause failure event non-credible for the plant.

7. Plant Arrangement

(a) Containment Enclosure Structure

The SMR-160 reactor is housed within a containment structure (CS) protected by a containment enclosure structure (CES). Nearly half of the CS and CES is embedded underground. These structures share a common basemat with the reactor auxiliary building and house all safety related systems and spent fuel.



Plant layout of SMR-160

(b) Reactor Auxiliary Building

The reactor auxiliary building houses many of the plant auxiliary systems. It contains the new and spent fuel handling facilities as well as the control room complex. This building is designed to process spent fuel for interim on-site storage within Holtec International HI-STORM UMAX modules (an underground dry cask storage technology), without any modification to the standard design.

(c) Balance of Plant

The steam turbine and associated systems are housed within the turbine building structure at grade. The SMR-160 features an axial/side exhaust steam turbine, optionally configured for air cooled condensation.

The SMR-160 electric power system consists of the main generator, main transformer, auxiliary transformers, non-safety diesel generators and safety related 1E batteries. The power supply to the plant AC power system during normal plant operation is provided from the main generator. The electrical system is designed to permit isolated operation in "island-mode" as well as start-up operations independent of the grid or "black-start."

8. Design and Licensing Status

Pre-application activities for the SMR-160 have commenced with multiple international regulators in parallel with development of commercial project opportunities. The project execution plan projects initial operation of the first deployed reactors by the mid-2020s.

9. Development Milestones

2012	Conceptual design of SMR-160 commencement
2015	Conceptual design completed for SMR-160
2019	Preliminary design completed for SMR-160
2020	Completion of pre-application activities and commencement of commercial project licensing

Westinghouse SMR (Westinghouse Electric Company LLC, USA)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse Electric Company LLC, USA
Reactor type	Integral PWR
Coolant/moderator	Light water
Thermal/electrical capacity, MW(t)/MW(e)	800/>225
Primary circulation	Forced circulation
System pressure (MPa)	15.5
Core inlet/exit temperatures (°C)	294/324
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies	89
Fuel enrichment (%)	< 5
Fuel burnup (GWd/ton)	> 62
Fuel cycle (months)	24
Main reactivity control mechanism	CRDM, boron
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	65000
RPV height/diameter (m)	28/3.7
Seismic design	Based on CEUS sites
Distinguishing features	Incorporates passive safety systems and proven components of the AP1000 plant and earlier Westinghouse designs
Design status	Concept design completed

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(t) 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.

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. Specific Design Features

(a) 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.

(b) 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.

(c) 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.

(d) 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.

(e) 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 refuelling.

(f) 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 refuelling operations through a bolted closure flange near the top of the integral reactor vessel.

4. Safety Features

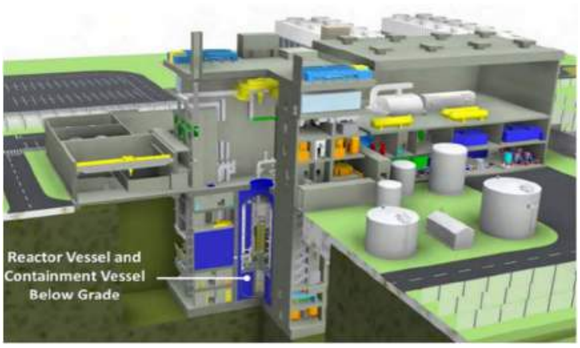
The Westinghouse SMR is an advanced passive plant where the safety systems are designed to mitigate accidents through the use of natural driving forces such as gravity flow and natural circulation flow. The plant is not reliant on alternating current (ac) power or other support systems to perform its 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.

(a) Engineered Safety System Approach and Configuration

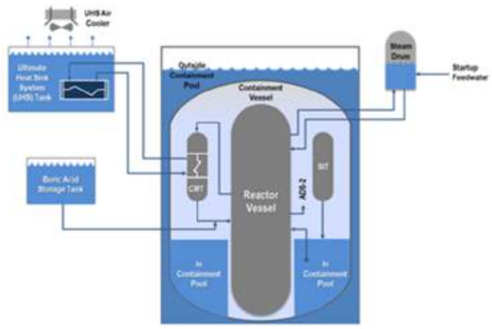
The Westinghouse SMR is designed with passive safety systems that utilize the natural forces of evaporation, condensation, and gravity. The design basis and licensing of passive systems were first implemented in the design of the AP1000 plant. Elements of these systems are described in the following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

(b) 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 ultimate heat sink (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.



Below-grade location of Westinghouse SMR reactor and containment vessels



Three diverse decay heat removal methods of the Westinghouse SMR

(c) 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.

5. 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.

6. 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.

7. Plant Arrangement

(a) Reactor Building

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.

(b) 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.

Plant layout of Westinghouse SMR

i. 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.

ii. Electric Power Systems

The Westinghouse SMR onsite power system consists of a main ac power system and a direct current (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 Class 1E and non-Class 1E dc power systems. Safety-related dc power is provided to support reactor trip and engineered safeguards actuation. Batteries are sized to provide the necessary dc power and uninterruptible ac power for items such as 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.

8. Design and Licensing Status

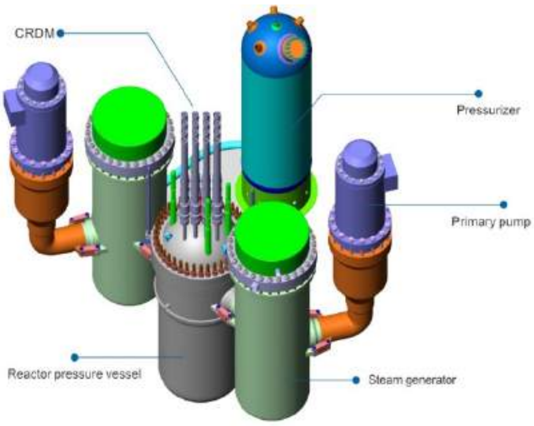
The Westinghouse SMR concept design has made substantial progress in support of U.S. and UK licensing. Westinghouse is considering a number of business models for the successful deployment of the Westinghouse SMR product globally. In addition, in February 2015, the U.S. Nuclear Regulatory Commission (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.

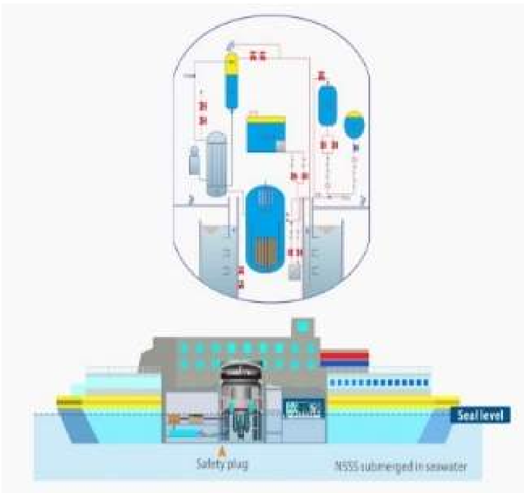
9. Development Milestones

2015	Conceptual design completed

WATER COOLED
SMALL MODULAR REACTORS
(MARINE BASED)

ACPR50S (CGN, China)





Reactor building inside ship

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	China General Nuclear Power Group (CGNPC), People's Republic of China
Reactor type	Loop type PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	200/50
Primary circulation	Forced circulation
System pressure (MPa)	15.5
Core inlet/exit temperatures (°C)	299.3/321.8
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies	37
Fuel enrichment (%)	< 5
Fuel burnup (GWd/ton)	< 52
Fuel cycle (months)	30
Main reactivity control mechanism	Control rod driving mechanism (CRDM), solid burnable poison and boron solution
Approach to engineered safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	--
RPV height/diameter (m)	7.2/2.2
Seismic design	--
Distinguishing features	Floating power boat, once-through steam generator, passive safety system
Design status	Completion of conceptual/program design, preparation of project design

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 flexible 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.

2. Target Application

As 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 for islands together with solar energy and wind power.

3. Specific Design Features

(a) Design Philosophy

The ACPR50S adopts design simplification to reduce cost and investment risks to be competitive with conventional offshore energy sources. Modular design is adopted through standardized and streamlined manufacturing, aiming for shorter construction period as well as lower cost. A long refuelling cycle allows for higher load factors.

(b) 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 reactor coolant pumps (RCP) and a pressurizer (PZR), all of which are interconnected by short reactor coolant system legs. The compact layout of the primary loop reduces the probability of LOCA significantly. The primary cooling system is based on forced circulation during normal operation. The system has natural circulation capability and heat removal capacity up to 10% thermal power.

(c) Reactor Core

The low power density design with a low 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 37 fuel assemblies (FAs) of ACPR50S core, with an axial length of 2.2 m, have a square 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.

(d) Reactivity Control

Core reactivity is controlled by means of control rods, solid burnable poison and soluble boron dispersed in the primary coolant. Burnable poison rods flatten the 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).

(e) Reactor Pressure Vessel and Internals

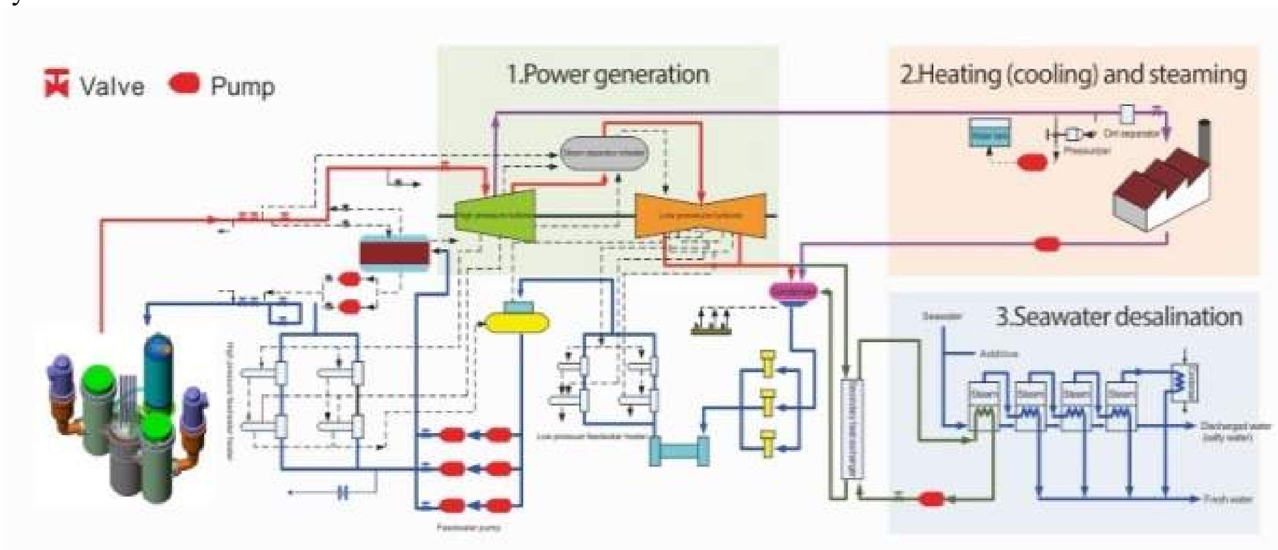
ACPR50S reactor pressure vessel is 2.2 m in diameter and 7.2 m high. It envelopes and fixes the core and the RPV internals, so that the fission reaction of the nuclear fuel is limited in one space.

(f) Reactor Coolant System

The ACPR50S primary cooling 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, as 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.

(g) 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 active and passive secondary residual heat removal system (ASHR & PSHR), which remove the decay heat from the primary system.



Energy cascade supply

(h) 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 normal plant operation 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).

4. 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.

(a) Engineered Safety System Approach and Configuration

The ACPR50S is designed with passive safety systems that 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.

(b) 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. This decay heat is ultimately removed by the cooling water in the cooling water tank outside the containment eventually. If power supply is still available and, if the normal decay heat removal pathway fails under non LOCA accident, the feed water system will be started instead of the SHR.

(c) Emergency Core Cooling System

The SIS is a very important engineered safety system which is developed to cool and boride the core following DBA and extends the no operator action time of plant operators to 7 days. Its main function is to control and mitigate the consequences of the accident and prevent the DBA from becoming a serious beyond design basis accident (BDBA). The core cooling is completely driven by the natural force following the DBA such as LOCA, which simplifies the system's composition and operation greatly. 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, medium-pressure and low-pressure safety injection, ADS is used.

(d) Containment System

The containment system is to contain radioactive material and protect the environment against primary coolant leakage. The containment system of ACPR50S is 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 0.3 MPa. 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. 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 that can lead to a pressure rise inside containment 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 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.

5. Plant Safety and Operational Performances

ACPR50S is suitable for all kinds of sea states such as swing, concussion, vibration, typhoon, seaquake and so on. The operational power range can be justified from 20% to 100% Pn (Power (nuclear)) and can be operated steadily for a long period for any power level of 20% to 100% Pn to satisfy the power demand.

6. Instrumentation and Control Systems

The I&C system design for ACPR50S is based on defence in depth concept, compliance with the single failure criterion and diversity. 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, one technical support centre, and two separate remote shutdown stations to ensure control and operation of the plant under normal and accident conditions.

7. Plant Arrangement

A single reactor module with the electrical, the steam generator, and auxiliary nuclear facilities is installed inside a non-propelled barge 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, radiation waste treatment are installed in the onshore basement.

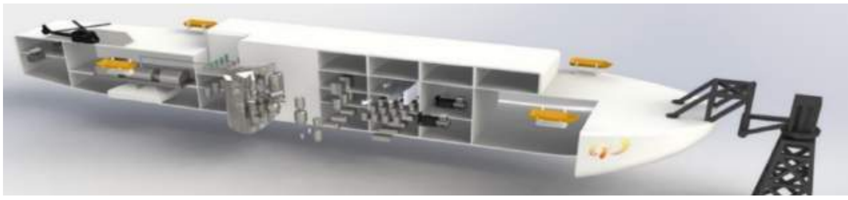
(a) Floating Platform

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. The figure below 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.

(b) Onshore Basement

The onshore basement of ACPR50S houses the fuelling building, the radiation waste treatment building, and other balance of plant (BOP) buildings. Refuelling and overhauling is performed in the onshore basement.

General plant arrangement in the float platform



(c) Control room

The compact control room is designed for one-man operation under normal conditions of the plant and is located in the ship (offshore). 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 simulator.

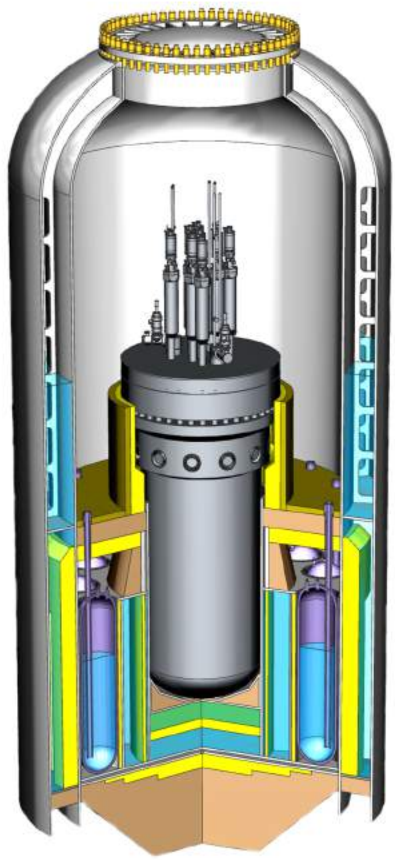
8. Design and Licensing Status

The ACPR50S has completed the preliminary design work and is now preparing for detail design. The National Nuclear Safety Administration（NNSA） is performing the safety review for this small modular reactor. The ACPR50S plans to apply for the construction permit to the NNSA in 2019. An industrial demonstration plant of ACPR50S is being planned to be constructed in China, with a target of startup commissioning in 2022.

9. 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;
2016-2019	Project Design Stages
2016	Completion of design software suitability evaluation; starting Preliminary Design; Completion of Codes and Standards study
2017-2018	Completion of Preliminary Design and Preparation of Construction; Starting key equipment procurement for Project Construction Stage
2018-2019	Completion of PSAR review; Obtaining construction permit; Starting construction design
2019-2021	Project Construction Stage
2020	Submission of FSAR and EIA; Completion of construction and installation; and obtaining fuel loading permit
2021~	Start-up commissioning and connection to grid

ABV-6E (Afrikantov OKBM, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Afrikantov OKBM, Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	38/6~9
Primary circulation	Natural circulation
System pressure (MPa)	16.2
Core inlet/exit temperatures (^{o}C)	250/325
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies	121
Fuel enrichment (%)	<20
Fuel burnup (GWd/ton)	N/A
Fuel cycle (months)	120-144
Main reactivity control mechanism	Control rod driving mechanism
Approach to engineered safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	20000 (basic design)
RPV height/diameter (m)	6/2.4
Seismic design	7 per Richter scale (basic design)
Distinguishing features	Natural circulation in the primary circuit
Design status	Final design

1. Introduction

The ABV-6E is reactor plant (RP) as a part of nuclear power system (NPS) that produces 14 MW(t) 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 PWR reactors and recent achievements in the field of nuclear power plant (NPP) safety. The main objective of the project is to develop small, shipyard fabricated, multipurpose transportable NPP for safe operation over 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.

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.

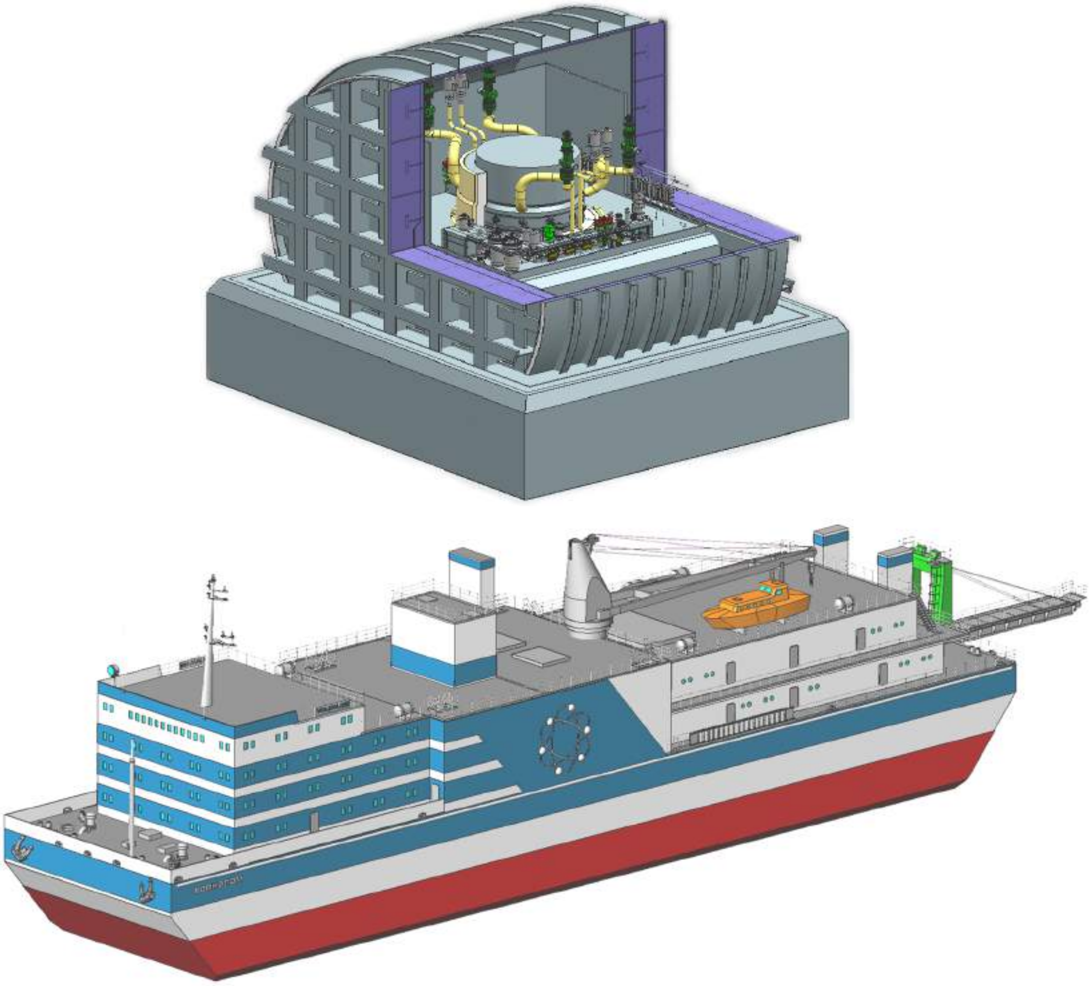
3. Specific Design Features

(a) 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%.



FPU includes 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.

(b) 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 less than 20% enriched uranium-235. Special stainless steel is used as fuel cladding.

(c) 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).

(d) 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.

(e) 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.

4. Safety Features

(a) 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 use of natural circulation in the primary coolant circuit and passive safety systems as primary safety systems. The autoprotective 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.

(b) Residual Heat Removal System

In emergency modes, a combined-type residual heat removal system (RHRS) 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 RHRS, there are no active RHRS channels in the ABV 6E reactor design, which allows the output of emergency power supply sources to be reduced. The passive RHRS 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 RHRS functions, i.e. of maintaining the parameters of the primary circuit in the design limits for an unlimited time.

(c) 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.

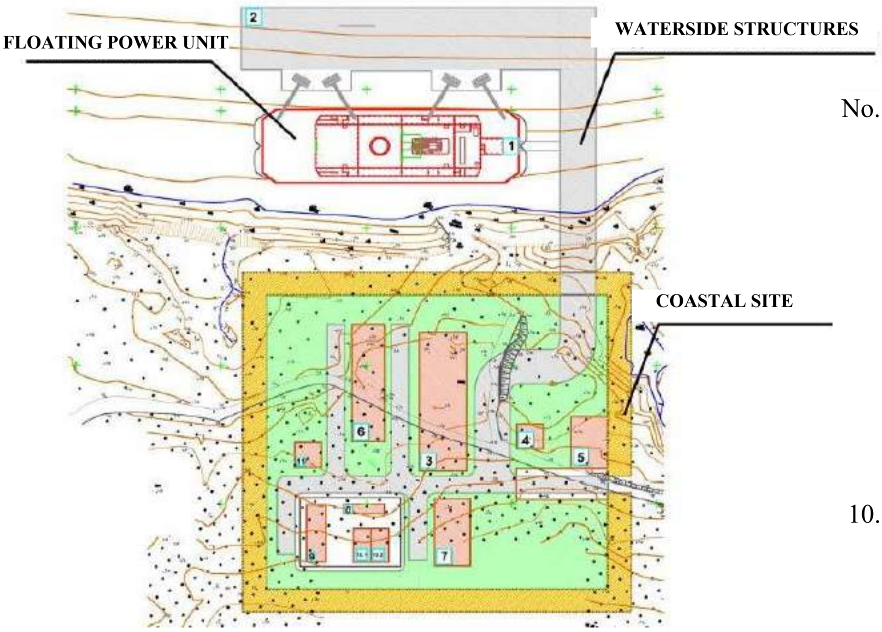
(d) Containment System

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.

5. Plant safety and Operational Performances

The NPP with ABV-6E generates electricity and heat in the power range of 20–100 %Nnom with the continuous operation time of 26 000 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.

6. Plant Arrangement



No. in plan	Title
1	Floating power unit
2	Waterside structures
3	Integrated process building
4	Treatment facilities
5	Access control point
6	Guard room
7	Administrative building/shelter
8	Indoor switchgear
9	Substation-shared control room
10.1, 10.2	Diesel-generator sets
11	Garage

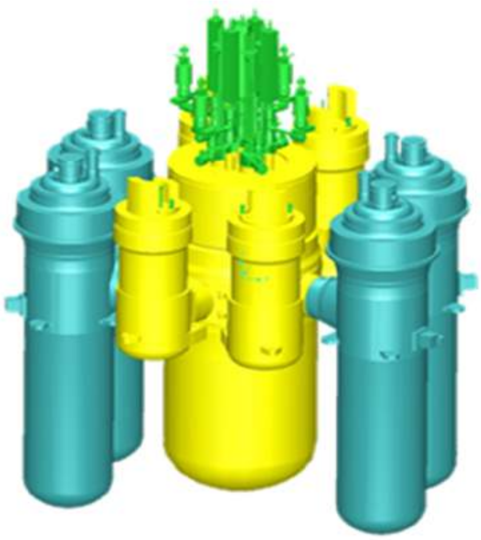
7. Design and Licensing Status

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

8. Development Milestones

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)

KLT-40S (Afrikantov OKBM, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Afrikantov OKBM, Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	150/35
Primary circulation	Forced circulation
System pressure (MPa)	12.7
Core inlet/exit temperatures (°C)	280/316
Fuel type/assembly array	UO_2 pellet in silumin matrix
Number of fuel assemblies	121
Fuel enrichment (%)	18.6
Fuel burnup (GWd/ton)	45.4
Fuel cycle (months)	30-36
Main reactivity control mechanism	Control rod driving mechanism
Approach to engineered safety systems	Active (partially passive)
Design life (years)	40
Plant footprint (m^2)	--
RPV height/diameter (m)	4.8/2.0
Seismic design	9 point on the MSK scale
Distinguishing features	Floating power unit for cogeneration of heat and electricity; onsite refuelling not required; spent fuel take back to the supplier
Design Status	Under tests, planned commercial start 2019-2020

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.

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.

3. Specific Design Features

(a) 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 delivered fossil fuel is used.

(b) 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.

(c) 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 refuelling of reactor after every 2.5–3 years of operation. Refuelling is performed 14 days after reactor shutdown when the levels of residual heat releases from spent FAs have reached the required level. The spent nuclear fuel is initially stored on board at the FNPP and then returned to Russian federation. No special maintenance or refuelling ships are necessary. Single fuel loading is done in order to provide maximum operation period between refuelling. The fuel is loaded in the core all at once with all fuel assemblies being replaced at the same time.

(d) 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.

(e) 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.

(f) 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).

4. 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, etc. Passive cooling channels with water tanks and in-built heat exchangers ensure reliable cooling to 24 hours.

(a) Engineered Safety System Approach and Configuration

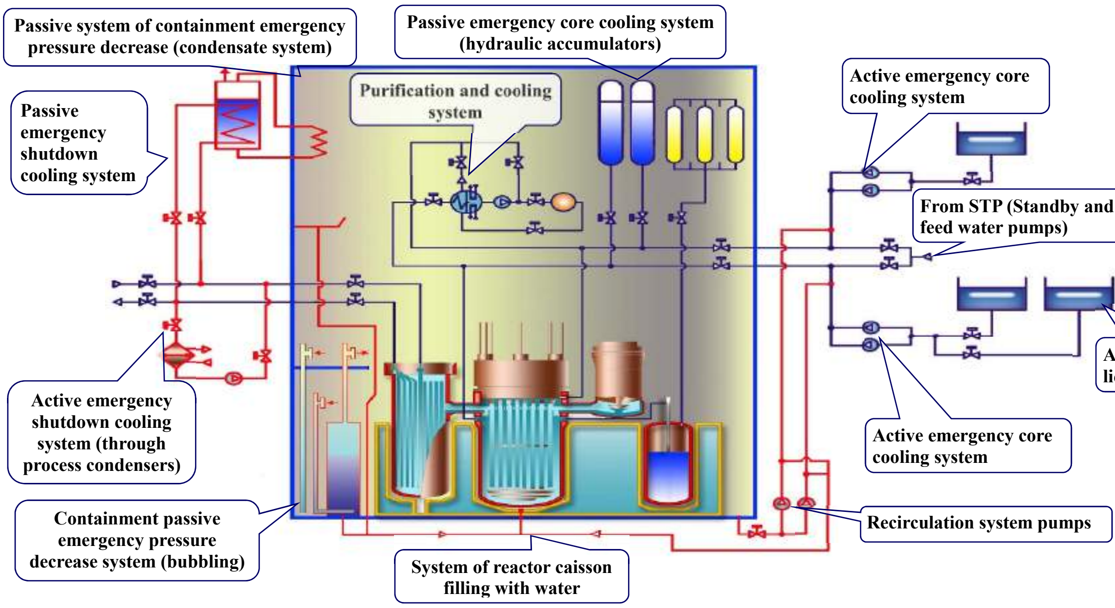
The active components of the protection system are scram actuators for six groups of the control rods.

(b) 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 two secondary passive cooling channels via steam generators, one active secondary cooling channel via steam generators and one active cooling channel via the primary/third heat exchanger.

(c) Emergency Core Cooling System

The 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 and 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 high-pressure ECCS subsystem with makeup, high-pressure ECCS subsystem with hydraulic accumulators and Low-pressure ECCS subsystem with recirculation pumps.



(d) 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 refuelling or maintenance operations.

5. Plant Safety and Operational Performances

The KLT-40S NPP ensures electricity and heat generation within the power range of 10% to 100% for a continuous operation of 26,000 hrs. 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 FPU and NPP buildings are designed to withstand the crash of an aircraft of 10 tons. Based on analysis, the radiation emission limits are satisfied for all conditions.

6. Electric Power Systems

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.

7. Plant Arrangement

The coastline line of the FCNPP has the complex engineering building with equipment to distribute and transmit electricity to loads and to prepare and transfer heating water to loads and 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.

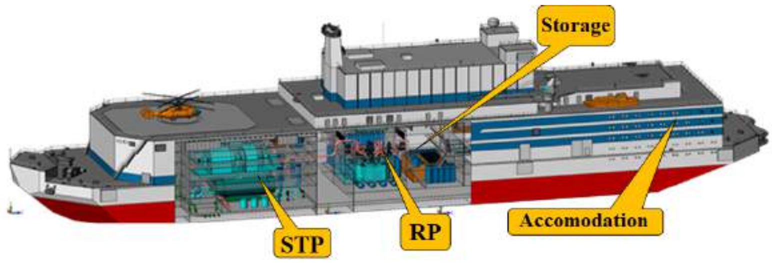
(a) Reactor Building

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 electrotechnical 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. Onboard the FPU, storages for spent cores and means are arranged that ensure reactor reloading.

(b) 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.

General cross-section view of the FPU



(c) 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.

8. 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 construction of The Akademik Lomonosov was completed in 2017. The Akademik Lomonosov will be commissioned in 2019 in the town of Pevek in Chukotka region.

9. Development Milestones

1998	The first project to build a floating nuclear power plant was established
2002	The environmental impact assessment 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 was selected as the site for the installation of NPPs. JSC "Baltiysky Zavod" undertook charge of construction, installation, testing and commissioning the first FPU
2017	Completion of construction and testing of the floating power unit at the Baltic shipyard
2018	Dock-side trials, fuelling, final tests completion with reactor core, attainment of reactor's first criticality
2019	Transportation of FNU to the town of Pevek
2019-2020	Commercial startup

RITM-200M (Afrikantov OKBM, Russian Federation)

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RITM-200 Reactor

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Afrikantov OKBM, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	175/50
Primary circulation	Forced circulation
System pressure (MPa)	15.7
Core inlet/exit temperatures (°C)	277/313
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies	241
Fuel enrichment (%)	<20
Fuel burnup (GWd/ton)	-
Fuel cycle (months)	120
Main reactivity control mechanism	Control rod driving mechanism
Approach to engineered safety systems	Combined active and passive
Design life (years)	60
Plant footprint (m^2)	-
RPV height/diameter (m)	9.2 / 3.5
Seismic design	0.3g
Distinguishing features	Integral reactor, in-vessel corium retention, double containment.
Design status	Reactors manufactured and installed in icebreakers, floating NPP under development

1. Introduction

RITM series reactors RITM-200 and RITM-200M are the latest development in III+ generation SMR line designed by the OKBM Afrikantov and has incorporated all the best features from its predecessors. RITM series reactor based floating NPP are a referenced technology and currently are at a rather advanced development stage and will be available for commercial implementation in short/medium term. RITM-200 based floating NPP modularity enables to provide power both for commercial/local needs (for example 100 MW) and regional needs (300 MW). RITM-200M reactor is the further RITM series development with increased refuelling cycle up to 10 years.

Being initially designed for icebreakers, RITM series reactors are already responding to the key market requirements, such as small size, long refuelling cycle and flexible load following capabilities (100-30-100 % N_{nom}). Six RITM-200 reactors have been already manufactured for nuclear icebreakers. Four reactors are already installed on «Sibir» and «Arktika» icebreakers. Regarding these factors RITM series reactors are the most optimized solution in terms of its size and other technical and economic parameters.

2. Target Application

RITM series reactors are multi-purpose application reactors for electrical power generation, water desalination, and district or industrial heating. RITM-200 reactor is the heart of three new universal nuclear icebreakers manufactured to provide all season navigation along the Northern Sea Route. RITM series reactor based floating NPPs are suitable for small, isolated, or distributed grids.

3. Specific Design Features

(a) Design Philosophy

RITM series reactors are the evolutionary development of the reactors (OK-150, OK-900, KLT-40 series) for Russian nuclear icebreakers with a total operating experience of more than 50 years (more than 400 reactor-years). By incorporating the steam generators into the reactor pressure vessel, the reactor system and containment is very compact compared to the KLT-40. RITM design makes it possible to increase electric output (40% more) and reduces the dimensions (45% less) and the mass (35% less) in comparison with KLT-40S. While integral reactor configuration almost eliminates the classic large loss-of-coolant accident (LOCA) and the other inherent features and active and passive safety systems apply concepts of diversity, redundancy, physical separation, and functional independence to achieve the necessary safety level and reliability.

(b) Nuclear Steam Supply System

RITM series nuclear steam supply system consists of the reactor core, four steam generators integrated in the reactor pressure vessel, four canned main circulation pumps (MCP), and pressurizer. The primary cooling system is based on forced circulation during normal operation and allows natural circulation for emergency condition.

(c) Reactor Core

RITM series adopts a low enriched cassette core similar to KLT-40S that ensures long time operation without refuelling and meets international non-proliferation requirements. The core consists of 241 fuel assemblies with fuel enrichment up to 20%. The core has the assigned service life of 11.5 years.

(d) Reactivity Control

Control rods are used for reactivity control. A group of control rods drive mechanisms is intended to compensate for the excessive reactivity at start up, power operation and reactor trip. A group of shutdown rods is designed for fast reactor shut down and to maintain it in the subcritical condition in case of accident. The design of control and shutdown rods is based on the drives used in KLT-40S reactor.

(e) Reactor Coolant System

The reactor pressure vessel (RPV) is thick-walled cylindrical pressure vessel with an integrally welded bottom head and a removable top head. The reactor is designed as an integral vessel with the main circulation pumps (MCP) 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 generate steam of 295°C at 3.82 MPa flowing at 261 (280) t/h. The conventional MCPs are used. The pump is vane, single step, and has a sealed asynchronous electric motor with one winding.

(f) Steam Generator

The RITM uses once through (straight tube) SGs. The configuration of the steam generating cassettes makes possible to compactly install them in the RPV.

(g) 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 requires no electric power. 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.

4. Safety Features

The safety concept of the RITM is based on the defence-in-depth principle combined with the inherent safety features and use of passive systems. Properties of inherent safety features 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 optimally combines passive and active safety systems to cope with abnormal operating occurrences and design basis accidents.

- Passive pressure reduction and cooling systems have been included (system reliability is confirmed by test bench);
- 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 MCP failures.

The exposure dose for the staff in normal operation and design basis accidents does not exceed 0.01% of the natural radiation limit. The public exposure dose in case of severe accident is below the value requires protective measures.

(a) Engineered Safety System Approach and Configuration

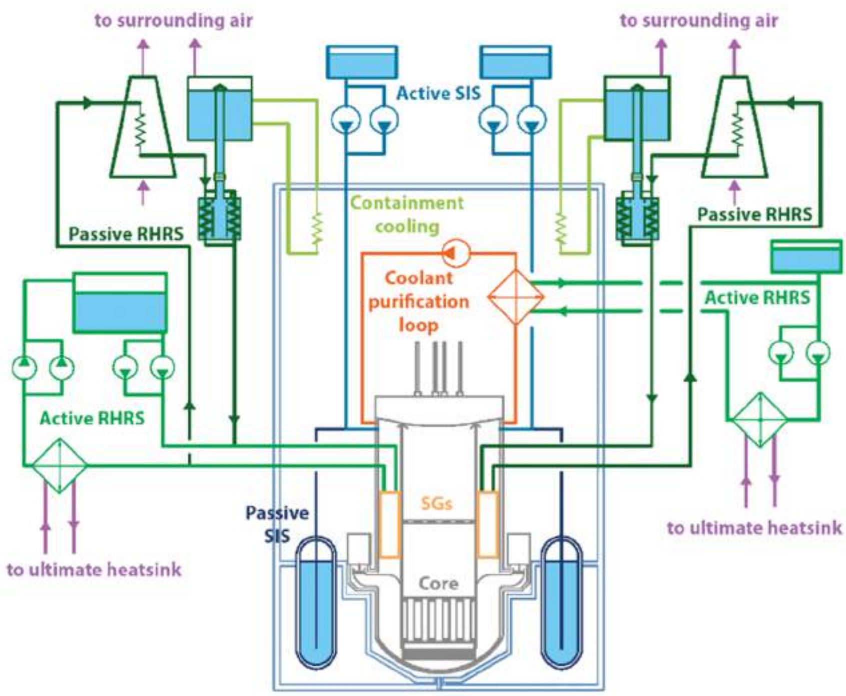
The high safety level of RITM series reactors is achieved both by inherent safety features and a combination of passive and active safety systems. Moreover, redundancy of safety system equipment and channels and their functional and/or physical separation are provided to ensure high reliability. Safety 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 pressure to ensure reactor trip and initiate the safety systems. Shutdown rods drop into the core by gravity with spring assist when power is removed from electromechanical clutches consequently ensure reactor shutdown even in case of total station black out.

(b) Residual Heat Removal System

The residual heat removal system (RHRS) consists of four safety trains:

- Active safety loop with forced circulation through steam generator.
- Active safety loop with forced circulation through primary-third circuit heat exchanger of primary circuit coolant purification loop.
- Two passive safety loops with natural coolant circulation from water tanks through steam generators. Evaporated in steam generators, water condenses in air cooled heat exchangers and flow back to tanks with water heat exchangers. After complete water evaporation from the tanks, the air-cooled exchangers continue provide cooling for unlimited time. Combination of air and water heat exchangers allows to minimize dimensions of the heat exchangers and water tanks.

All safety train are connected to different steam generators and provide residual heat removal in compliance with the single failure criterion. Active safety trains consist of water tank, pumps, and heat exchanger to ultimate heatsink.



RITM series safety systems

(c) Emergency Core Cooling System

The emergency core cooling system consists of safety injection system (SIS) for water injection in primary circuit to mitigate the consequences of a break loss-of-coolant accident. The system is based on active and passive principles with redundancy of active elements in each channel and consists of:

- Two passive pressurized hydraulic accumulators;
- Two active channels with water tanks and two make-up pumps in each channel.

In combination with the residual heat removal system the passive safety trains anticipate a post-accident grace period of 72 hours without operator action or power in case of combination of LOCA and total station blackout.

(d) Containment System

RITM is placed within the hermetically sealed envelope with dimensions of 6m×6m×15.5m (6.8m×6.8m×16m) around the reactor vessel to localize possible radioactive releases. In case of severe accident thick wall of the reactor vessel keeps molten corium within the reactor. Water filled caisson under the reactor provides the reactor vessel cooling. The envelope integrity is ensured by overpressure relief valve, containment cooling system, and passive autocatalytic recombiners.

5. Instrumentation and Control Systems

An automated control system is provided in the RITM based 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 control of the power plant.

6. Plant Arrangement

Floating nuclear power plant with RITM series reactors is designed to supply electricity, thermal power, and desalinated water to coastal or isolated territories, offshore installations, islands, and archipelagos. Spent nuclear fuel storage is removed due to extended refuelling cycle. Floating NPP can be rapidly delivered to the site by sea. The only needs to launch operation is docking pier and onshore power transmitting infrastructure.

Floating NPP with RITM reactors

7. Design and Licensing Status

The RITM design was developed in conformity with Russian laws, norms and rules for nuclear power plants and safety principles developed by the world community and IAEA recommendations. The RITM design adopts optimal combination of passive and active safety systems. Reactors are currently manufactured and installed in nuclear icebreakers, with the floating NPP design under development.

8. Development Milestones

2012	Detailed design of RITM-200
2016	First RITM-200 installed in Arktika icebreaker
2016	Conceptual design of floating NPP with RITM-200M
2018	Start of detailed design of floating NPP with RITM-200M
2020	Arktika and Sibir icebreakers with RITM-200 in service

SHELF (NIKIET, Russian Federation)

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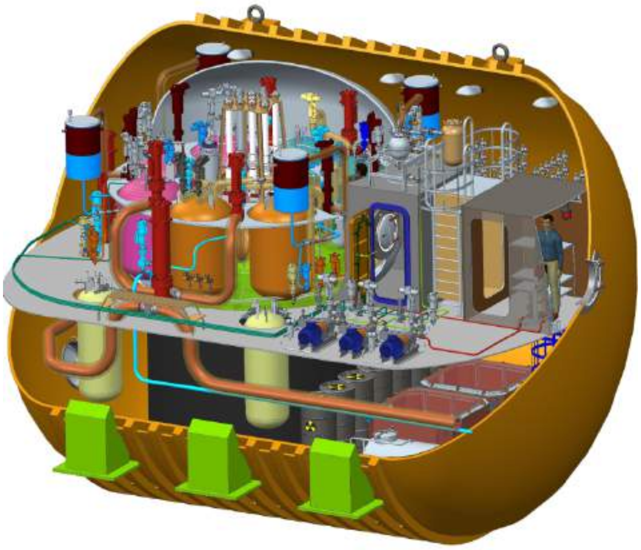
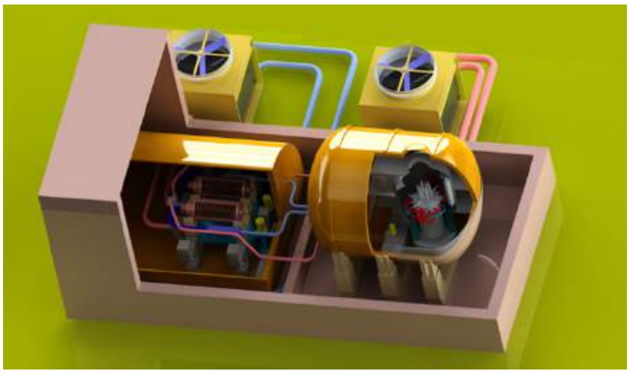




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	28.4/6.6
Primary circulation	Forced and natural circulation
System pressure (MPa)	14.7
Core inlet/exit temperatures (ºC)	270/310
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies	163
Fuel enrichment (%)	19.7
Fuel burnup (GWd/kg)	up to 160
Fuel cycle (years)	6
Main reactivity control mechanism	Control rod driving mechanism
Approach to engineered safety systems	Combined active and passive
Design life (years)	60
Plant footprint (m^2)	8 000
RPV height/diameter (m)	14/8 (12/8 RF only)
Seismic design	SSE 8 (MSK-64)
Distinguishing features	Power source for users in remote and hard-to-reach locations; can be used for both floating and submerged nuclear power plants
Design status	Detailed design underway

1. Introduction

A power unit based on the SHELF reactor is designed as a local power source for users in remote and hard-to-reach locations. The power unit is a power capsule to generate 6.6 MW(e). The power capsule is developed in two options: containing only all reactor components, and the capsule of a bigger size that includes also 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 as both, floating and submerged nuclear power plants. Engineering design of SHELF is similar to a large degree to the 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 refuelled, exhausted equipment is repaired or replaced during its lifecycle, and decommissioned 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 eliminates the need for plant operators inside the power unit and keeps the module unmanned during the automated operation period equal to 8000 hours. Within the period of automated operation, the scheduled maintenance is conducted every year within 15 days of duration.

2. Target Application

SHELF reactor is designed for the 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 is 6.6 MW(e) while thermal power is 28.4 MW(t).

Depending on the consumer requirements 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 desalination plant with a capacity of 500 m^3/h of fresh water.

In case of land based plant, the heat is removed by external heat exchangers cooled by mechanical air pumping. The SHELF plant does not require local water sources. Decay heat removal in case of submerged deployment is provided by sea water cooling.

3. Specific Design Features

(a) Design Philosophy

SHELF is a water-cooled reactor of integral layout with a combined (forced and natural) coolant circulation modes. The fuel campaign is six (6) years with the design capacity factor of 80%. The components for SHELF are assembled inside a cylindrical power capsule with an inner diameter of 8 m and a length of 14 m (dimensions are given for the option of reactor plant only). The reactor equipment is installed in the rear portion of the capsule and the TGP equipment is installed in its front. Besides power capsule, there is a compact module housing the auxiliary systems including an air conditioning system for module inner space 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.

(b) Reactor Core

The core is of the heterogeneous cartridge type and consists of 163 hexahedral fuel assemblies (FA) of three different types. A number of FAs contains burnable poison and control absorber rods. 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 nuclear power plant are as follows:

- Potentially higher nuclear fuel burn-up (up to 160 GWd/ton);
- 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 present 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 generation of corrosive oxidative radiolysis products. 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 h. 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 refuelling according to the adopted concept is performed on the specialized enterprise basis. In-service fuel handling is not envisaged.

(c) 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 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 and instrumentation and control system. 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.

(d) 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).

(e) 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.

(f) 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.

4. 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:

2. A defence-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 and 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:

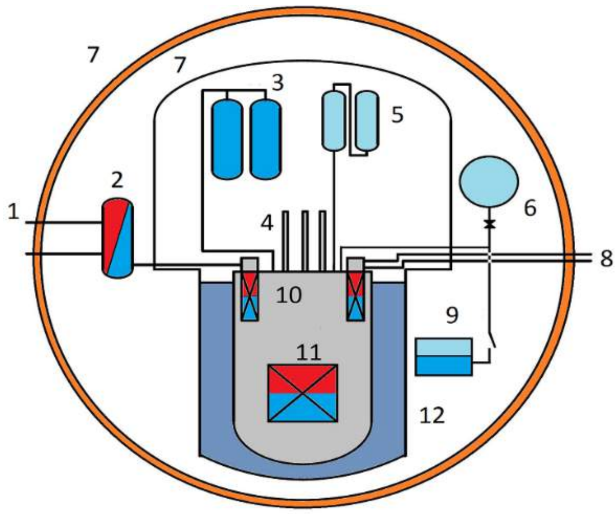
High reliability level of safety systems is achieved through implementing the following principles:

- Passive operation not requiring any actions for initiation; 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.

Key systems of the SHELF reactor facility:
1) To atmosphere;
2) ACS;
3) Pressurizer;
4) Shim group and EP;
5) ECCS;
6) Makeup;
7) Containment;
8) To TGP;
9) Absorber tank;
10) SG;
11) Core;
12) IBS



5. Plant Safety and Operational Performances

The electric power of one SHELF unit is 6.6 MW(e), and the thermal power is 28.4 MW(t). 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.

6. Instrumentation and Control Systems

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:

- Normal operation comprises of 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 40% Nnom with natural primary coolant circulation, switchovers from one steady power level to another in the above power ranges at a preset rate, switchover from natural primary coolant circulation to forced circulation and scheduled automated deactivation.
- Anticipated operational occurrences like 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 and 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.
- 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.

7. 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.

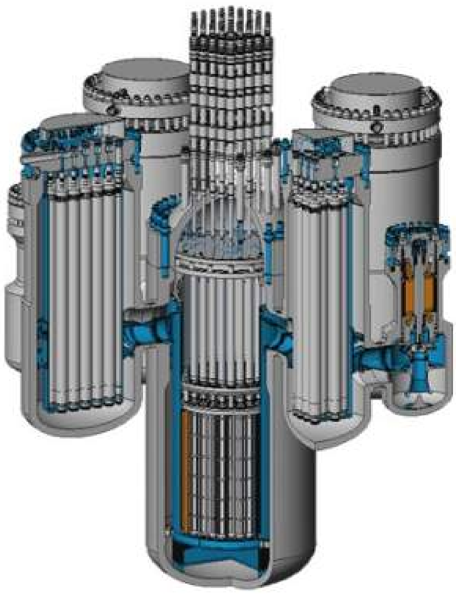
8. Design and Licensing Status

The licensing of the nuclear plant design is scheduled for 2019-2020.

9. 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
2017 - 2019	Detailed design of power capsule with SHELF reactor facility only (without TGP inside) with an electric power of 6.6 MW at NIKIET

VBER-300 (Afrikantov OKBM, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Arikantov OKBM, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	917/325
Primary circulation	Forced circulation
System pressure (MPa)	16.3
Core inlet/exit temperatures (^{o}C)	292/328
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies	85
Fuel enrichment (%)	4.95
Fuel burnup (GWd/ton)	50
Fuel cycle (months)	72
Main reactivity control mechanism	Control rod driving mechanism and soluble boron
Approach to engineered safety systems	Hybrid (active and passive) system
Design life (years)	60
Plant footprint (m^2)	--
RPV height/diameter (m)	9.3/3.9
Seismic design	0.25 (g)
Distinguishing features	Power source for transportable FNPPs, cogeneration options, compact design
Design status	Licensing stage

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.

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. Specific Design Features

(a) 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. The VBER-300 reactor unit incorporates the reactor and four steam generators – MCPs 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.

(b) Nuclear Steam Supply System

The separation heat exchangers are designed to extract heat energy from the nuclear heat source without mixing the fluids circulating within the nuclear plant with those employed in the process heat application. In the VBER-300 design, separation heat exchangers are thermally coupled indirectly via heat exchangers coupled with the secondary loop supporting the power conversion system. In this configuration, a stream of the steam generated via steam generators (for any of the 2, 3, and 4 SGs) and partially expanded in the turbines is extracted at an intermediate pressure for circulation within the separation heat exchangers.

(c) Reactor Core

The reactor core comprises 85 hexagonal fuel assemblies (FAs) which are placed in the reactor cavity in nodes of a regular triangular lattice with a space interval of 236 mm. Pelletized UO_2 fuel with an enrichment of up to 5% licensed and tested in VVER reactors is used. FA of unified design to increase the fuel efficiency is utilized. The VBER-300 design concept allows a flexible fuel cycle for the reactor core with standard VVER FAs. The fuel cycles are 3*2 years and 4*1.5 years. The number of FAs in the refuelling batch is either 15 or 30; maximal fuel burnup does not exceed 60.0 GWd/ton U for the cycle with 30 fresh FAs in the reloading batch and maximum initial uranium enrichment. Each FA contains guide tubes that allow insertion/withdrawal of control rods. Reactor core also uses gadolinium fuel elements, which contain gadolinium in the UO_2 fuel pellet and has the same geometry as the regular fuel pellet.

(d) 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.

(e) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) consists of the reactor core and internals with an overall height of 9.3 m and a diameter of 3.9 m. 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.

(f) 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.

(g) 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.

(h) 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.

4. 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).

(a) 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.

(b) 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.

(c) 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.

(d) Containment System

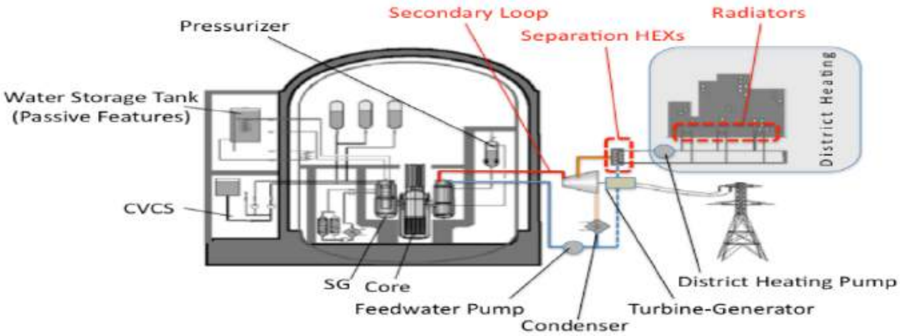
The land-based VBER-300 containment system includes a double protective pressure envelope formed by an inner carbon steel shell and an outer reinforced concrete containment structure. In addition, localizing reinforcement is provided to protect the pressure boundary represented by all auxiliary systems 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.

5. Plant Safety and Operational Performances

The VBER-300 safety concept is based on the state-of-the-art defence-in-depth principles. 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.

6. 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.



(a) Reactor Building

The inner steel shell is a leak-tight cylindrical enclosure 30 m in diameter that is covered with the semi-spherical 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 semi-spherical dome. Building structures of the outer protective enclosure are designed for external accidental exposures, including an aircraft crash and air shock wave.

(b) 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—are housed in a relatively small area.

i. 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.

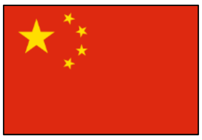
7. Design and Licensing Status

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 design.

8. 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.
2007–2009	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
2008–2011	R&D for the VBER-460/600 NPP design
2011–2012	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

HIGH TEMPERATURE
GAS COOLED
SMALL MODULAR REACTORS



HTR-PM (Tsinghua University, China)

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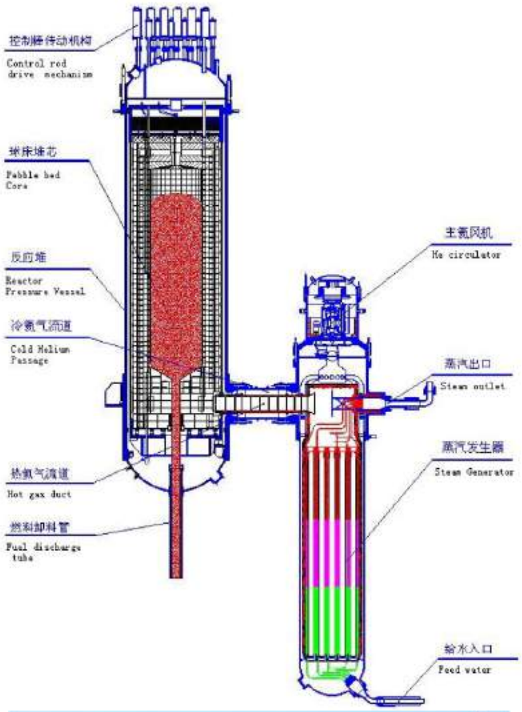


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	INET Tsinghua University, People’s Republic of China
Reactor type	Modular pebble bed high temperature gas-cooled reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	2x250/210
Primary circulation	Forced circulation
System pressure (MPa)	7
Core inlet/exit temperatures (^{o}C)	250/750
Fuel type/assembly array	Spherical elements with coated particle fuel
Number of fuel spheres	420,000 (in each reactor module)
Fuel enrichment (%)	8.5
Fuel burnup (GWd/ton)	90
Fuel cycle (months)	On-line refueling
Main reactivity control mechanism	Control rod insertion
Approach to engineered safety systems	Combined active and passive
Design life (years)	40
Plant footprint (m^2)	--
RPV height/diameter (m)	25/5.7(inner)
Seismic design	0.2 (g)
Distinguishing features	Inherent safety, no need for offsite emergency measures
Design status	Under construction

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 many experiments on the HTR-10 to verify crucial inherent safety features of modular HTRs, including (i) Loss of off-site power without scram; (ii) Main helium blower shutdown without scram; (iii) Withdrawal of control rod without scram; and (iv) Helium blower trip without closing outlet cut-off valve.

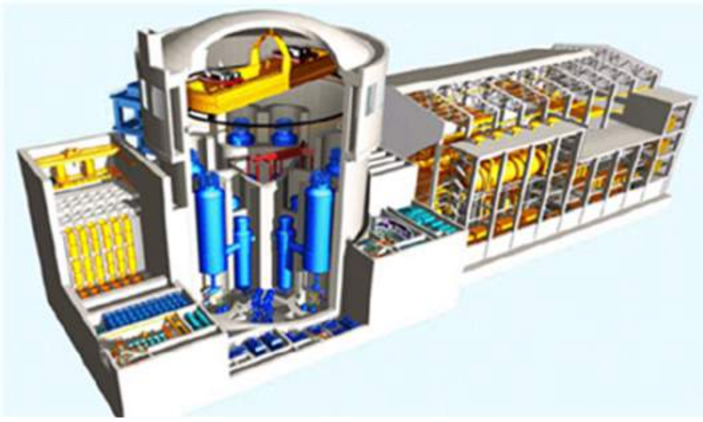
The second step of HTR development in China began in 2001 when the high-temperature gas-cooled reactor pebble-bed module (HTR-PM) project was launched. The first concrete of the HTR-PM demonstration power plant was poured on December 9, 2012, in Rongcheng, Shandong Province. In support of manufacturing first of a kind equipment and licensing, large scale engineering facilities were constructed and all tests have been completed. The civil work of the nuclear island’s buildings has been completed in 2016 with the first of two reactor pressure vessels installed in March 2016. Currently all major equipment has been manufactured and, except for the steam generators, already installed. The power plant is scheduled to start power generation in 2019.

2. Target Application

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.

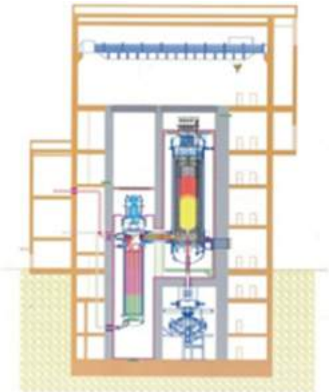
A conceptual design has been finished for a 600 MW(e) multi-module HTR-PM nuclear power plant, which consists of six reactor modules coupling to one steam turbine. Each reactor module has the same design as the HTR-PM demonstration plant, with independent safety systems and shared non-safety auxiliary systems. The footprint of a multi-module HTR-PM plant is not remarkably different from that of a PWR plant generating the same power. The figure below shows the 600 MW(e) HTR-PM nuclear power plant layout. Future sites have been identified for possible deployment.

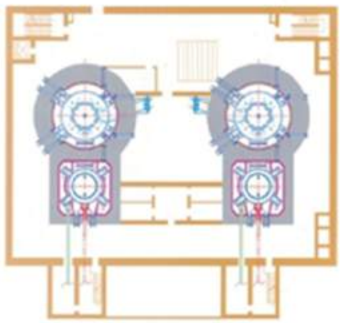


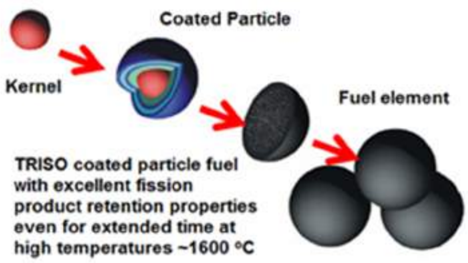
3. Specific Design Features

(a) Design Philosophy

The HTR-PM consists of two pebble-bed reactor modules coupled with a 210 MW(e) steam turbine, as shown below. 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(t), 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.







(b) Reactor Core and Power Conversion Unit

The primary helium coolant works at the pressure of 7.0 MPa. The rated mass flow rate is 96 kg/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.

(c) Fuel Characteristics

Fuel elements are spherical ones. Every fuel element contains 7 g heavy metal. The enrichment of U-235 is 8.5%. Uranium kernels of about 0.5 mm 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 5 cm in diameter. Surrounding the fuel containing graphite matrix is a 5 mm thick graphite layer. This is illustrated above.

(d) 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. When a fuel sphere reaches the target burnup they will be discharged into the spent fuel storage tank, otherwise they are re-inserted into the reactor to pass the core once again.

(e) 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.

(f) 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 all are 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.

4. Safety Features

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.

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.

(a) Engineered Safety System Approach and Configuration

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.

(b) 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.

(c) 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 1620ºC 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.

5. Plant Safety and Operational Performances

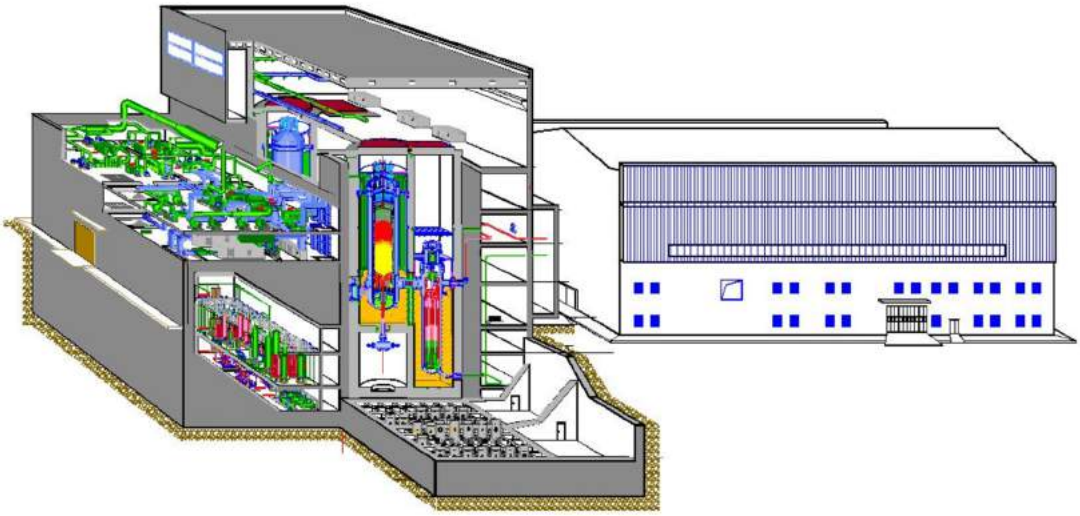
The HTR-PM demonstration power plant is under construction. Due to online fuel loading mode of HTR-PM, better availability factor can be expected compared with other power plants operating in a mode of periodic fuel loading.

6. 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.

7. Plant Arrangement

The nuclear island contains reactor building, nuclear auxiliary building, spent fuel storage building and I&C building, as shown below. The steam turbo-generator, which is similar to that of a conventional fossil-fired power plant, is housed in the turbine building.



8. 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. Final approval of the FSAR is expected soon.

9. 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	Fuel plant achieved expected production capacity
2019	First operation expected

GTHTR300 (Japan Atomic Energy Agency, Japan)

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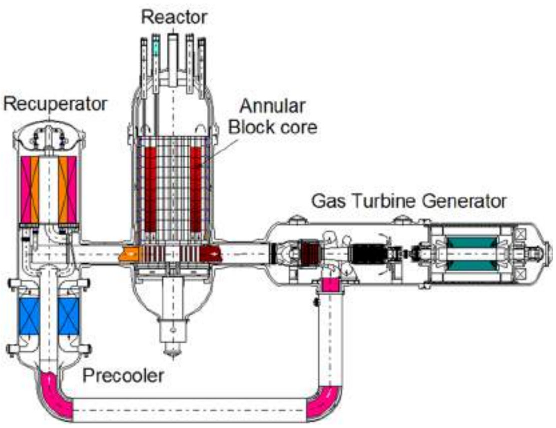
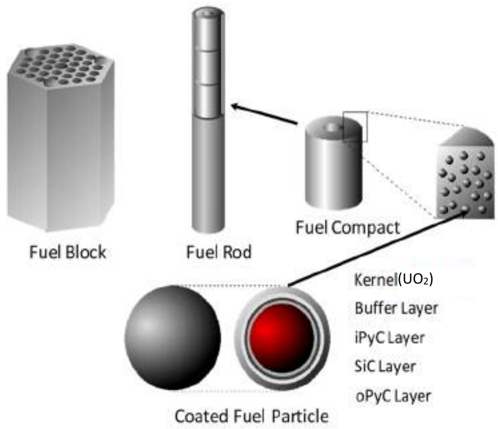




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JAEA jointly with MHI, Toshiba/IHI, Fuji Electric, KHI, NFI, etc, Japan
Reactor type	Prismatic HTGR
Coolant/moderator	Helium
Thermal/electrical capacity, MW(t)/MW(e)	<600/100~300
Primary circulation	Forced by gas turbine
System pressure (MPa)	7
Core inlet/exit temperatures (°C)	587-633/850-950
Fuel type/assembly array	UO_2 TRISO ceramic coated particle
Number of fuel columns	90
Fuel enrichment (%)	14
Fuel burnup (GWd/ton)	120
Fuel cycle (months)	48
Main reactivity control	Control rod insertion
Design life (years)	60
RPV height/diameter (m)	23/8
Seismic design	>0.18 g automatic shutdown
Distinguishing features	Multiple applications of power generation, H_2 production, process heat, steelmaking, desalination, district heating
Design status	Pre-licensing basic design completed

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 2030s. 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 as listed in the table above. The design employs a direct-cycle helium gas turbine to simplify the plant by eliminating water and steam systems while delivering 45-50% power generating efficiency. The design incorporates 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 users, in particular to industries, so as to minimize cost and loss of high temperature heat transmission. Dry cooling becomes economically feasible due to high temperature (above 150°C) heat rejection from the gas turbine cycle, making inland and remote siting possible without a large source of cooling water.

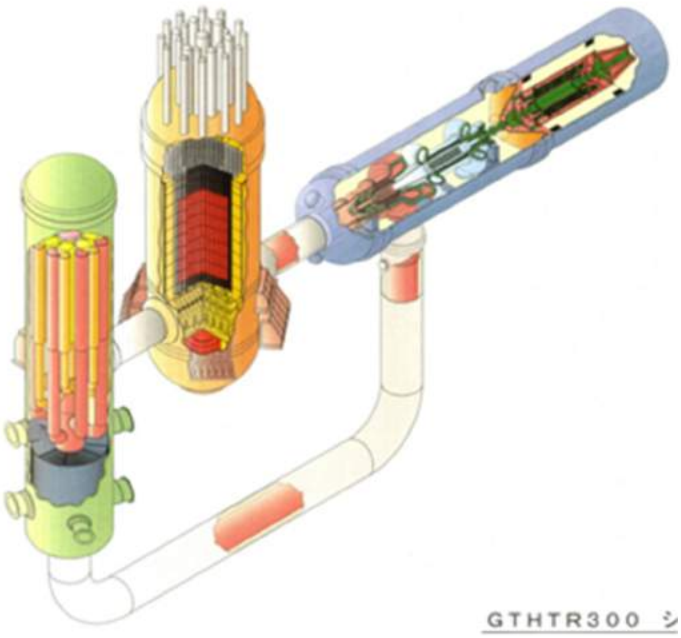
2. Target Applications

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

3. Specific Design Features

(a) Design Philosophy

The overall goal of the GTHTR300 design and development 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 by pursuing system simplicity, economic competitiveness and originality, namely the SECO design philosophy. More specifically, all design variants are built on the premise that they share common system technologies to the maximum extent possible, including 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 for hydrogen production. The helium gas turbine and the IS process, for which JAEA has been carrying out research and development, are particularly compatible with high temperature helium cooled reactor providing the basis for high efficiency and economical production. The design includes such distinct attributes as conventional steel reactor pressure vessel, horizontal gas turbine installation, and the three-vessel system arrangement (see figure below) among others, reduce requirement and risk of technology development. Since the technologies are shared by several systems, the benefit of investing in any one development is increased.



(b) Reactor Core and Power Conversion Unit

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 by adding an IHX between the reactor and gas turbine that can accept the various roles of cogeneration while sharing equipment designs with GTHTR300. While the reactor technologies required for the GTHTR300 are developed mainly with construction and operation of JAEA's 30 MW(t) and 950^{o}C test reactor, we are separately developing and testing balance of plant key 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 super alloy heat exchanger capable of transferring 950^{o}C reactor heat to the hydrogen production process.

(c) Fuel Characteristics

The fuel design is coated fuel particle of less than 1 mm in diameter. Each particle consists of a UO_2 kernel coated by four layers of low and high density pyro-carbon and silicon carbide. The all ceramic particle fuel is heat resistant up to 1600^{0}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.

(d) Fuel Handling System

The fuel handling system consists of fuel loading machines, door valves, a control rod exchange machine, and a transport carriage. The fuel loading machine is used to remove fuel blocks from core and load fuel blocks to core and spent fuel storage facility. The door valves are devised at the interface between fuel loading machine and spent fuel storage facility to maintain airtightness and radiation shielding performance. The control rod exchange machine is installed for removal of control rods from reactor and loading of used control rods to maintenance pit. The transport carriage is used for the transportation of the fuel loading and control rod exchange machines.

(e) Reactivity Control

Reactivity control system consists of control rods, control rod drive mechanisms and reserve shut down systems. The system is used to adjust control rod position for reactivity control as well as shut down reactor in case of reactor scram. GTHTR300 has 30 pairs of control rod and reserve shut down systems. The control rods and reserve shut down system channels are located in the reflector blocks on the inner and outer boundaries of the fuel region.

(f) Reactor Pressure Vessel and Internals

The reactor core consists of graphite hexagonal blocks, one-third of which are fuel blocks arranged in an annular region while the other two-thirds are reflector blocks arranged inside and outside of the fuel region. Each fuel block has 57 coolant holes with fuel rods forming annular-shaped coolant channels. A permanent reflector which surrounds side replaceable reflectors contains coolant channels. The helium coolant from the reactor inlet is introduced to the channel. A core barrel is installed between reactor internals and reactor pressure vessel to support internal structure laterally. A flow path is devised to introduce low temperature helium coolant from the primary system between RPV and core barrel part in order to isolate the RPV from high temperature helium coolant. The flow configuration enables to apply conventional carbon steel SA533/SA508 for reactor pressure vessel.

4. Safety Features

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

- The ceramic coated particle fuel maintains containment integrity under a temperature limit of 1600oC.
- 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.

(a) Engineered Safety System Approach and Configuration

An engineered safety system in the GTHTR300 consists of a reactor cavity cooling system and confinement. GTHTR300 safety design is based on philosophy of maintaining safety functions relying on inherent and passive safety features. Accordingly, the system is designed not to rely on active components or operator actions in principle.

(b) Decay Heat Removal/ Reactor Cooling Philosophy

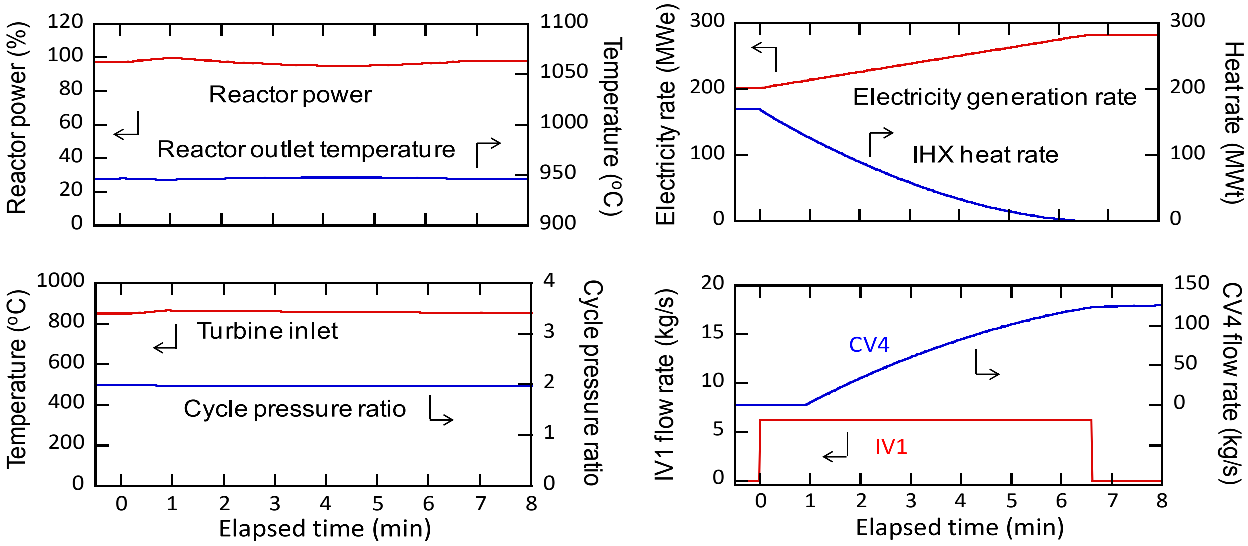
GTHTR300 design is required to remove the core decay and residual heat from the outside surface of the reactor vessel by the natural convection and radiation, and to transfer it to an ultimate heat sink in the operational states and in the accident conditions so that the design limits for fuel, the reactor coolant pressure boundary and structures important to safety are not exceeded. In accordance with the requirement, a low power density core with annular configuration of the fuel region is employed to maintain the fuel temperatures below the design limits during passive decay heat removal. A reactor cavity cooling system, an air-cooled, passive decay heat removal system is used to transfer heat received from reactor pressure vessel to atmosphere.

(c) Containment Function

GTHTR300 employs a vented confinement rather than a conventional high-pressure, airtight containment used in LWRs. The confinement is designed to meet requirements designated in safety design. The confinement is designed to release helium coolant blown out from primary system in case of depressurized loss-of-forced cooling accidents. Dampers are devised and recloses when the pressure between outside and inside of confinement equalizes. The design leak rate is set to 20%/day to limit the amount of air ingress to the reactor core.

5. Plant Operational Performance

The ability to follow variable power and heat loads is simulated as shown in the following figure. The simulation shows the plant response to an electric demand increase of 5%/min with corresponding reduction in heat rate, which is the maximum required for ramp load follow. The reactor remains at 100% power at all times. Starting from a base cogeneration ratio of 203 MW(e) electricity and 170 MW(t) net heat, the turbine power generation is increased to follow the increasing electric demand by increasing primary coolant inventory with opening of inventory valve IV1. The IHX heat rate is lowered by lowering secondary loop flow of the IHX. As the primary exit temperature of the IHX begins to rise, the valve CV4 is opened to direct cold flow from the compressor discharge to mix with the hot exit gas of the IHX primary side with the goal to maintain turbine inlet temperature at the rated 850°C. The power sent out to external grid increases to 276 MW(e) in as little as a few minutes. The pressure in the reactor increases to 7 MPa from 5 MPa. To return to the base cogeneration state, the control would be reversed by reducing primary coolant inventory and closing the bypass valve VC4.



6. Instrumentation and Control Systems

Instrumentation and control system consists of reactor and process instrumentations, control systems, safety protection system, and engineered safety features actuating systems. Six fundamental controls: turbine bypass control, inventory control, reactor outlet temperature control, turbine inlet temperature control, process heat supply rate control and IHX differential pressure control, are combined in the basic plant control to provide controllability for a variety of transients including loss-of-load and electric load following as shown above.

7. Plant Arrangement

The reactor building is a below-grade, steel concrete structure housing 4 units of reactor systems consists of subsystems including reactor module, gas turbine module and heat exchanger modules. The co-axial pipe connecting the modules are aligned at the same vertical level. The reactor module is fixed in horizontal direction while gas turbine and heat exchanger modules are allowed movement due to thermal expansion in the direction.

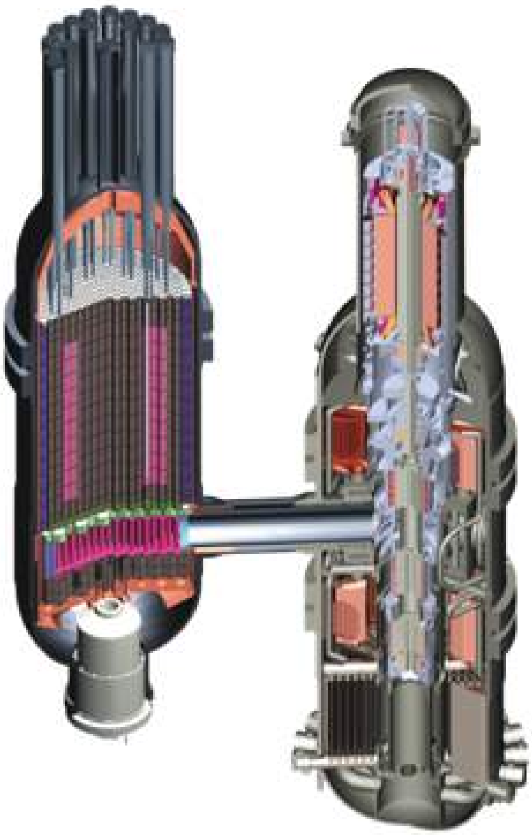
8. Design and Licensing Status

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 in 2030s.

9. Development Milestones

2003	Basic design of GTHTR300 completed
2004	Begin development of key technologies
2005	Cogeneration plant design (GTHTR300C)
2014	IS process continuous H2 production test facility construction
2015	Basic design for HTTR-connected gas turbine and H2 test plant (HTTR-GT/H2)
2020~	HTTR-GT/H2 test plant construction and operation (planned)
2030~	Construction of lead commercial plant (planned)

GT-MHR (OKBM Afrikantov, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	OKBM Afrikantov, Russian Federation
Reactor type	Modular Helium Reactor
Coolant/moderator	Helium
Thermal/electrical capacity, MW(t)/MW(e)	600/288
Primary circulation	Forced circulation
System pressure (MPa)	7.2
Core inlet/exit temperatures (°C)	490/850
Fuel type/assembly array	coated particle fuel, hexagonal prism graphite block of 0.36m
Number of fuel blocks	~1020
Fuel enrichment (%)	LEU or WPu
Fuel burnup (GWd/ton)	100-720 (depends on fuel cycle)
Fuel cycle (months)	25
Main reactivity control mechanism	Control rod insertion
Approach to engineered safety systems	Hybrid (active and passive)
Design life (years)	60
Plant footprint (m^2)	9110
RPV height/diameter (m)	29/8.2
Seismic design	8 points (MSK 64)
Distinguishing features	Inherent safety characteristics; no core melt; high temperature process heat capabilities; small number of safety systems
Design status	Preliminary design completed; key technologies are being demonstrated

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(t)) and NPP power variation as a function of module number. This provides good manoeuvring characteristics of the reactor plant (RP) for regional power sources.

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. Specific Design Features

(a) 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 the figure above. 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.

(b) Reactor Core and Fuel Characteristics

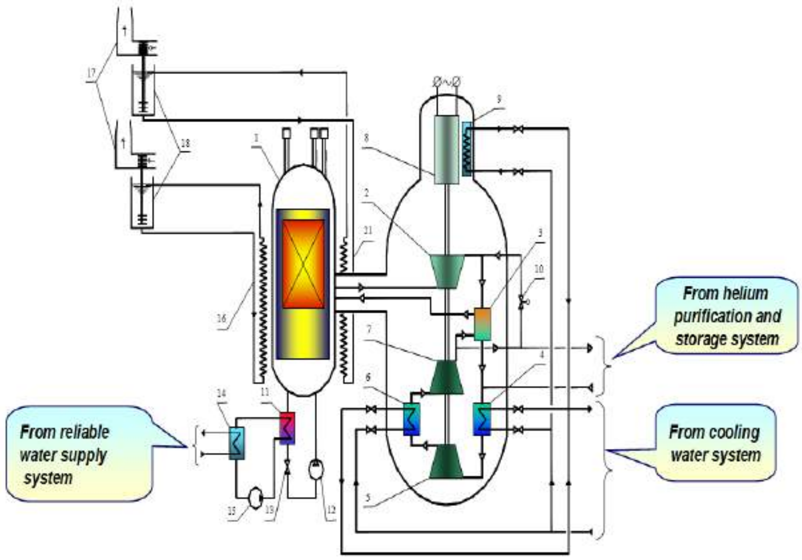
Coated particle fuel is used. The fuel kernel (U or Pu oxide) is coated by a first porous layer of pyro carbon, followed by a dense layer of pyro carbon, a silicon carbide layer and an outer dense layer of pyro carbon. Thousands of coated particles and graphite matrix material are made into a fuel compact 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.

(c) Power Conversion System Flow Diagram

1. Reactor
2. Gas turbine
3. Recuperator
4. Pre-cooler
5. Low pressure compressor
6. Intercooler
7. High pressure compressor
8. Generator
9. Generator cooler
10. Bypass valve

11~15. SCS components

16. Surface cooler of reactor
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. 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.

(d) Fuel Handling System

Fuel handling operations are performed using the protective containers to avoid fuel assembly damage and radioactive product release. Appropriately shielded containers are provided to protect the personnel against radiation impacts during dismantling of the reactor unit components.

(e) Reactivity Control

Two independent reactivity control systems based on different operation principles are used to execute reactor emergency shutdown and maintenance in a sub-critical state. These systems are: 1) Electromechanical reactivity control system based on control rods moving in the reactor core channels and in the inner and outer reflectors; 2) Reserve shutdown system (RSS) based on spherical absorbing elements that fill-in channels in the fuel assembly stack over the whole height of a fuel assembly. Control rods with boron carbide absorbing elements located in the reflector used during normal operation and hot shutdown, and rods located in the core used for scram.

(f) Reactor Pressure Vessel and Internals

The reactor pressure vessel, made of chromium-molybdenum steel, is 29 m in height with an outer diameter (across flanges) of 8.2 m. Prerequisites and conditions excluding brittle fracturing of the reactor vessel include keeping the fast neutron fluence on the reactor vessel and the vessel temperature below the allowable limits. In-vessel structures, namely, prismatic fuel blocks, reflectors, and core support structure are made of graphite, and metallic structures are made of chromium-nickel alloy. Service life of the reactor vessel and internals is 60 years.

4. Safety Features

Safety objectives for the GT-MHR are achieved, first of all, by relying on the *inherent safety features* incorporated in the plant design. The design features, which determine the inherent safety and ensure thermal, neutronic, chemical and structural stability of the reactor unit, 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;
- The temperature and power reactivity coefficients are negative that provides the reactor safety in any design and accident conditions.

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.

(a) Engineered Safety System Approach and Configuration

In addition to the inherent (self-protection) features of the reactor, the GT-MHR plant incorporates safety systems based on the following principles: 1) Simplicity of both system operation algorithm and design; 2) Usage of natural processes for safety system operation under accident conditions; 3) Redundancy, physical separation and independence of system channels; 4) Stability to the internal and external impacts and malfunctions caused by accident conditions; 5) Continuous or periodical diagnostics of system conditions; 6) Conservative approach used in the design, applied to the list of initiating events, to accident scenarios, and for the selection of the definitive parameters and design margins.

(b) Reactor Cooling Philosophy

High heat storage capacity of the reactor core and high acceptable temperatures of the fuel and graphite allow passive shutdown cooling of the reactor in accidents, including LOCA (heat removal from the reactor vessel by radiation, conduction and convection), while maintaining the fuel and core temperatures within the allowable limits. The GT-MHR design provides for no dedicated active safety systems. Active systems of normal operation, such as the power conversion unit and the shutdown cooling system are used for safety purposes. These systems remove heat under abnormal operation conditions, during design basis accidents (DBA) and in beyond design basis accidents (BDBA). Emergency heat removal can also be carried out by the reactor cavity cooling system (RCCS). Heat from the reactor core is removed through the reactor vessel to the RCCS surface cooler, the heat tubes and then to the atmospheric air due to natural processes of heat conduction, radiation and convection. Water and air in the RCCS channels circulate driven by natural convection.

(c) Containment Function

Passive localization or radioactivity is provided by the containment designed for the retention of helium-air fluid during accidents with primary circuit depressurization. The containment is also designed for the external loads, which may apply at seismic impacts, aircraft crash, air shock wave, etc. Activity release from the containment into the environment is determined by the containment leakage level, which is about 1 % of the volume per day at an emergency pressure of 0.5 MPa.

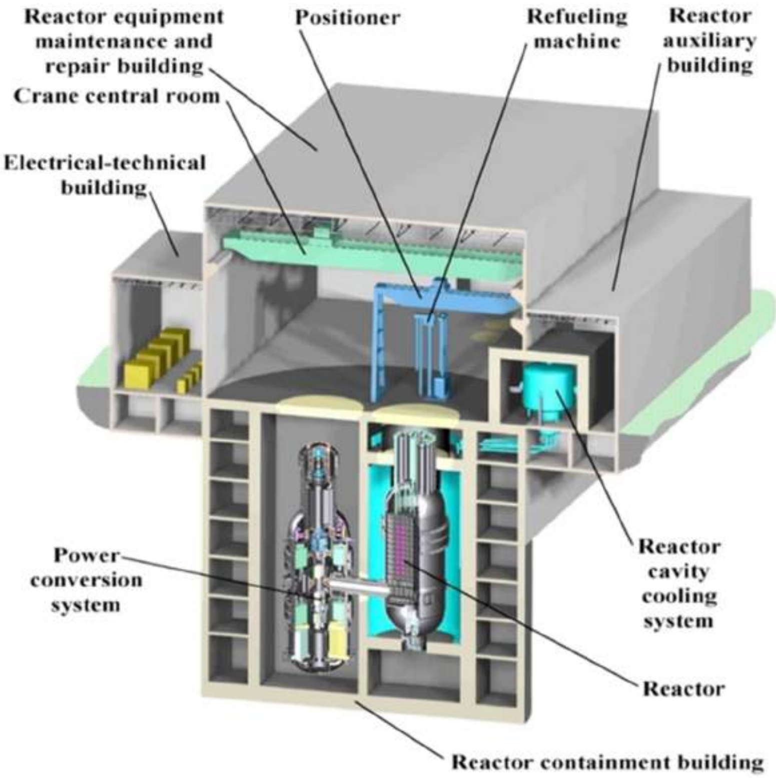
5. Plant Safety and Operational Performances

All safety systems are designed with two channels. Fulfilment of the regulatory requirements on safety, proven by a compliance with both deterministic and probabilistic criteria, is secured by an exclusion of the active elements in a channel or by applying the required redundancy of such active elements inside a channel, as well as via the use of the normal operation systems to prevent design basis accidents.

6. Instrumentation and Control Systems

The GT-MHR NPP control and support safety systems (CSS) are intended to actuate the equipment, mechanisms and valves, localizing and support safety systems in the pre-accidental conditions and in accidents; to monitor their operation; and to generate control commands for the equipment of normal operation systems used in safety provision algorithms. The CSS are based on the principles of redundancy, physical and functional separation, and safe failure. The CSS sets are physically separated so that internal (fire, etc.) or external (aircraft crash, etc.) impacts do not lead to a control system failure to perform the required functions. The CSS provide automated and remote control of the equipment of safety systems from the independent main and standby control rooms. Principal technical features are selected using the concept of a safe failure - blackouts, short-circuits, or phase breaks initiate emergency signals in the channels or safety actions directly.

7. Plant Arrangement



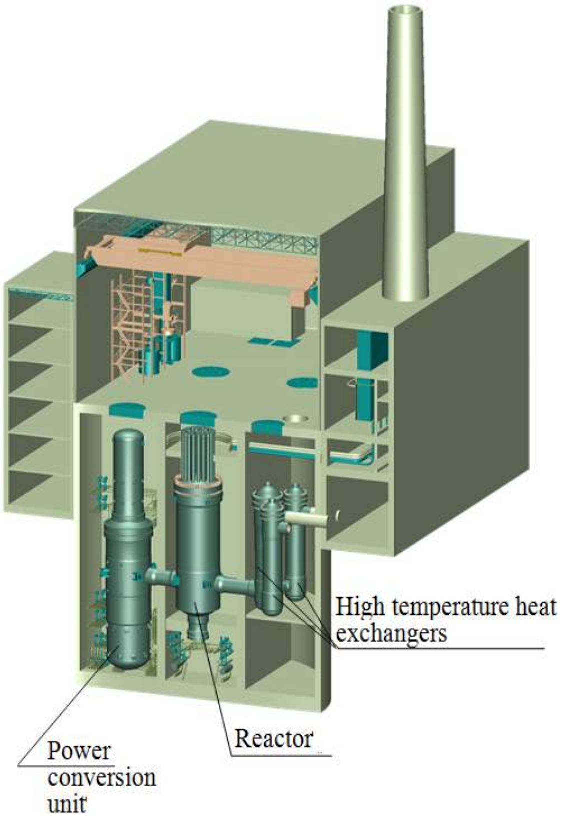
8. Design and Licensing Status

Reactor plant preliminary design and demonstration of key technologies for Pu-fuelled option completed.

9. 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
2014	Completion of demonstration of key technologies for Pu-fuelled core
Since 2014	Use of principal reactor unit design features as a basis for MHR-T design (U-fuelled option)

MHR-T Reactor/Hydrogen Production Complex (OKBM Afrikantov, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	OKBM Afrikantov, Russian Federation
Reactor type	Modular helium high-temperature reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	4x600/4x205.5
Primary circulation	Forced circulation
System pressure (MPa)	7.5
Core inlet/exit temperatures (°C)	578/950
Fuel type/assembly array	Coated particle fuel, hexagonal prism graphite block of 0.36m
Number of fuel blocks	~1020
Fuel enrichment (%)	20
Fuel burnup (GWd/ton)	125
Fuel cycle (months)	30
Main reactivity control mechanism	Control rod insertion
Approach to engineered safety systems	Hybrid (active and passive)
Design life (years)	60
RPV height/diameter (m)	32.8/6.9
Seismic design	8 points (MSK 64)
Distinguishing features	Multi-module HTGR dedicated to hydrogen production / high temperature process heat application.
Design status	Conceptual design

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 coupling 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 (GT-MHR design), 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 high level of safety even in case of total loss of primary coolant.

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. Specific Design Features

(a) Design Philosophy

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, H_2, CO_2, and CH_4) in the course of a thermochemical reaction.

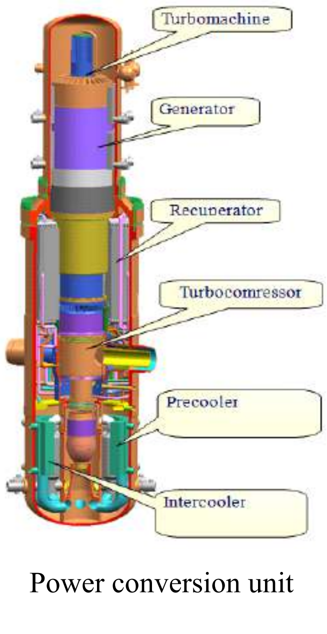
(b) Reactor Core and Fuel Characteristics

Coated particle fuel is used. The fuel kernel (U oxide) is coated by a first porous layer of pyro carbon, followed by a dense layer of pyro carbon, a silicon carbide layer and an outer dense layer of pyro carbon. 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.

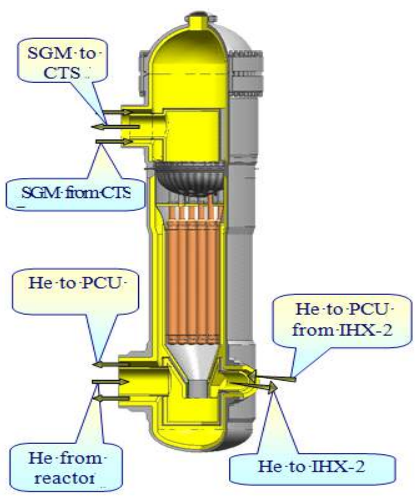
The coated particles will contain almost all fission products with temperatures up to 1600°C. About 1 billion fuel particles of the same type were manufactured and tested in Russia. The standard fuel cycle is to utilize low enriched uranium (LEU) in a once through mode. The MHR-T show good proliferation 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.

(c) 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.



Power conversion unit



High temperature heat exchanger section

The material of the heat exchange surface of the module is a heat-resistant alloy. 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%.

(d) Reactor Coolant System

Working media in circulation circuits are helium of the primary 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.

(e) Fuel Handling System

Fuel handling operations are performed using the protective containers to avoid fuel assembly damage and radioactive product release. Appropriately shielded containers are provided to protect the personnel against radiation impacts during dismantling of the reactor unit components.

(f) Reactivity Control

Two independent reactivity control systems based on different operation principles are used to execute reactor emergency shutdown and maintenance in a sub-critical state. These systems are: 1) Electromechanical reactivity control system based on control rods moving in the reactor core channels and in the inner and outer reflectors; 2) Reserve shutdown system based on spherical absorbing elements that fill-in channels in the fuel assembly stack over the whole height of a fuel assembly. Control rods with boron carbide absorbing elements located in the reflector used during normal operation and hot shutdown, and rods located in the core used for scram.

(g) Reactor Pressure Vessel and Internals

Reactor Pressure Vessel made of chromium-molybdenum steel is 29 m height with outer diameter (across flanges) 8.2 m. Prerequisites and conditions excluding brittle fracturing of the reactor vessel include keeping the fast neutron fluence on the reactor vessel and the vessel temperature below the allowable limits. In-vessel structures, namely, prismatic fuel blocks, reflectors, and core support structure are made of graphite, and metallic structures are made of chromium-nickel alloy. Service life of the reactor vessel and internals is 60 years.

4. Safety Features

The safety features of the MHR-T reactor are the same as for the GT-MHR. Safety objectives for the MHR-T are achieved, first of all, by relying on the inherent safety features incorporated in the plant design. The design features, which determine the inherent safety and ensure thermal, neutronic, chemical and structural stability of the reactor unit, 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;
- The temperature and power reactivity coefficients are negative that provides the reactor safety in any design and accident conditions.

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.

(a) Engineered Safety System Approach and Configuration

Special considerations are devoted to external impacts from the hydrogen production sector. In addition to the inherent (self-protection) features of the reactor, the MHR-T plant incorporates safety systems based on the following principles: 1) Simplicity of both system operation algorithm and design; 2) Usage of natural processes for safety system operation under accident conditions; 3) Redundancy, physical separation and independence of system channels; 4) Stability to the internal and external impacts and malfunctions caused by accident conditions; 5) Continuous or periodical diagnostics of system conditions; 6) Conservative approach used in the design, applied to the list of initiating events, to accident scenarios, and for the selection of the definitive parameters and design margins.

(b) Reactor Cooling Philosophy

High heat storage capacity of the reactor core and high acceptable temperatures of the fuel and graphite allow passive shutdown cooling of the reactor in accidents, including LOCA (heat removal from the reactor vessel by radiation, conduction and convection), while maintaining the fuel and core temperatures within the allowable limits. The MHR-T follows the GT-MHR design principles with no dedicated active safety systems (active systems of normal operation are used for safety purposes) and with emergency heat removal also possible by the reactor cavity cooling system (see GT-MHR for more details).

(c) Containment Function

The approach is the same as for the GT-MHR with passive localization or radioactivity provided by the containment as well as external loads (see GT-MHR for more details).

5. Plant Safety and Operational Performances

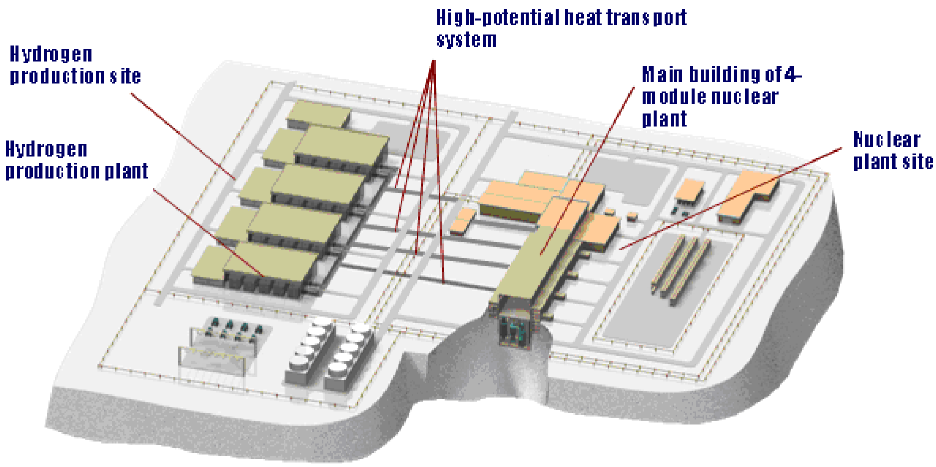
All safety systems are designed with two channels. Fulfilment of the regulatory requirements on safety, proven by a compliance with both deterministic and probabilistic criteria, is secured by an exclusion of the active elements in a channel or by applying the required redundancy of such active elements inside a channel, as well as via the use of the normal operation systems to prevent design basis accidents.

6. Instrumentation and Control Systems

The MHR-T NPP control and support safety systems (CSS) are intended to actuate the equipment, mechanisms and valves, localizing and support safety systems in the pre-accidental conditions and in accidents; to monitor their operation; and to generate control commands for the equipment of normal operation systems used in safety provision algorithms. The CSS are based on the principles of redundancy, physical and functional separation, and safe failure. The CSS provide automated and remote control of the equipment of safety systems from the independent main and standby control rooms. Principal technical features are selected using the concept of a safe failure - blackouts, short-circuits, or phase breaks initiate emergency signals in the channels or safety actions directly.

7. 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 and special external impacts such as 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.



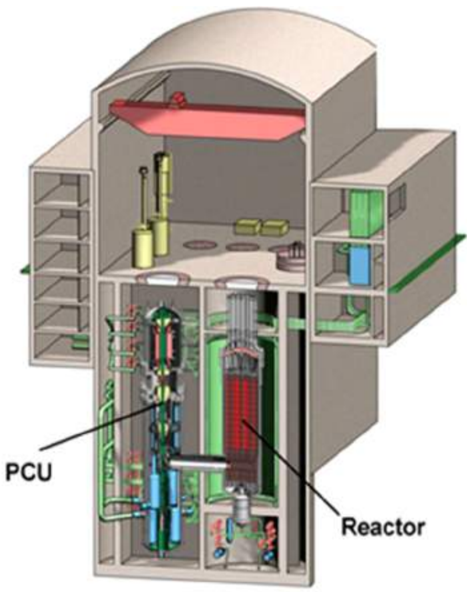
8. Design and Licensing Status

Feasibility study of plant application for large-scale hydrogen production completed.

9. Development Milestones

2001	Pre-conceptual proposal
2005	Conceptual design completed
2007	Elaboration of technical requirements
2017	Feasibility study of plant application for large-scale hydrogen production

MHR-100 (OKBM Afrikantov, Russian Federation)



MHR-100 GT layout

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	OKBM Afrikantov, Russian Federation
Reactor type	Modular helium reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	215/25-87 (depends on configuration)
Primary circulation	Forced circulation
System pressure (MPa)	4-5
Core inlet/exit temperatures (^{o}C)	490-553/795-950
Fuel type/assembly array	Hexagonal prism (0.2 m x 0.65 m height) graphite blocks with coated particle fuel
Number of fuel assemblies	About 1600 blocks with more than 600 fuel compacts in each block
Fuel enrichment (%)	LEU < 20% enriched
Main reactivity control mechanism	Control rod insertion
Approach to engineered safety systems	Hybrid (active and passive)
Seismic design	8 points (MSK 64)
Distinguishing features	A multipurpose reactor for electricity production, heat generation, and hydrogen generation; high-temperature heat supply to oil refinery plant
Design status	Conceptual design

1. Introduction

The designs represented are based on the world-wide experience in the development of experimental HTGR plants. Russia has more than 40 years' experience in the development of HTGR plants of various power (from 100 to 1000 MW) and for various purposes. It 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, conventional power stations, with electric capacity ~300 MW(t), are deployed all over the territory of Russia. These are adapted to regional systems and provide the electric power needs of Russia. This 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.

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

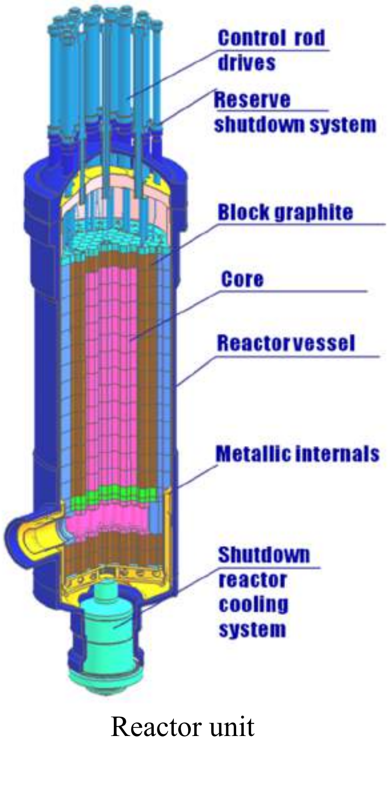
3. Specific Design Features

(a) Design Philosophy

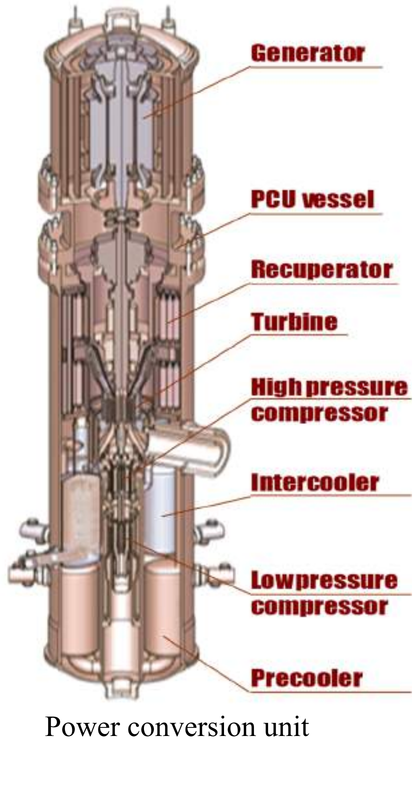
The reactor power and its design are universal for all the different power and process heat options with only the coolant parameters that are different. The reactor unit power level (215 MW(t)) was selected according to: (i) the regional power industry and district heat supply needs; (ii) the manufacture 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.

(b) Reactor Core and Fuel Characteristics

Coated particle fuel is used. The fuel kernel (U oxide) is coated by a first porous layer of pyro carbon, followed by a dense layer of pyro carbon, a silicon carbide layer and an outer dense layer of pyro carbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with hundreds of compacts inserted into fuel channels of each Hexagonal Prism graphite block. The core is a cylindrical arrangement of vertical stacks of fuel blocks (fuel columns). The standard fuel cycle is to utilize low enriched uranium (LEU) in a once through mode.



Reactor unit



Power conversion unit

(c) Power Conversion Unit

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

(d) Fuel Handling System

Fuel handling operations are performed using the protective containers to avoid fuel assembly damage and radioactive product release. Appropriately shielded containers are provided to protect the personnel against radiation impacts.

(e) Reactivity Control

Two independent reactivity control systems based on different operation principles are used to execute reactor emergency shutdown and maintenance in a sub-critical state. These systems are: (i) Electromechanical reactivity control system based on control rods moving in the reactor core channels and in the inner and outer reflectors; (ii) Reserve shutdown system based on spherical absorbing elements that fill-in channels in the fuel assembly stack over the whole height of a fuel assembly. Control rods with boron carbide absorbing elements located in the reflector are used during normal operation and hot shutdown, and rods located in the core are used for scram.

(f) Reactor Pressure Vessel and Internals

Reactor Pressure Vessel made of chromium-molybdenum steel has dimensions similar to those typical for a standard VVER-1000 reactor (Russian LWR). Prerequisites and conditions excluding brittle fracturing of the reactor vessel include keeping the fast neutron fluence on the reactor vessel and the vessel temperature below the allowable limits. In-vessel structures, namely, prismatic fuel blocks, reflectors, and core support structure are made of graphite, and metallic internals are made of chromium-nickel alloy. Service life of the reactor vessel and internals is 60 years.

4. Safety Features

Safety objectives for the MHR-100 reactor are achieved, first of all, by relying on the *inherent safety features* incorporated in the plant design. These design features ensure thermal, neutronic, chemical and structural stability of the reactor unit. 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.

(a) Engineered Safety System Approach and Configuration

In addition to the inherent (self-protection) features of the reactor, the MHR-100 plant incorporates safety systems based on the following principles: (i) Simplicity of both system operation algorithm and design; (ii) Usage of natural processes for safety system operation under accident conditions; (iii) Redundancy, physical separation and independence of system channels; (iv) Stability to the internal and external impacts and malfunctions caused by accident conditions; (v) Continuous or periodical diagnostics of system conditions; (vi) Conservative approach used in the design, applied to the list of initiating events, to accident scenarios, and for the selection of the definitive parameters and design margins.

(b) Decay Heat Removal / Reactor Cooling Philosophy

High heat storage capacity of the reactor core and high acceptable temperatures of the fuel and graphite allow passive shutdown cooling of the reactor in accidents, including LOCA (heat removal from the reactor vessel by radiation, conduction and convection), while maintaining the fuel and core temperatures within the allowable limits. The MHR-100 design provides for no dedicated active safety systems. Active systems of normal operation, such as the power conversion unit and the shutdown cooling system are used for safety purposes. These systems remove heat under abnormal operation conditions, during design basis accidents (DBA) and in beyond design basis accidents (BDBA).

(c) Containment Function

Passive localization or radioactivity is provided by the containment designed for the retention of helium-air fluid during accidents with primary circuit depressurization. The containment is also designed for the external loads, which may apply at seismic impacts, aircraft crash, air shock wave, etc. Activity release from the containment into the environment is determined by the containment leakage level, which is about 1 % of the volume per day at an emergency pressure of 0.5 MPa.

5. Plant Safety and Operational Performances

All safety systems are designed with two channels. Fulfilment of the regulatory requirements on safety, proven by a compliance with both deterministic and probabilistic criteria, is secured by an exclusion of the active elements in a channel or by applying the required redundancy of such active elements inside a channel, as well as via the use of the normal operation systems to prevent design basis accidents.

6. Instrumentation and Control

Control and support safety systems are intended to actuate the equipment, mechanisms and valves, localizing and support safety systems in the pre-accidental conditions and in accidents; to monitor their operation; and to generate control commands for the equipment of normal operation systems used in safety provision algorithms.

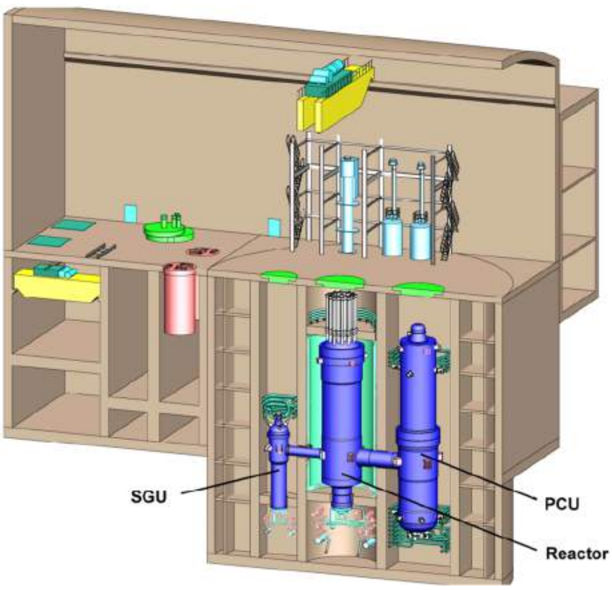
7. Design Variants and Plant Arrangements Based on the Modular MHR-100

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

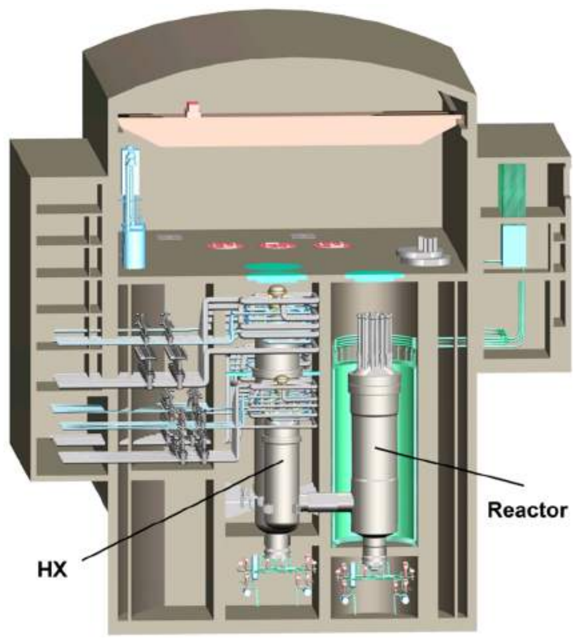
- Modular high-temperature helium-cooled reactors with typical high level of inherent safety;
- Fuel cycle with fuel in the form of multilayer coated fuel particles (based on UO_2), high burnup and possibility to disposal the spent fuel blocks without additional reprocessing;
- High-performance high-temperature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

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.



MHR-100 SE



MHR-100 SMR

MAJOR TECHNICAL PARAMETERS

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 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

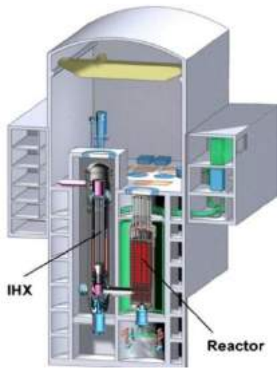
MHR-100 SMR

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.

MHR-100 OR



MAJOR TECHNICAL PARAMETERS	
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

8. Design and Licensing Status

Optimization of reactor core design. Feasibility study of MHR-100-SMR plant application for large-scale hydrogen production, technical and economical evaluation of the plant potential to supply hydrogen to the expected market.

9. Development Milestones

2014	Conceptual design completed
2018	Feasibility study of plant application for large-scale hydrogen production

AHTR-100 (Eskom Holdings SOC Ltd., South Africa)

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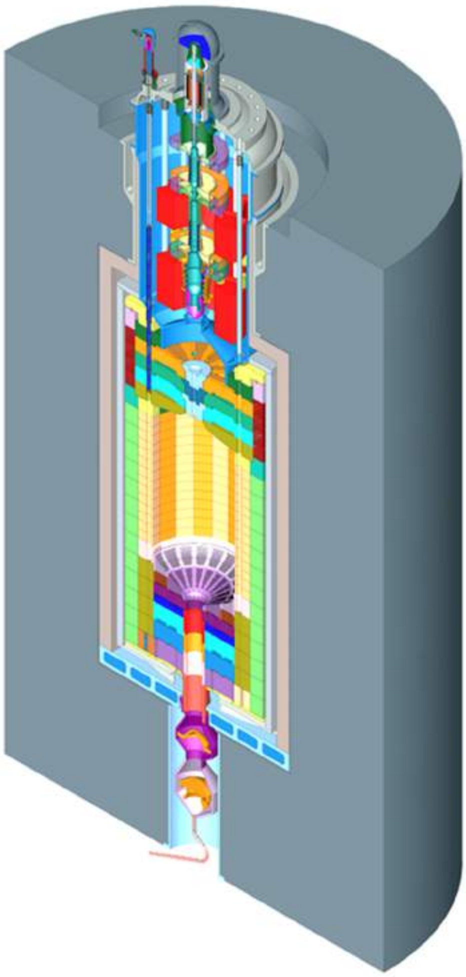
MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Eskom Holdings SOC Ltd., South Africa
Reactor type	Modular high temperature gas cooled reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	100/50
Primary circulation	Forced circulation
System pressure (MPa)	9
Core inlet/exit temperatures (°C)	406/1200
Fuel type/assembly array	Pebble bed with coated particle fuel
Number of fuel assemblies	~110,250 in core
Fuel enrichment (%)	LEU or WPu
Fuel burnup (GWd/ton)	86
Fuel cycle (months)	N/A; online / on-power refuelling
Main reactivity control mechanism	Control rod insertion, negative temperature coefficient
Approach to engineered safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	To be confirmed
RPV height/diameter (m)	11.4/6.05 (outer) 2.6 (inner)
Seismic design	0.4g PGA for main power system design
Distinguishing features	Inherent safety characteristics; no core melt; high efficiency; small number of safety systems
Design status	Concept design completed; R&D activities in progress



Reactor system configuration of AHTR-100

1. Introduction

When the PBMR® was defined in the mid-1990s, it was based on the industrially demonstrated technology of the German Arbeitsgemeinschaft Versuchsreaktor (AVR) proof of concept reactor, and the German Thorium High Temperature Reactor (THTR 300) commercial scale reactor, integrated with a direct Brayton cycle helium turbine power conversion unit, based on existing industrial gas turbines.

The PBMR® approach was to avoid any fundamentally new technologies and to move directly to the 'demonstration' reactor, which was planned to also be a first of class of the commercial machine. While many tests were done to confirm the performance that was achieved earlier, there were few new design elements, except for the integration of the reactor with a helium gas turbine.

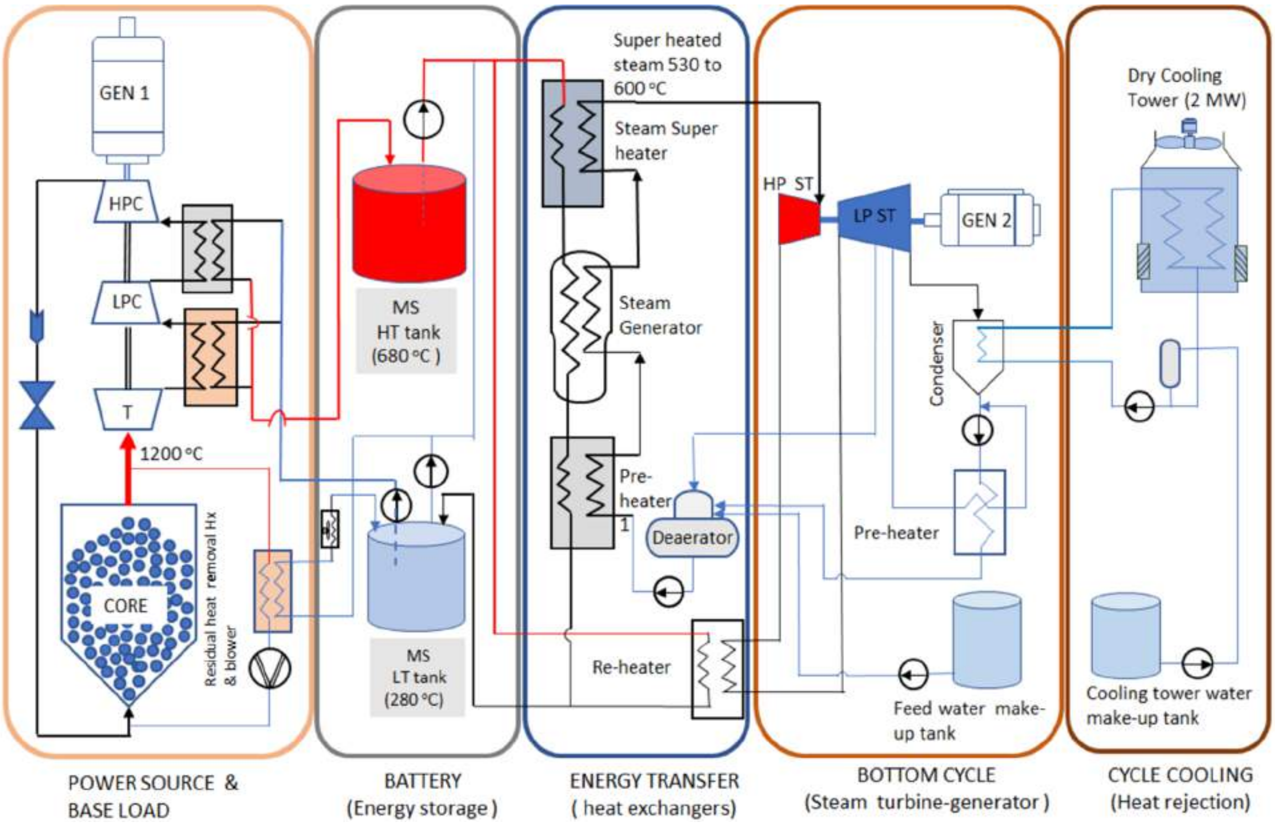
Given these potential advances available with current international technology, but not planned to be applied by other programs, significant performance improvements could be achieved over the performance envisaged for the original PBMR®.

In particular the use of carbon-composite materials to achieve higher operating temperatures (more than 1000°C), and the use of molten salts as heat storage, would result in higher thermal efficiencies and better flexibility. While all these technologies have been demonstrated at different scales in other industries, their detailed application to high temperature reactor (HTR) designs would require an industrial-scale reactor plant to prove their suitability for a commercial reactor. In order to implement these advances in a demonstration plant, the AHTR-100 was conceptualised by Eskom Holdings SOC Ltd. – PBMR SOC Ltd. is wholly owned by Eskom Holdings SOC Ltd.

With an output of temperature of 1200°C the AHTR-100 is classified as a very high temperature reactor (VHTR). Specific demonstrated nuclear technologies, such as the fuel design, will however remain the same as that of the PBMR®.

2. Target Application

The AHTR-100 can produce electricity at high efficiency via a combined direct helium Brayton cycle and Rankine bottoming cycle with an intermediate heat storage for load following or process heat applications or a bottoming steam cycle, as depicted in the figure below.



AHTR-100 schematic with topping, bottoming cycle and energy storage unit.

3. Specific Design Features

(a) Design Philosophy

As in the PBMR®, the AHTR-100 is a high-temperature helium-cooled, graphite moderated pebble bed reactor but with a once-through 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.

(b) Reactor Core and Fuel Characteristics

The core neutronic design results in a small cylindrical core with a diameter of 2.6 m. The effective cylindrical core height is 9.35 m. In steady state (equilibrium core) operation the fuel sphere powers (average 0.91 kW) and operational temperatures (1200°C) fulfil the design criteria. The core contains ~ 110 250 fuel spheres or "pebbles" with a packing fraction of 0.61. The fuelling scheme employed is the continuous on-line once-through 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.

The fuel kernel (UO_2) is coated by a first porous layer of pyro carbon, followed by a dense layer of pyro carbon, a silicon carbide layer and an outer dense layer of pyro carbon. About 13330 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.

(c) Power Conversion System

A Brayton power conversion with direct gas turbine is adopted as topping cycle. 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. Heat exchangers (up to 3) to remove heat to the bottoming cycle is included.

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 primary side of the first high temperature heat exchanger, then the pre-cooler, the low-pressure compressor, intercooler, high pressure compressor and then on to the high-pressure intercooler 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.

This primary cycle will operate in baseload maximum capacity continually and provide 30% of the total plant electricity. This limits reactor operating transients to startup, full load operation, and shut down.

The heat exchangers contain molten salt coolant in the secondary side removing heat from the primary circuit and storing it for use in a load following Rankine cycle.

(d) Reactivity Control

Excess reactivity is limited by once through, continuous refuelling cycle 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 capability are provided by two separate and diverse control rod and small absorber sphere (SAS) systems while the overall negative reactivity temperature coefficient is negative over the total operational range.

(e) Fuel Handling System

Fuel spheres are circulated in the online handling system by means of a combination of gravitational flow and pneumatic conveying processes using helium at system operating pressure, as the transporting gas. The system functions as an online fuel replenishing system. This involves fresh fuel replenishment, fuel unloading, and discharging used fuel to the used fuel vessels.

(f) Reactor Pressure Vessel and Internals

The average core height is 9.35 m and the reflector thickness 0.95 m. The side reflectors are manufactured from nuclear grade graphite blocks that are stacked in columns to make up the geometry of the core. The side reflector columns have borings for the control rods, as well as riser channels for the incoming coolant gas. All the blocks are connected with graphite keys to prevent diversion of the coolant flow. The whole of these ceramic core internals is housed in a stainless-steel Core Barrel that is supported on the bottom of the prestressed concrete reactor pressure vessel.

4. 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 modifications by AHTR-100 in the same manner as with the PBMR®. Its 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 AHTR, like 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 through a heat pipe system on a decreasing curve without any core failure or release of radioactivity to the environment.

(a) Engineered Safety System Approach and Configuration

The AHTR builds on the PBMR® nuclear reactor system that is designed to derive maximum safety benefits from its inherent passive safety characteristics which are; designed to rule out core melt, all ceramics fuel, coated particle provides excellent containment for the fission product activity, large negative temperature feedback, the helium coolant is chemically inert (single phase), large thermal capacity lead to slow thermal transients, no common mode failure in the core (a single fuel failure does not lead to additional failures), ingress of water into core eliminated by design and air ingress limited.

(b) Decay Heat Removal/ reactor Cooling Philosophy

The reactor cavity cooling system (RCCS) is a means to remove residual heat passively for a defined time, and indefinitely with the use of a passive heat pipe system. The use of a pre-stressed concrete pressure vessel in effect insulates the core from the atmosphere and as a result, the system requires passive heat removal by the heat pipe system. In the event of the loss of active core cooling by the main circulation system, the heat pipe system is activated automatically through the temperature rise and are able to limit the increase in fuel temperature in the most affected region of the core to below the allowable fuel temperature limit.

(c) Containment Function

As with the PBMR®, the most important barriers to fission product release are the coatings of the fuel particles. A second barrier is provided by the helium pressure boundary. A third barrier is the confinement building. 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.

5. Plant Safety and Operational Performances

As in the PBMR®, the AHTR-100 safety does not rely on engineered systems that may fail but on the inherent design and the laws of physics. The risk metrics core damage frequency and large early release frequency are not applicable, but the same concepts are reflected in the immediate and delayed release category definitions. The design of the AHTR based on the PBMR® represents a significant advancement in plant safety with an estimated delayed release category frequency of 1.0E-5 per reactor year while maintaining an expected capacity factor of 95 percent.

The AHTR concept is directed to be a simplistic design, exhibit inherently safety characteristics and high operational efficiency. The operating modes, states and transitions are under definition but it is specified that the unit is able to shutdown with no human intervention requirements, in the event of LOFC.

6. Instrumentation and Control Systems

As in the PBMR®, the AHTR system consists of an inherently stable and slow acting heat source (Reactor Unit), due to its large thermal capacity, which makes it nearly self-regulating, coupled to a fast-acting power conversion machine. The Power Conversion Unit therefore require active control to remain stable under all anticipated operating scenarios. The reactor power is adjusted by changes in the helium mass flow rate in the power conversion unit. The helium inventory system is used to change the pressure (mass adjusted through changes in density) and power control is subsequently performed in combination with a bypass valves.

7. Design and Licensing Status

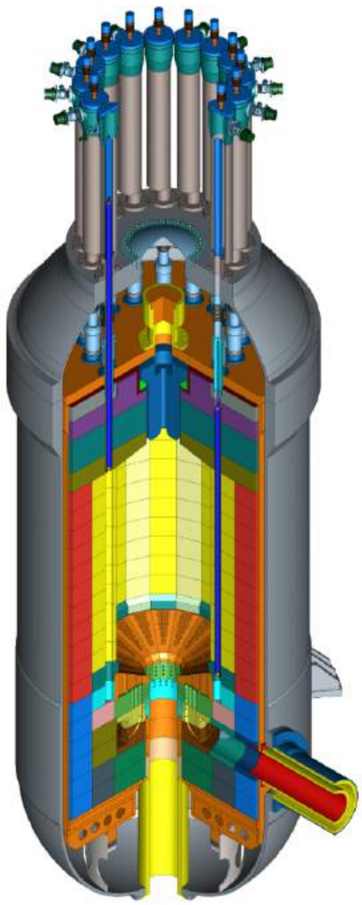
The design basis for the proof of concept machine has been completed for a direct cycle machine. The intent is to test and proof several aspects of the technology prior to implementation in the commercial power plant. The layout for the overall plant is being developed with operating modes, states and transitions progressively defined.

The licensing framework for the proof of concept is also complete and the nuclear regulator is appraised on the effort of the developments in the project. Reactor Plant Conceptual Phase has been completed with key R&D work continuing in the field of qualifying materials and design and construction of demonstration components.

8. Development Milestones

2010	PBMR® Project in care and maintenance since 2010.
2016	AHTR-100 R&D activities commence.
2017	AHTR-100 Version 1 concept completed.
2018	R&D activities continue.

HTMR-100 (Steenkampskraal Thorium Limited, South Africa)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	STL, South Africa
Reactor type	HTGR
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	100/35 single module plant
Primary circulation	Forced circulation
System pressure (MPa)	4
Core inlet/exit temperatures (°C)	250/750
Fuel type/assembly array	TRISO particles in pebbles: LEU, Th/LEU, Th/HEU or Th/Pu
Number of fuel units	~150 000 pebbles; around 125 to 150 pebbles/day throughput
Fuel enrichment (%)	Various, see description below
Fuel burnup (GWd/ton)	80-90
Fuel cycle (months)	Online fuel loading
Main reactivity control mechanism	Absorber rods in the reflector
Approach to engineered safety systems	Passive
Design life (years)	40 full power years
Plant footprint (m^2)	5000 buildings only
RPV height/diameter (m)	15.3/5.9 flange
Seismic design	0.3 g for generic site; (0.5 g under consideration)
Distinguishing features	No core meltdown, modular design, reduced construction time, no active engineered safety 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.
Design status	Advanced concept phase

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 35 MW 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.

2. Target Application

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. Specific Design Features

(a) 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) fuelling 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.

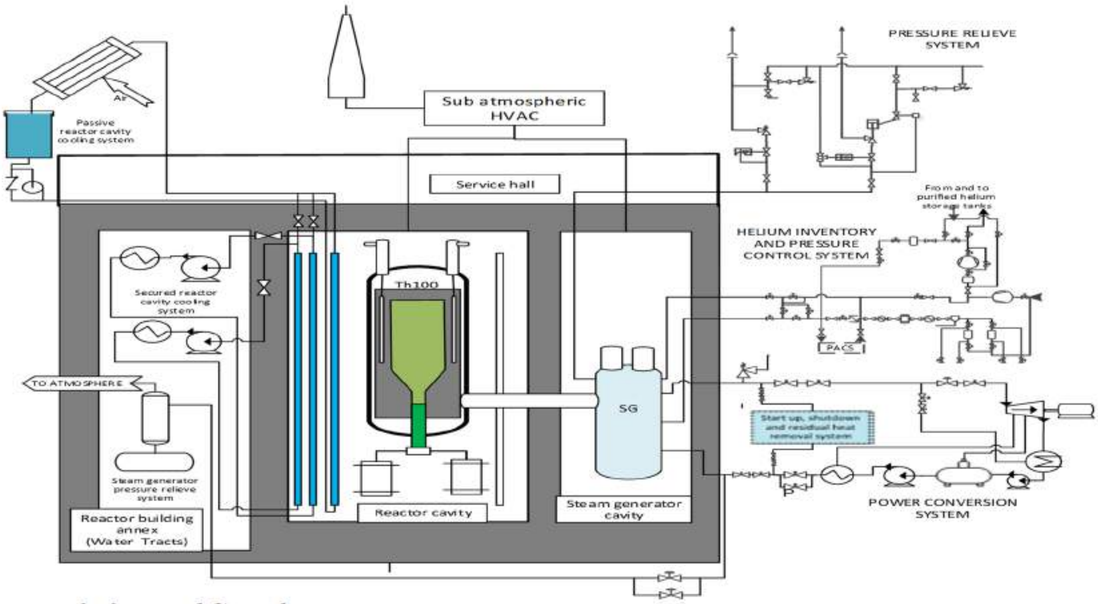
The HTMR-100 plant design caters for different site and client requirements. It allows flexibility in protection against external events and flexibility in multi module configuration and power capacity.

(b) 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 flow through the core is from top to bottom where the heated gas is collected in a hot plenum. From the plenum the hot gas flows through a connecting pipe to the steam generator. 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.



(c) 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

- LEU: 10% enrichment, (7g HM/sphere)
- Th/LEU: 50% LEU by mass 20% enrichment, 50% Th (10-12g HM/sphere)
- Th/HEU, 10% HEU by mass 93% enrichment, 90% Th by mass (10-12g HM/sphere)
- 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.

(d) Fuel Handling System

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.

(e) 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 B4C 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.

(f) 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 and vessel head which is bolted to 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.

4. Safety Features

(a) Engineering 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. 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.

(b) 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.

(c) 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) fails.

5. Plant Safety and Operational Performance

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.

6. Instrumentation and Control Systems

The Automation System (ATS) comprises that group of safety and non-safety C&I systems that provide automated protection, control, monitoring and human-system interfaces. The three specific systems in the HTMR-100 system structure define control and instrumentation are plant control, data and instrumentation system, equipment/investment protection system and protection system.

7. Plant Arrangement

(a) 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.

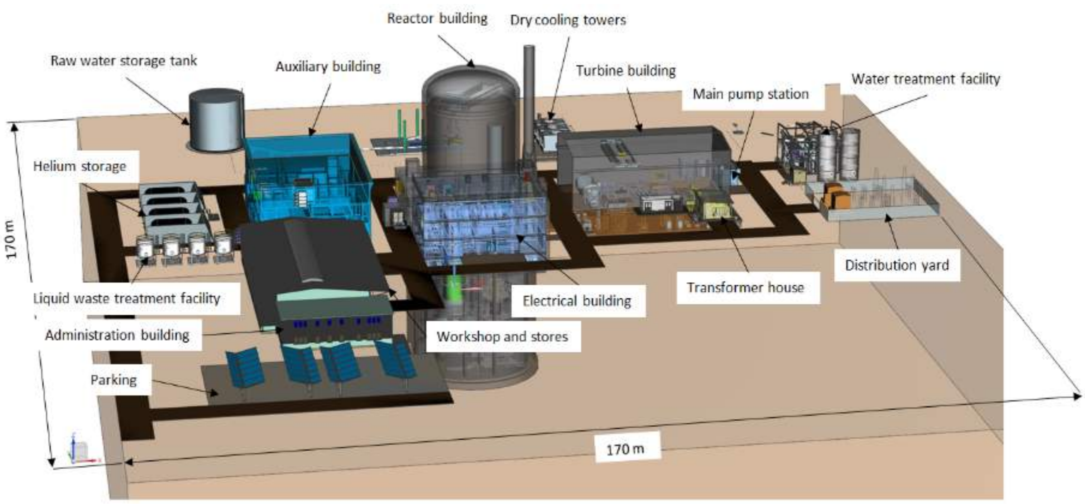
(b) Balance of Plant Building

i. 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 11 kVAC and 400 VAC diesel generator sets and steam safety valves.

ii. 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.



8. Design and Licensing Status

Conceptual design is in an advanced stage. The core neutronic, thermo-hydraulic and heat transfer analyses are being done to optimize the performance and verify the safety analysis. Nuclear Regulator engagement is planned for 2019 with the aim of commencing the pre-assessment for licensing in order to reach design certification status at the end of the Concept Phase.

9. Development Milestones

2012	Project started
2019	Preparation for Pre-license application
2019	Conceptual design completed

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

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Reactor system configuration of PBMR®-400

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Pebble Bed Modular Reactor SOC Ltd (PBMR®), South Africa
Reactor type	Modular high temperature gas cooled reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	400/165
Primary circulation	Forced circulation
System pressure (MPa)	9
Core inlet/exit temperatures (ºC)	500/900
Fuel type/assembly array	Pebble bed with coated particle fuel
Number of fuel assemblies	~452,000 in core
Fuel enrichment (%)	9.6% LEU or WPu
Fuel burnup (GWd/ton)	92 (depends on fuel cycle)
Fuel cycle (months)	N/A; online / on-power refuelling
Main reactivity control mechanism	Control rod insertion, negative temperature coefficient
Approach to engineered safety systems	Active
Design life (years)	40
Plant footprint (m^2)	4200 (main structures only)
RPV height/diameter (m)	30/6.2 (inner)
Seismic design	0.4g PGA for main power system design
Distinguishing features	Inherent safety characteristics; no core melt; high efficiency; small number of safety systems
Design status	Preliminary design completed; test facilities demonstration; project stopped in 2010. in care and maintenance.

1. Introduction

The pebble bed modular reactor (PBMR®) is 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 hold 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(t) adopted from the HTR-Module design to reach 400 MW(t) with an annular core.

Due to the world financial crisis in 2008 and short-term funding constraints a rethink of the product priorities led 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.

2. Target Application

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

3. Specific Design Features

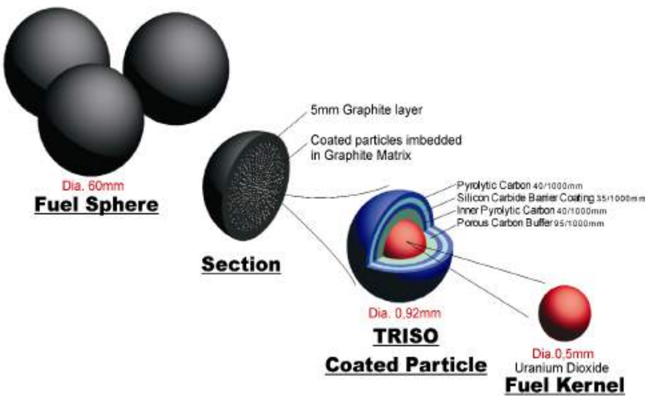
(a) Design Philosophy

The PBMR®-400 is a high-temperature helium-cooled, graphite moderated pebble bed reactor with a multi-pass fuelling scheme. The design objectives and features mean 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 objective there shall be no need for engineered or moving mechanical components to ensure this objective is met while the exposure to plant personnel will be significantly lower than today's best international practice.

(b) Reactor Core and Fuel Characteristics

The core neutronic design is 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 power (maximum 2.7 kW per sphere) and operational temperatures (<1100^0C) fulfil the design criteria set. The core contains ~ 452 000 fuel spheres or "pebbles" with a packing fraction of 0.61. The fuelling scheme employed is a continuous on-line multi-pass system. 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 spheres are circulated six times through the reactor. This reduces power peaking and maximum fuel temperatures in normal operation and loss of coolant conditions.

The coated particle pebble fuel used is shown below. The fuel kernel (UO_2) is coated first by a porous layer of pyro carbon, followed by a dense layer of pyro carbon, a silicon carbide layer and an outer dense layer of pyro carbon. About 15,000 of these 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.



(c) Power Conversion System

The Brayton cycle power conversion cycle 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 the power turbine. The design incorporates a single shaft for the turbine, the compressors and the power generator. The flow of the Helium is depicted in the figure below. 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.

(d) Reactivity Control

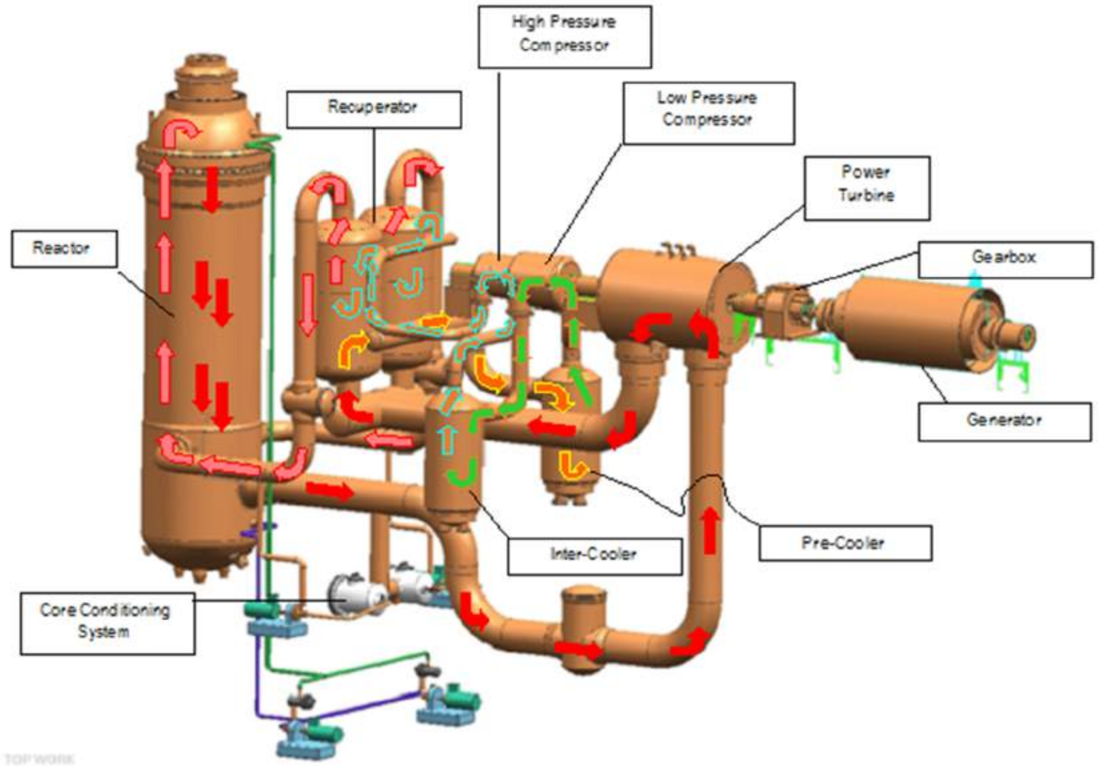
Excess reactivity is limited by continuous refuelling while adequate passive heat removal ensures an inherent safe design with no event with significant fission product release being possible. Adequate reactivity control and long-term cold shutdown capability is provided by two separate and diverse systems while the overall reactivity temperature coefficient is negative over the total operational range. The reactivity control system facilitates load following between 40% and 100%.

(e) Fuel Handling System

Fuel spheres are circulated in the online handling system by means of a combination of gravitational flow and pneumatic conveying processes using helium at system operating pressure, as the transporting gas. The system functions as an online fuel replenishing system. This involves fuel unloading, discharging spent and damaged/worn fuel to used fuel vessels, reloading fresh fuel and fuel that can be returned to the reactor.

(f) Reactor Pressure Vessel and Internals

The average core height is 11 m and the annulus thickness is fixed at 0.85 m. The centre reflector diameter is 2 m and contains eight borings for the Reserve Shutdown System, consisting of borated graphite spheres of 10 mm diameter. The centre and side reflectors are manufactured from nuclear grade graphite blocks that are stacked in columns to make up the geometry of the core. The side reflector columns have borings for the control rods, as well as riser channels for the incoming coolant gas. All the blocks are connected with graphite keys to prevent diversion of the coolant flow. The whole of these ceramic core internals is housed in a stainless-steel Core Barrel that is supported on the bottom of the Reactor Pressure Vessel.



PBMR®-400 power conversion unit components and layout, and He flow.

4. 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 modifications by PBMR®. Its 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 significant radioactivity to the environment.

(a) Engineered Safety System Approach and Configuration

The PBMR® nuclear reactor system is designed to derive maximum safety benefits from its inherent passive safety characteristics which are; designed to rule out core melt, all ceramics fuel, coated particle provides excellent containment for the fission product activity, large negative temperature feedback, Helium coolant is chemically inert (single phase), large thermal capacity lead to slow thermal transients, no common mode failure in the core (a single fuel failure does not lead to additional failures), ingress of water into core eliminated by design and air ingress limited.

(b) Decay Heat Removal/ Reactor Cooling Philosophy

The reactor cavity cooling system provides 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 reactor pressure vessel 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.

(c) Containment Function

The most important barriers to fission product release are the coatings of the fuel particles. A second barrier is provided by the helium pressure boundary. A third barrier is the confinement building. 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.

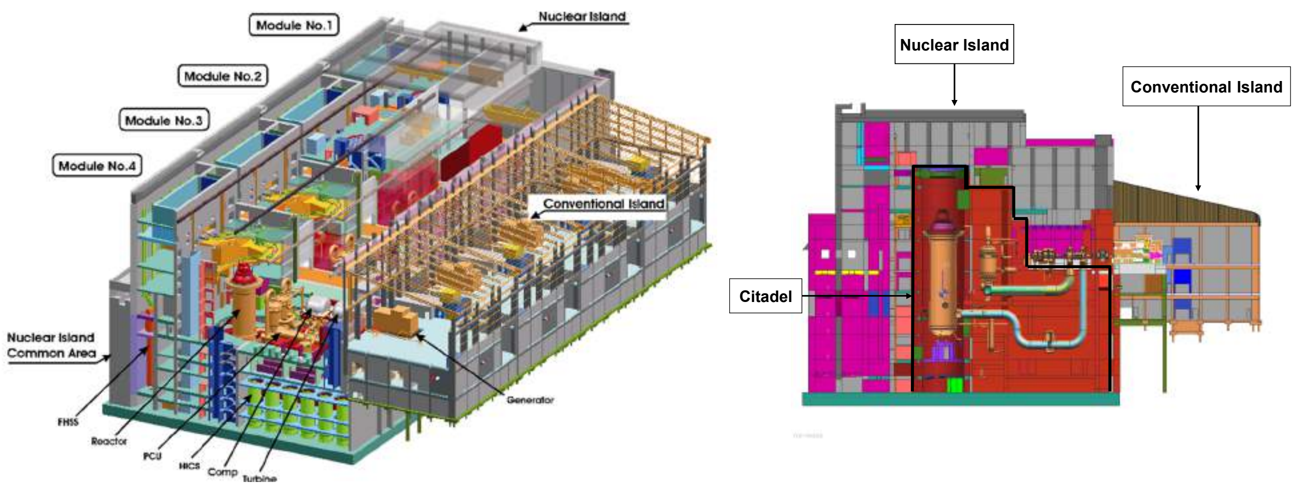
5. Plant Safety and Operational Performances

The PBMR®-400 safety does not rely on engineered systems that may fail but on the inherent design and the laws of physics. The risk metrics core damage frequency and large early release frequency are not applicable, but the same concepts are reflected in the immediate and delayed release category definitions. The design of the PBMR® represents a significant advancement in plant safety with an estimated delayed release category frequency of 1.0E^{-5} per reactor year while maintaining an expected capacity factor of 95 percent.

6. Instrumentation and Control Systems

The PBMR® system consists of an inherently stable and slow acting heat source (reactor unit), due to its large thermal capacity, which makes it nearly self-regulating, coupled to a fast-acting power conversion machine. The power conversion unit therefore require active control to remain stable under all anticipated operating scenarios. The reactor power is adjusted by changes in the helium mass flow rate in the power conversion unit. The helium inventory system is used to change the pressure (mass adjusted through changes in density) and power control is subsequently performed in combination with a bypass valves.

7. Plant Arrangement



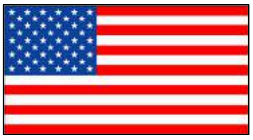
PBMR®-400 building layout

8. Design and Licensing Status

The reactor plant preliminary design was completed and demonstration of key technologies were underway when the project was terminated in 2010.

9. 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 sent for irradiation testing at INL.
2009	Alternative process heat markets and designs explored.
2010	Project closure.
2018	Project in care and maintenance.



SC-HTGR (FRAMATOME INC., USA)

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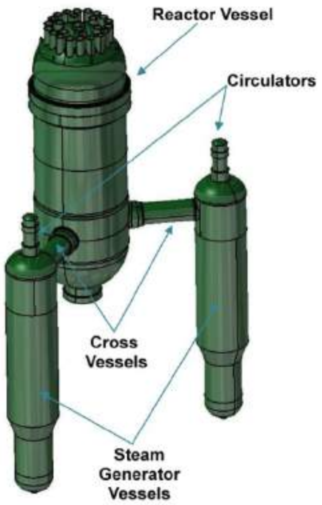
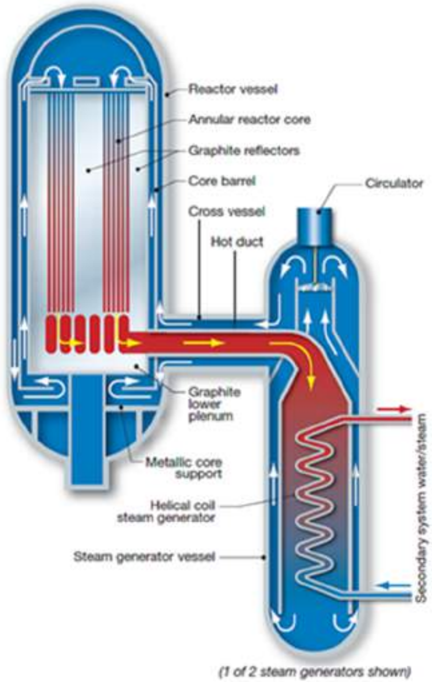




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Framatome Inc., USA
Reactor type	Prismatic block HTGR
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	625/272
Primary circulation	Forced circulation
System pressure (MPa)	6
Core inlet/exit temperatures (^{o}C)	750/325
Fuel type/assembly array	UCO TRISO particle fuel in hexagonal graphite blocks
Number of fuel assemblies	Annular core, 102 columns, 10 blocks / column
Fuel enrichment (%)	<20
Fuel burnup (GWd/ton)	< 170
Fuel cycle (months)	½ of the core replaced every 18 months; 21 days outage
Main reactivity control mechanism	Control rods (gravity insertion) Independent reserve shutdown system (gravity insertion) large negative temperature coefficient
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (km^2)	0.2 (4 module plant)
RPV height/diameter (m)	21 / 8.2
Seismic design	0.35g
Distinguishing features	Coated particle fuel; passive decay heat removal; passive safety; high temperature process steam; vented reactor building; zero EPZ (emergency planning zone); underground construction
Design status	Concept design

1. Introduction

The Framatome SC-HTGR is a modular, graphite-moderated, helium-cooled, high temperature reactor with a nominal thermal power of 625 MW(t) and a nominal electric power capability of 272 MW(e). 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 Framatome'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 General Atomics MHTGR and the HTR-MODUL.

2. Target Application

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. Specific Design Features

(a) Design Philosophy

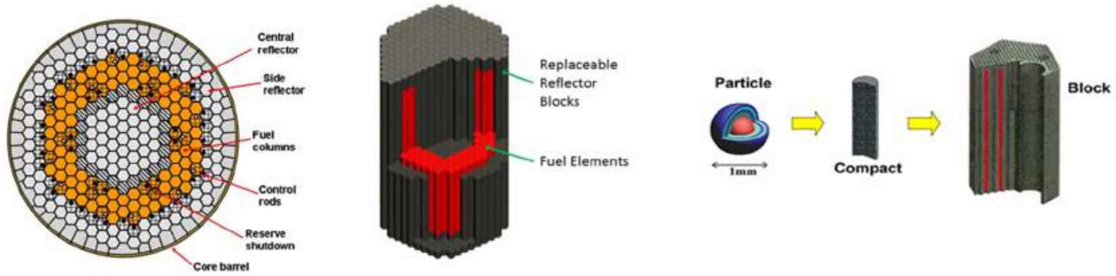
The SC-HTGR is designed around proven helium-cooled, graphite moderated reactor technology, and passive decay heat removal, the heart of which is the TRISO coated fuel particles.

(b) Reactor and Power Conversion System

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 SA-508/533, a standard PWR vessel material, for the primary vessels without requiring separate cooling or special thermal protection. For the reference plant steam cycle concept, the reactor power level is 625 MW(t).

(c) 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. 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.



(d) Reactor Core Layout

In the reference plant the fuel blocks are configured into a 102 column annular core surrounded by graphite reflector elements. 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. The reference plant can be scaled from 625 MW(t) to 50 MW(t) using the same fuel blocks in scaled arrangements.

(e) 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.

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/GW(e)-yr that equates to a natural uranium feedstock utilization of about 224 MT/GW(e)-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.

(f) 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.

(g) Fuel Handling System

Refueling is performed using robotic systems with the primary coolant boundary intact. Following shutdown, the primary system temperature is reduced and then the helium inventory is reduced to slightly sub-atmospheric. Refueling access is then gained through the control rod drive penetrations at the top of the reactor vessel. The robotic refuelling equipment is computer controlled using predetermined block movement sequences.

(h) Reactor Pressure Vessel and Internals

The reactor vessel is part of the vessel system which is the primary pressure-retaining components and also includes the cross vessels and steam generator vessels. The reactor core, reflector elements, core support structure, and core restraint devices are installed in the reactor vessel. The reactor core components, together with elements of the reactor internal components, constitute a graphite assembly that is supported on a graphite core support pillars and restrained by a metallic core support assembly. The reactor internal components consist of the upper core restraint elements, permanent graphite side reflector elements, graphite core support pillars, metallic core support assembly, and the upper plenum shroud.

4. Safety Features

(a) Description of Safety Concept

The primary safety objective of the SC-HTGR design is to limit the dose from accidental releases so that the U.S. EPA protective action guides are met at an exclusion area boundary (EPZ) 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 (RCCS) panels, which provide natural circulation cooling during both normal operation and accidents, so there is no need for the system to actuate, 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 unintended access.

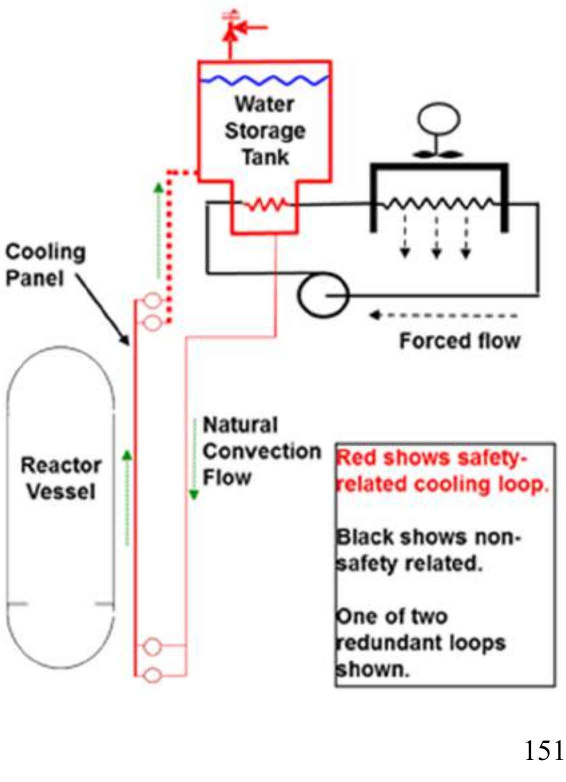
(b) Engineered Safety System Configuration and Approach

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 the SC-HTGR, throughout the modular HTGR licensing history.

(c) 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 RCCS by thermal radiation and natural convection. This heat removal path remains effective even if all primary coolant has been lost.



The RCCS, shown above, 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.

(d) 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 is vented to the atmosphere during a primary system depressurization accident. 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.

5. Plant Operational Performance

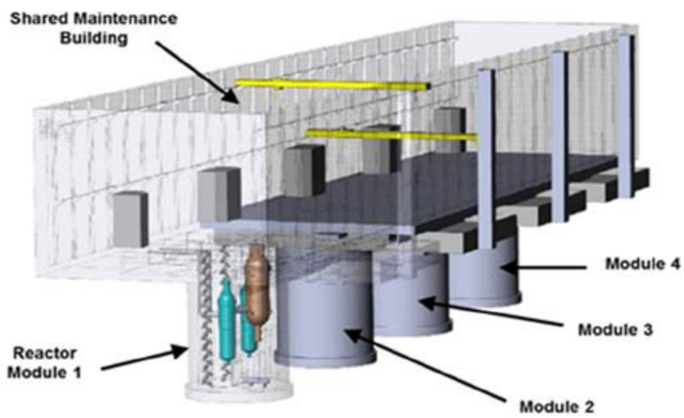
The nominal electricity generation performance of the SC-HTGR system has been evaluate, taking into account preliminary efficiency estimates for the helium circulators, feedwater pumps, turbine, generator and other plant electrical loads. The net electrical output from each 625 MW(t) reactor module is 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%.

6. Instrumentation and Control Systems

The SC-HTGR instrumentation and controls (I&C) include the instruments used for plant protection, monitoring, and control. The plant design goal is to utilize commercially proven I&C hardware and software, which will be commercially available units and will have already demonstrated reliability.

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 below. 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.

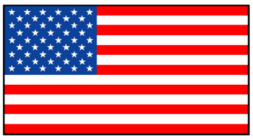


8. Design and Licensing Status

A concept design and preparatory work for pre-licensing application has been completed.

9. 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)
2017	Completed pre-conceptual design plant technical requirements document
	Participated in HTGR reactor building response to depressurization events project
	RCCS experiments/testing at Argonne National Laboratory
2018	Continue support of NRC advanced reactor licensing modernization



Xe-100 (X Energy, LLC - USA)

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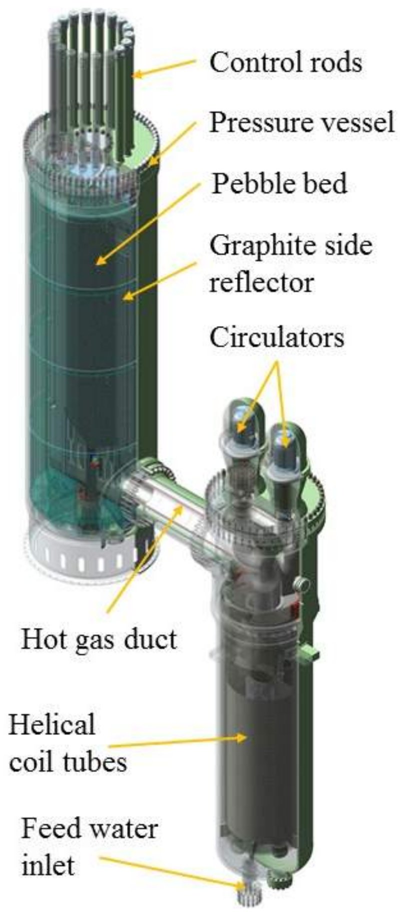


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	X Energy, LLC - USA
Reactor type	Modular HTGR
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	200/75
Primary circulation	Forced helium circulation
System pressure (MPa)	16.5
Core inlet/exit temperatures (^{o}C)	260/750
Fuel type/assembly array	UCO TRISO / pebbles
Number of fuel units	220,000 pebbles per reactor
Fuel enrichment (%)	15.5
Fuel burnup (GWd/ton)	160
Fuel cycle (months)	Online fuel loading
Main reactivity control mechanism	Thermal feedback & control rods
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	216 m X 246 m (4 reactor modules with 4 turbines)
RPV height/diameter (m)	20 m / 4.88 m
Seismic design	0.3 g
Distinguishing features	Online refuelling, core cannot melt and fuel damage minimized by design, independent fission product barriers, potential for advanced fuel cycles
Design status	Conceptual design development

1. Introduction

The Xe-100 is a pebble bed high-temperature gas-cooled reactor with continuous thermal rating of 200 MW. It features a continuous refuelling system with low enriched fuel spheres or pebbles of approximately 15.5 wt% entering the top of the reactor and passing through the core six (6) times to achieve a final average burnup of 160 GWd/tHM.

2. Target Application

Process heat applications, desalination, electricity and co-generation.

3. Specific Design Features

(a) Design Philosophy

A major aim of the Xe-100 design is to improve the economics through system simplification, component modularization, reduction of construction time and high plant availability.

(b) Reactor Core

The Xe-100 core comprises of approximately 220,000 graphite fuel elements (pebbles) each containing approximately 18,000 UCO TRISO coated particles. The core is graphite moderated with online refueling capability. The advantage of online refueling is that the core excess reactivity is maintained at below 2% which means that no burnable poisons are needed to ensure that the reactor reactivity remains within safe shutdown limits at all times. This also improves the neutron economy of the core and helps the Xe-100 to achieve an average burnup of 160 GWd/tHM. At full power approximately 175 fresh pebbles are added daily and a similar number are also removed as spent fuel.

The core geometry (i.e. aspect ratio), power density, heavy metal loading and enrichment level have been optimized to ensure that decay heat can be removed during even the most severe accident scenario such as a total loss of power along with the loss of the helium heat transfer fluid. During such an event, known as a depressurized loss of forced cooling (DLOFC), the decay heat is removed passively through making use of the thermal characteristics of the core and graphite core support structures.

(c) Fuel Characteristics

TRistructural ISOtropic (TRISO) particles are embedded in a graphite matrix pebble to form the fuel element. Particles contain coated uranium oxide and carbide (UCO) kernels enriched at 15.5 wt% and are slightly smaller in diameter (425 µm) than the usual UO_2 (500 µm) fuel kernels used in Germany and China. The optimized moderation ratio (NC/NA) yields a heavy metal loading of around 7 g/pebble. This enables the Xe-100, under worst case water ingress scenarios, to be shut down with its reactivity control and shutdown system (RCSS). Moreover, the graphite shell does not melt but sublimes (changes into vapor) at >~ 3,920°C (4,200 K) and fuel temperature never exceeds 1,100°C during normal operation. Therefore, X-energy does not have to bear the same magnitude of costs related to the pressure vessel, containment building, or safety systems as those of a traditional nuclear plant.

(d) Fuel Handling System

The fuel handling system (FHS) moves fresh fuel pebbles, upon arrival at the plant, to the reactor where they remain until the fuel has been fully utilized. The pebbles are then removed from the reactor and transferred to the spent fuel storage system. The FHS is comprised of four main subsystems/components: New fuel loading system; Fuel unloading & recirculation system; Fuel burnup-measurement system; Spent fuel handling & storage system.

The FHS is a closed system which allows for 100% accountability of the fuel as it enters and exits the reactor. Each time the fuel passes through the reactor the burnup is measured to determine the amount of useful fuel available. If the fuel is not fully spent, it is recycled through the reactor and remains in the fuel handling system until spent and is then deposited into a spent fuel cask. These casks are stored onsite for the life of the plant.

(e) Reactivity Control

First and foremost, the reactor relies on a strong negative temperature coefficient to ensure nuclear stability at all times. For operational reactivity control the reactor has a RCSS comprised of a bank of nine control rods with B_4C as the main control poison. A second bank of nine rods remains in the fully withdrawn position acting as reserve shutdown system primarily used for maintenance shutdown. The negative temperature coefficient alone will shut the reactor down to a safe shutdown condition without the need for active reactivity control systems. The control rod and shutdown rods can however individually shut down the reactor in a controlled shutdown operation. To achieve indefinite shutdown at temperatures of about 100°C for maintenance, both banks are inserted. Due to continuous fuelling, a minimum excess reactivity margin can be maintained. This margin is functionally selected to allow for start-up when performing load-follow operation (100%-40%-100%) and is sufficient to cover the effect of Xeon decay.

(f) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) and internal structures are designed for a 60-year life.

4. Safety Features

The intrinsic safety characteristic of the plant is guaranteed by a relatively low power density of 4.8 MW/m^3, high thermal inertia of the graphitic internals and a strong negative temperature coefficient of reactivity over the total operational regime of the reactor. Also, the use of qualified UCO TRISO coated particle fuel provides excellent retention of fission products at the source. The pressure boundary provides a further independent physical barrier to retain the small amount of fission products that may end up circulating in the helium and in graphite dust particles. The reactor building venting route also minimizes the release of fission products by venting through filtered release vents.

(a) Engineered Safety System Approach and Configuration

The primary engineered safety systems are designed to be passive. Unintended plant transients are avoided due to the small excess reactivity resulting from continuous fuelling. The RCSS insertion depth during normal operation binds around 1.4 Niles (1 nile = 1000pcm), allowing for load-follow operation within the range of 100% - 40% -100%. Any spurious signal that would cause full withdrawal of the RCSS would therefore only translate to a higher temperature and will not cause fuel damage.

(b) Decay Heat Removal/ Reactor Cooling Philosophy

Passive decay heat removal is possible, while the fuel temperature remains below admissible values. The radionuclides remain inside the fuel even throughout extreme upset events. If the active heat removal system is not available, then the core heat is removed passively through: Conduction between the pebbles and side reflector; Convection and thermal radiation to the core barrel, RPV; and, Reactor Cavity Cooling System (RCCS). Loss of the RCCS does not result in a safety concern as decay heat can be safely dissipated into the building structures and finally to the environment.

(c) Containment Function

Xe-100 "functional containment" is based on TRISO coated particles serving as the primary barrier to radionuclide release. The fuel element matrix contributes to additional resistance and adsorber surface in diffusing radionuclides. The helium pressure boundary (HPB) is the secondary independent barrier while the reactor building serves as final barrier. In the event of a break in the HPB a building flap will open, serving to let the helium escape to atmosphere through a filtered release vent to remove radionuclides.

5. Plant Safety and Operational Performances

The design has the following inherent safety characteristics and design features:

Non-metallic fuel elements – meltdown proof and efficient retention of radionuclides in the TRISO- coated particle fuel during normal operation allows for relatively clean helium circuits and plant operations with low contamination of cooling gas and radioactivity release;

Helium – Chemically and radiologically inert helium is an effective heat transport fluid. Moreover, it does not influence the neutron balance. Helium allows for very high coolant temperatures;

Graphite core structures – allows for high-temperature operations and provides high thermal inertia to the reactor resulting in slow transient response during a loss of active cooling.

(a) Engineered Safety System Approach and Configuration

The following is credited as safety systems (active and passive):

- Coated particle fuel elements;
- Reactor protection system (RPS);
- Core support structures;
- RPV;
- Reactor building

(b) Operational transients and accidents

(i) Key safety features to limit plant transients:

The RCSS insertion depth during normal operation binds around 1.4 delta k-eff. Any spurious signal that would cause full withdrawal of the RCSS would therefore only translate to a higher temperature that would remain below an allowable value shown experimentally not to cause any fuel damage. Furthermore, because the reactor core and its internals are mostly graphite, this provides a high thermal inertia that would cause any transient to be slow-acting.

(ii) Key safety features to avoid core damage:

Features include the reactor core with a low power density, which is very robust and has a high thermal capacity to make the reactor thermally stable during all operational and controlled procedures. Strong negative temperature coefficients also contribute to the excellent inherent safety characteristics.

(iii) Key safety features to contain core damage:

Core meltdown proof – no Core Damage Frequency

(iv) Key safety features to reduce or eliminate large offsite release;

Multiple – independent fission product barriers:

- Qualified UCO TRISO coated particle fuel provides retention of fission products at the source;
- ASME designed pressure boundary provides a further reliable physical barrier to retain the small amount of fission products that may end up circulating in the helium and in graphite dust particles;
- A filtered and vented reactor building.

(v) Diversity and redundancy:

A series of independent fission product barriers provides redundancy and diversity. Failure of any one individual barrier will not impact the performance of another neighboring system/barrier.

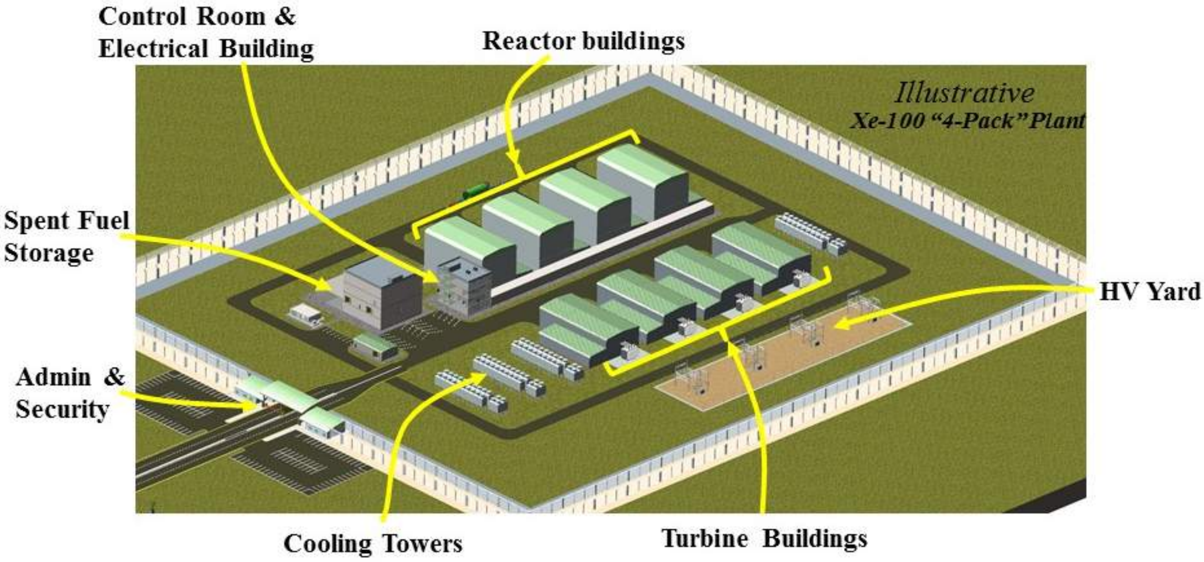
(vi) Worst accident scenario and release:

The depressurized loss of forced coolant (DLOFC) is the worst-case accident scenario. This assumes the RCSS has also failed to insert. Under this scenario no fuel damage will be experienced.

6. Instrumentation and Control System

The I&C system consists of three layers: Distributed control system, investment protection system, and reactor protection system. The human machine interface is configured in such a way that no operator action is required to ensure safe shutdown of the reactor during all events.

7. Plant Arrangement



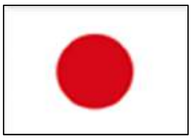
8. Design and Licensing Status

Conceptual design development and U.S. Nuclear Regulatory Commission pre-licensing phase.

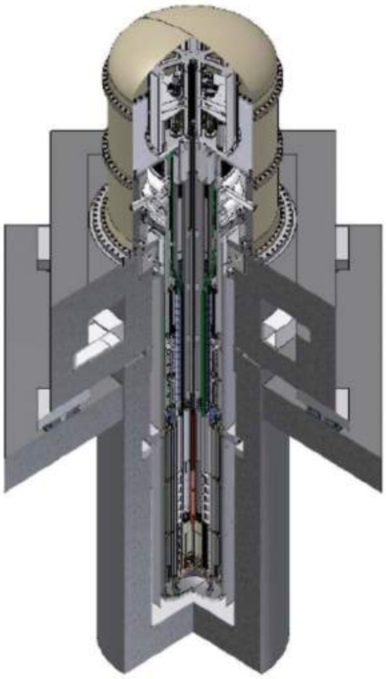
9. Development Milestones

2019	Conceptual Design Development Complete
2021	Basic Design Development Complete
2021	Application submitted to the U.S. Nuclear Regulatory Commission
2025	Start of Construction

FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS



4S (Toshiba Energy Systems & Solutions Corporation, Japan)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Toshiba Energy Systems & Solutions Corporation, Japan
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant/moderator	Sodium
Thermal/electrical capacity, MW(t)/MW(e)	30/10
Primary circulation	Forced circulation
System pressure (MPa)	Non- pressurized
Core inlet/exit temperatures (°C)	355/510
Fuel type/assembly array	Metal fuel (U-Zr alloy) based on enriched uranium
Number of fuel assemblies	18
Fuel enrichment (%)	< 20
Fuel burnup (GWd/t)	34
Fuel cycle (months)	N/A
Main reactivity control mechanism	Axially movable reflectors / fixed absorber
Approach to engineered safety systems	Hybrid (passive+active)
Design life (years)	60
Plant footprint (m^2)	157000
RPV height/diameter (m)	24/3.5
Seismic design	Seismic isolator
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.
Design status	Detailed design

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(t) or 10 MW(e) and 135 MW(t) 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, does not present in its basic design. 4S reactor cores are designed to have a lifetime of 30 years for the 30 MW(t) core and 10 years for the 135 MW(t) 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.

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 50 MW(e) 4S plant can produce fresh water at a rate of 168000 m^3/day; and
- Hydrogen and oxygen production system: hydrogen production rate for the 10 MW(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. Specific Design Features

(a) 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 improvement of public acceptance and safety, minimization of fuel cost and O&M cost, use of uranium fuel with enrichment less than 20%, adequate fuel burn-up and reduction in core size.

(b) 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.

(c) Reactor Core

The core and fuel are designed to eliminate the need for refuelling during approximately 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 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 reactor vessel auxiliary cooling system (RVACS).

(d) 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 burnup reactivity loss over the 30 years lifetime. Therefore, the reactivity control is unnecessary at the 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.

(e) 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/550ºC. The design lifetime of the RV is 60 years as well as the other components.

(f) 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.

(g) 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.

(h) 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.

(i) 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.

4. 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 practically 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 materials. 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 safety features of the 4S are applied.

(a) Engineered Safety System Approach and Configuration

In addition to the inherent safety 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. Both the reflector and shutdown rod are each capable of enough negative reactivity to shutdown the reactor.

(b) 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 a condenser. For decay heat removal during water steam system is not available upon accidents, two independent passive systems are provided; the 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.

(c) 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.

(d) Containment System

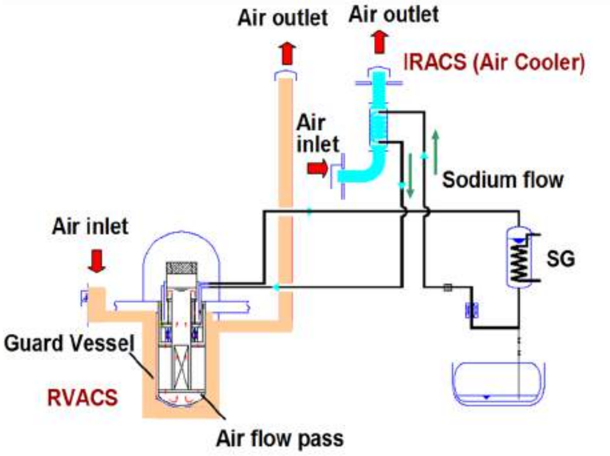
The 4S adopts a cylindrical/spherical containment system. The containment system consists of the GV and the top dome, which covers the upper region of the RV, a shielding plug and the equipment located on the shielding plug. The GV provides the second boundary for the primary sodium at the outer side of the RV. For the mitigation of sodium fire, nitrogen gas is provided inside the top dome.

5. Plant Safety and Operational Performances

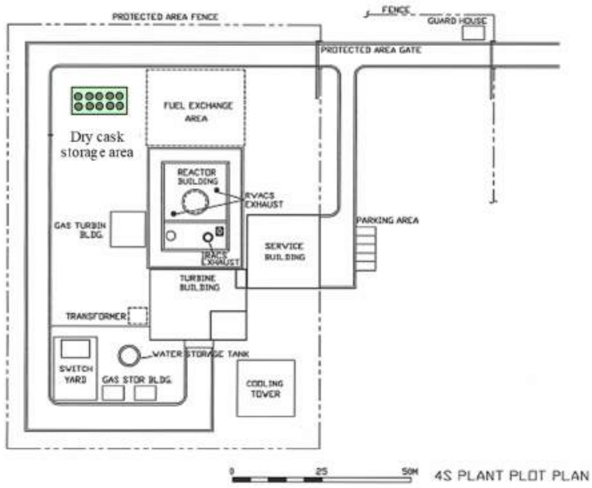
The 4S is designed to operate safely without active involvement of the plant operators. The design features to support such operation include burn-up reactivity swing is automatically compensated by the fine motion reflectors, no need in reloading and shuffling of fuel in the course of 30 years for the 10 MW(e)-4S, reduction in maintenance requirements achieved by adopting static devices and 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.

6. 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 class 1E instruments. The RPS is plant protection system to initiate reactor trip at abnormal plant operation condition.



Engineered safety system



4S plant layout

7. 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; and access and security considerations.

(a) 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 to 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.

(b) Balance of Plant

The BOP including a steam turbine system is located at ground level.

i. 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.

ii. 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.

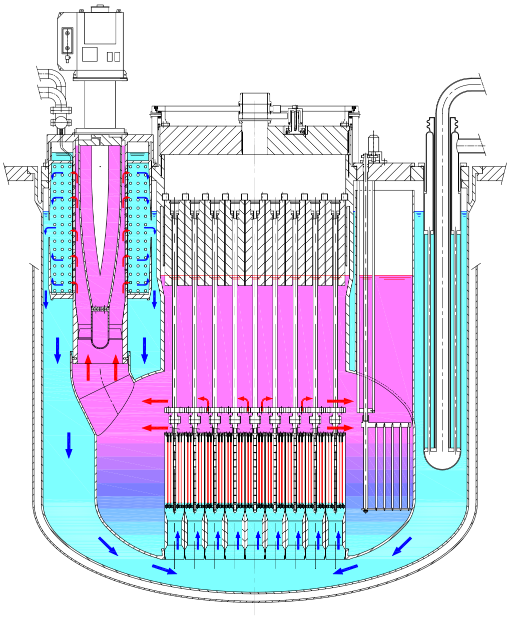
8. Design and Licensing Status

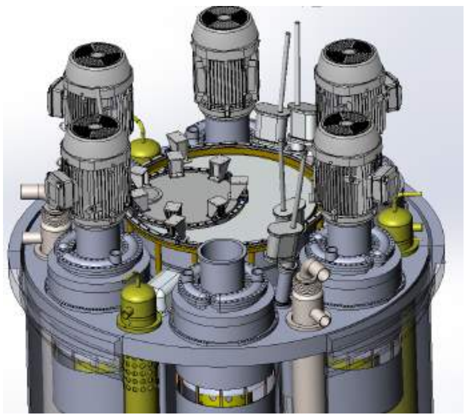
Licensing activities for the 4S design initiated with the U.S. NRC in 2007. In pre-application review, four meetings had been held in the past and fourteen technical reports have been submitted to the NRC. Toshiba is conducting the detailed design and safety analysis for design approval. In parallel, Toshiba continues to look for the customers.

9. Development Milestones

2007	Licensing activity for the 4S design initiated with the U.S. Nuclear Regulatory Commission (U.S. NRC)
2008	Completion of four times public meetings as pre-application review with the U.S. NRC
2013	Completion of submitting 14 technical reports to the U.S. NRC

LFR-AS-200 (Hydromine Nuclear Energy S.àr.l. (HNE), Luxembourg)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Hydromine Nuclear Energy S.à. r.l., Luxembourg
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant/moderator	Lead/none
Thermal/electrical capacity, MW(t)/MW(e)	480/200
Primary circulation	Forced circulation
System pressure (MPa)	Atmospheric, non-pressurized
Core inlet/exit temperatures (ºC)	420/530
Fuel type/assembly array	MOX, hexagonal
Number of fuel assemblies	61
Fuel enrichment (%)	14.6 - 20.4 - 23.2 in Pu
Fuel burnup (GWd/ton)	100
Fuel cycle (months)	80 months, 5-batches
Main reactivity control mechanism	Ex-core, reversed-flag type, rotating staff moves absorbers closer to or away from the core
Approach to engineered safety systems	Active +passive
Design life (years)	60
RPV height/diameter (m)	6.2/6
Seismic design	0.3 g peak ground accelerations
Distinguishing features	Active + passive walkaway safety. No intermediate loops. Simple, compact primary system: ≤ 1 m^3/MW(e) compact reactor building.
Design status	Preliminary design

1. Introduction

The LFR-AS-200 concept is an innovative reactor cooled by molten lead; LFR stands for lead-cooled fast reactor, AS stands for amphora-shaped, referring to the shape of the inner vessel and 200 is the electrical power in MW.

The embodied innovations fully exploit the lead properties and enhance the potential for future commercial deployment, owing to plant simplification and compactness, while behaving passively safe; the reactor represents a significant step forward LFR designs.

The innovations result in the achievement of a very compact reactor: the absence of intermediate loops, the primary system specific volume of less than 1 m^3/MW(e) and the compact reactor building are key-factors for competitive kWh cost.

2. Target Application

Market application is competitive energy production with use of stockpiled plutonium and perspective recycling of minor actinides without burden of long-lived transuranic in the waste.

The breeding ratio is 0.9 without blanket assemblies and can be reduced with core design adaptation if required.

3. Specific Design Features

(a) Design Philosophy

The LFR-AS-200 is an integral pool-type fast reactor with all the primary components installed inside the reactor vessel (RV). Main primary components are six innovative spiral-tube steam generators (STSGs), six mechanical pumps (MPs), flag-type control rods and three + three dip coolers belonging to two diversified, redundant decay heat removal (DHR) systems.

Thanks to the properties of lead, intermediate loops are suppressed with several, special precautions intended to deterministically eliminate any risk of important primary system pressurization: among them, water and steam collectors located outside the RV, short spiral-tube steam generator (STSG) partially raised above the lead free level of the cold collector.

Fuel assemblies (FAs) extend with a stem above the lead free level allowing suppression of the in-vessel refuelling machine. Refuelling is performed under visual control and sealed condition by means of two rotating plugs and an ex-vessel refuelling machine.

(b) Nuclear Steam Supply System

The heat is directly transferred from the primary lead to the water/steam system by means of six STSGs. Feedwater is supplied to the STSG at 340°C to eliminate risks of lead freezing. Steam at 500°C and 180 bar allows a net thermal efficiency of 42%.

(c) Reactor Core

The core is made of 61 hexagonal fuel assemblies (FAs) organized in a triangular lattice of 4 complete rings around the central position.

As lead is a low moderating and absorbing medium the pitch of the fuel rods can be relatively large to permit reduced coolant flow speed and keep the pressure loss below one bar.

The required burnup is achieved with 2400 EFPDs (equivalent full-power days) fuel residence time in the core, corresponding to about 80 months of full-power irradiation; every 16 months one fifth of the fuel assemblies (which has reached the limit of in-pile residence) is discharged and replaced with fresh fuel.

Lead interposed between the core and the amphora-shaped inner vessel (ASIV), larger in the lower part, protects the ASIV from neutron damage and allows elimination of the shielding assemblies, thus contributing to the reduction of RV diameter.

The FAs are substantially supported by buoyancy. Their heads can be interconnected, and the outer heads fixed also to the section of ASIV facing the cover gas, by means of cams, which are integral part of each head. The result is a self-sustaining core anchored to the inner profile of the ASIV, that acts as the core barrel in gas space; the classical supporting structures, diagrid and strongback, which are complicate and of difficult in-service inspection (ISI) are eliminated. Apart the radial restraint of the FAs performed by the bottom part of the ASIV, all other structures supporting the FAs, including the upper part of the ASIV and the reactor roof, are neither wetted by lead, nor exposed to significant transient, thermal stresses and neutron damage.

(d) Reactivity Control

The reactivity control during normal operation is performed by rotation of the staff of flag-type absorbers, which are located in the free space between the core and the lower part of the ASIV, and are thus moved closer to, and away from, the core in order to effectively compensate the criticality swing during an irradiation sub-cycle.

Each staff is contained in a vertical duct, the bottom end of which constitutes the penetration of the ASIV and the upper end terminates underneath the penetration of the reactor roof outside the large rotating plug, and hence without the need of disconnection of the staff drives for refuelling.

(e) Reactor Pressure Vessel and Internals

The RV is shaped as a cylindrical vessel with toro-spherical bottom head and flat roof. The free surface of lead is kept sufficiently below the roof to allow for a gentle thermal gradient between the vessel in contact with lead and the colder roof.

The plenum above the lead free surface level is filled by argon as cover gas.

The roof is made of a circular thick plate with penetrations for components of the primary system and a large-diameter, central upstand, which supports the rotating plugs.

The ASIV, supported by the roof of the reactor vessel, is a mechanical structure that in the upper part accommodates the heads of the FAs and in the lower, larger part, their active section.

(f) Reactor Coolant System

The six vertical, axial-flow MPs are each installed inside the inner shell of, and co-axial with, the STSG. The pump rests on, and is connected to, the upper support plate of the SG by means of a flange which closes the pump's shaft penetration through the reactor roof and supports the variable-speed electric motor of the pump.

The pump is characterized by a short, large-diameter, tapered hollow shaft containing lead brought in rotation by the shaft itself, in order to increase the mechanical inertia of the pump. There are no in-lead pump bearings.

Primary lead circulates inside the FAs from the bottom to the top, then radially across the stem of the FAs, upward to the six MPs and then to the six STSGs that are fed from the bottom. Hot lead flows radially through the perforated inner shell of the STSG and, once passes the tube spirals, flows into the cold collector through a circumferential window located just below the lead's free surface-level.

Cooled lead flows downward inside the cold collector to feed again the core. The RV is always in contact with cold lead below the creep temperature of the steels.

(g) Secondary System

There is no intermediate loop between primary lead and water/steam system. The elimination of the need for an intermediate coolant system to isolate the primary coolant from the water and steam of the energy conversion system represents a significant advantage and potential for plant simplification and improved economic performance.

(h) Steam Generator

The STSG is an innovative SG conceived for compactness and thus it offers several advantages in terms of reactor cost, safety, reactor operability and simplicity of the lead flow path. The SG tube bundle, partially raised above the lead-free surface level of the cold collector, is composed of a stack of spiral-wound tubes. The inlet and outlet ends of each tube are connected to the feedwater header and steam header, respectively, both arranged above the reactor roof to eliminate, in case of their failure, the risk of large water/steam release inside the reactor vessel.

The tube spirals, one spiral for each tube, are arranged one above the other and equally spaced.

The STSG is thermally almost equivalent to a pure counter-current SG, because the feedwater in the tubes circulates from the outer spiral to the inner spiral, while the primary coolant flows radially in opposite direction from the inner shell to the outer shell. Because the flow path of the primary coolant inside the bundle is short, lead flowrate can be increased while limiting the pressure loss.

(i) Pressurizer

The LFR-AS-200 is a lead-cooled fast reactor that operates with the cover gas at nearly atmospheric pressure. Hence, a pressurizer is not required.

4. Safety Features

The safety of the LFR-AS-200 is based on the properties of lead and on the specific design features. One of the most important characteristics of lead as a coolant is its chemical inertness. Lead is a coolant that does not undergo violent chemical reactions, which could possibly lead to high energy release in the event of accident conditions.

In a reactor cooled by lead there is a large margin between the operating temperature and the safety limit and the LFR-AS-200 exploits this margin for actuation of passive shutdown and passive decay heat removal systems, which do not need power sources, operator intervention and logics, and hence are also free from cyber-attacks.

The steam-cladding accidental reaction and resulted in the generation of hydrogen and associated explosions are excluded with lead coolant.

These features open the possibility of integrating in the reactor's concept the "fundamental safety option" of classifying the "core disruptive accidents" as "practically impossible". The ultimate goal is the elimination of the need of an emergency preparedness zone.

(a) Engineered Safety System Approach and Configuration

The reactor protection system is based on two independent systems for reactor shutdown.

The first shutdown system is constituted by rods driven by gravity close to, but outside the core. The rods travel in vertical ducts, which are laid out akin to the ducts of the control absorbers, i.e. there is no need of disconnection of the rod drives for refuelling, because they are located outside the large rotating plug. A tungsten ballast and the emerged part of the rods allow to overcome the buoyance.

The second shutdown system is constituted by rods driven by gravity into selected FAs.

In case of failure of the reactor protection system, expansors placed atop the core on the stem holding each FA provide a magnification effect on the radial core expansion coefficient. When the core outlet temperature exceeds the critical value Tc = 560°C at which the expansors enter in contact, the inherent response of the system is able to bring the system to a new safe equilibrium condition; under such conditions, this coefficient becomes by far the largest, thereby assuming the leading role in every transient.

(b) Decay Heat Removal System

DHR is performed by means of two diverse, redundant systems, each consisting of three identical loops, each loop rated 2.5 MW. Two loops are sufficient to remove the decay heat.

The loops of the first system are filled with lead. Each loop consists of a lead-lead dip cooler and of a lead-air cooler with interconnecting piping, and is passively operated and also passively actuated thanks to the thermal expansion of the cold branch of the loop, which actuates the louvers of the air cooler when its temperature exceeds 400°C.

Each loop of the second system consists of a lead-boiling-water dip cooler (2.5 MW nominal power), of a vessel with 20 m^3 storage water, of interconnecting piping and of a (1 MW) steam condenser installed in order to reduce the storage water consumption and to operate the system for unlimited time without need for make-up water. It is passively operated and actively actuated.

(c) Containment System

The reactor is provided with a concrete containment. The inert coolant operating at atmospheric pressure has a stored specific potential energy much lower than a LWR; this, along with the reduced inventory of secondary water/steam of the secondary circuit, allows for a significant reduction of the size of the containment building. A safety vessel eliminates any loss of coolant accident (LOCA) even in the event of a failure of the reactor vessel.

The containment will be protected from external missile; the same protection will be assured for the new and spent fuel buildings.

5. Plant Safety and Operational Performances

The LFR-AS-200 is designed to operate safely in priority reactor mode i.e. at constant power in the range 20% and full power. Possibility of reactor load following mode is being investigated, but not implemented. Load following mode in the range -10% +10% of Pn is possible, by means of adjusting the amount of spilled steam from the low-pressure body of the turbine and using stored hot water as balance heat sink and source.

During the normal operation the core inlet temperature is maintained constant at 420°C, the steam temperature is maintained at 500°C and the feedwater temperature is maintained constant at 340°C.

The reactor will operate with a ΔT_{core} = 110°C at Pn and 90°C at 20% Pn; this imposes the operation of the MPs at variable speed. Core inlet temperature is maintained constant (420°C) by control of the feedwater flow rate. Steam temperature is maintained constant (500°C) by the control rods.

6. Instrumentation and Control Systems

The instrument and control system will be realized by safety related and non-safety related systems. The safety related systems will include the reactor protection system and the remote shutdown system.

7. Plant Arrangement

The reactor building, the spent fuel building and the new fuel building are located on a common basement. The control room is located above the fuel building. The reactor building extends from 9 m below grade up to 18 m above grade. The turbine generator building is located at ground level.

In addition to the single-module configuration, two additional arrangements are studied at conceptual level:

- a two-modules configuration with a common turbine generator of 400 MW(e) and
- a four-modules configuration with a common turbine generator of 800 MW(e).

A common basement for reactor buildings and fuel buildings is foreseen also in case of multi modules configuration. Economics is expected at turbine generator level but also by the reduction of the number of spent fuel and new fuel buildings.

8. Design and Licensing Status

Not yet started.

9. Development Milestones

2014	Completion of the conceptual design
2018	Preliminary design ongoing

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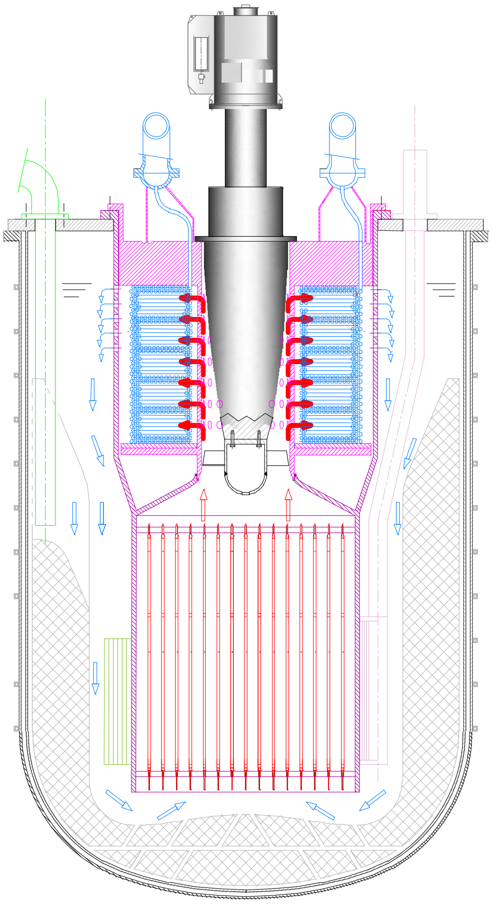
LFR-TL-X (Hydromine Nuclear Energy S.àr.l. (HNE), Luxembourg)



Scheme of LFR-TL-5, LFR-TL-10, LFR-TL-20

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Hydromine Nuclear Energy S.à.r.l., Luxembourg
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant/moderator	Lead/none
Thermal/electrical capacity, MW(t)/MW(e)	15/5; 30/10; 60/20
Primary circulation	Forced circulation
System pressure (MPa)	Atmospheric, non-pressurized
Core inlet/exit temperatures (°C)	360/420
Fuel type/assembly array	LEU, cylindrical cassette
Number of fuel assemblies	N/A
Fuel enrichment (%)	19.75
Fuel burnup (GWd/ton)	40
Fuel cycle (months)	≥100
Main reactivity control mechanism	Ex-core, reversed-flag type, rotating staff moves absorbers closer to or away from the core
Approach to engineered safety systems	Active +passive
Design life (years)	30
RPV height/diameter (m)	3.5/2
Seismic design	0.5g peak ground accelerations
Distinguishing features	Active + passive walkaway safety. No intermediate loops. Simple, compact primary system: about 1 m^3/MW(e) compact reactor building.
Design status	Conceptual design

1. Introduction

The LFR-TL-X is an innovative concept encompassing a family of very small modular reactors (vSMRs) cooled by molten lead; LFR stands for lead-cooled fast reactor, TL stands for transportable Long-lived and X (= 5, 10 or 20) is the electrical power in MW. It is a recent joint proposal by Hydromine and ENEA (Italy).

The objective of this conceptual design is to verify up to which extent it is possible to apply the simplifications embodied in the LFR-AS-200 to design a very small reactor with similar level of compactness.

An important cost parameter to be considered when designing a vSMR is the plant cost per unit power ($/W). It is possible, namely, to reduce the reactor size while reducing power, but the plant $/W ratio is likely to become prohibitively high, owing to the cost of the fuel handling machines and the buildings and facilities for storage of the fresh and spent fuel assemblies, which is relatively independent from the reactor power and hence increases the $/W ratio. It should be considered, too, that it is not wise, for risks of proliferation, to provide the predictably numerous vSMR plants with these fuel storage facilities.

A measure to overcome this proliferation and cost issue is to design vSMRs capable to be transported, complete of the core, to centralised facilities for fuel handling and maintenance of main components. For this design approach to become viable, the vSMR shall be provided with a long-life core and be capable of transport in upright position, in order not to affect its mechanical and thermal-hydraulic configuration while traveling.

The lead-cooled vSMRs derived from the LFR-AS-200 can be designed to comply with both features of long-life core and transportability in upright position. Long-life cores are possible owing to the high breeding capability of the fast reactor; transportability, that is bound to the compact reactor assembly, in particular to the short outline height, is a merit of the very compact pump-spiral-tube steam-generator (pump-STSG) assembly previously conceived by Hydromine for the LFR-AS-200.

No on-site refuelling being requested, it is possible to install only one pump-STSG assembly centreline above the core. This is the main characteristic of the LFR-TL-X reactors, which allows to balance the unfavourable scale effect with respect to the LFR-AS-200 and to approach the unprecedented title of merit of a specific volume of the primary system of about 1 m^3/MW(e) which is an outstanding feature of the LFR-AS-200.

The development of the Hydomine's transportable vSMR thereby exploited and further revealed the flexibility of the LFR, so that, rather than a reference configuration, several options have been investigated, which are different regarding the fuel type, the power level and the thermal cycle. In particular, the maximum power of such a reactor has not yet been defined, first results being positive, however, for increasing it beyond 20 MW(e).

2. Target Application

Assuming that compactness for easy transport and long-life are set, numerous new uses for LFR-TL-X can be identified, in particular for:

- Sites without interconnected grids;
- Offshore oil platforms;
- Mines;
- Islands;
- Naval propulsion.

The deployment of very small LFRs is facilitated by the unique safety features of the LFR, particularly for applications in oil platforms and/or naval propulsion: in exceptional accidents, like ship collision and sinking, frozen lead will always maintain the confinement of the core without water pollution and preserving potential for reactor recovery.

Because of the low operating temperature, the LFR-TL-X can use the same steels already employed in sodium cooled fast reactors and can thereby be deployed in the short term.

3. Specific Design Features

(a) Design Philosophy

The LFR-TL-X is an integral pool-type fast reactor with all the primary components installed inside the reactor vessel (RV). Main primary components are, the innovative STSG, the mechanical pump (MP), flag type control rods, dip coolers of the decay heat removal (DHR) system.

Thanks to the properties of lead, intermediate loops can be eliminated with several special precautions to deterministically eliminate any risk of important primary system pressurization: among them water and steam collectors located outside of the RV, and a short STSG.

Reactor shutdown and DHR is performed by means of diversified and redundant systems which are passively operated and actively actuated but can also be passively actuated when the primary system exceeds certain pre-set threshold temperatures.

(b) Nuclear Steam Supply System

The heat is directly transferred from the primary lead to the water/steam system by means of one STSG. Feedwater is supplied to the STSG at 330°C which is above the lead melting point to eliminate risks of lead freezing. Superheated steam is produced at 400°C, 130-140 bar.

(c) Reactor Core

The core is a monolithic, cylindrical bundle of pins arranged on a triangular lattice so as to be removable in whole for replacement by a fresh core in a centralized facility.

Fuel considered is low-enrichment uranium in metal or oxide form, although more advanced fuels like nitride or carbide can also be considered.

Because of the non-proliferation issue, the enrichment is kept below 20% (19.75%). The consequence is that for very small reactors, the volume of the core is mainly dictated by the need of a sufficient mass of fuel to reach criticality and ensure a reactivity margin to compensate for the reactivity swing during burn up.

The mass of required uranium remains in the range of 2.5-3 t while the thermal power varies from 15 to 60 MW regardless of the fuel form.

(d) Reactivity Control

The reactivity control during normal operation is performed by actuation of ex-core absorbers which can effectively compensate the criticality swing. The control devices are located in the free space between the inner vessel which supports and contain the core and the RV.

(e) Reactor Pressure Vessel and Internals

The RV is shaped as a cylindrical vessel with toro-spherical bottom head and flat roof. The free surface of lead is kept sufficiently below the roof to allow for a gentle thermal gradient between the vessel in contact with lead and the colder roof.

The plenum above the free level is filled by argon as cover gas.

The roof is made of a circular thick plate with small-diameter penetrations for the dip coolers of the DHR1 and of the control and shutdown devices, and a large-diameter central penetration for the inner vessel.

The inner vessel supports the core in the bottom part and in the upper part supports the STSG which contains the MP. All internals are hung to the reactor roof and have no connection with the reactor vessel. The MP, STSG, inner vessel and core are co-axial with the reactor vessel in a matryoshka-type configuration. All primary system components can be removed independently from the core.

(f) Reactor Coolant System

The vertical axial-flow MP is installed centreline in the available space inside the tube bundle of the STSG. The pump rests on and is connected to the upper support plate of the SG by means of a flange which closes the pump's shaft penetration through the reactor roof and supports the variable-speed electric motor of the pump.

The pump is characterized by a short, large-diameter, tapered hollow shaft containing lead brought in rotation by the shaft itself, in order to increase the mechanical inertia of the pump. There are no in-lead pump bearings.

Primary lead circulates inside the cassette core from the bottom to the top, then is conveyed by a funnel-shaped structure to feed upward the MP and then the STSG, which is thereby fed from the bottom. Hot lead flows radially through the perforated inner shell and, once past the tube spirals, flows into the cold collector through a circumferential window located just below the lead's free-level.

Cooled lead flows downward inside the cold collector to feed again the core. The RV is always in contact with cold lead below the creep temperature of the steels.

(g) Secondary System

There is no intermediate loop between primary lead and water/steam system. The elimination of the need for an intermediate coolant system to isolate the primary coolant from the water and steam of the energy conversion system represents a significant advantage and potential for plant simplification and improved economic performance.

(h) Steam Generator

The STSG is an innovative SG conceived for compactness and because it offers several advantages in terms of reactor cost, safety, reactor operability and simplicity of the lead flow path. The SG tube bundle is composed of a stack of spiral-wound tubes, arranged one above the other and equally spaced. The inlet and outlet end of each tube are connected to the feedwater header and steam header, respectively, both arranged above the reactor roof to eliminate, in case of their failure, the risk of large water/steam release inside the reactor vessel.

The SG is thermally almost equivalent to a pure counter-current SG, because the feedwater in the tubes circulates from the outer spiral to the inner spiral, while the primary coolant flows radially in opposite direction from the inner shell to the outer shell. Because the flow path of the primary coolant inside the bundle is short, its speed can be increased while keeping limited the pressure loss.

(i) Pressurizer

The LFR-TL-X is a lead-cooled fast reactor that operates with the cover gas at nearly atmospheric pressure. Hence, pressurizer is not required.

4. Safety Features

The safety of the LFR-TL-X is based on the favourable properties of lead and on the specific design features.

One of the most important characteristics of lead as a coolant is its chemical inertness. Lead is a benign coolant that does not undergo fast chemical reactions which could possibly lead to energy release in the event of accident conditions.

Lead has also good retention capability of volatile fission products and in extreme conditions the reactor can be directly cooled by jets of water, as done at Fukushima, with the further advantage that frozen lead builds up its own sarcophagus (barrier made of frozen lead) and definitively stops radionuclide dispersion.

In a reactor cooled by lead there is a large margin between the operating temperature and the safety limit and the LFR-TL-X exploits this margin for actuation of passive shutdown and passive decay heat removal systems, which do not need power sources, operator intervention and logics, and hence are also free from cyber-attacks.

The steam-cladding accidental reaction and resulted in the generation of hydrogen and associated explosions are excluded with lead coolant.

The ultimate goal is the elimination of the need of an emergency preparedness zone.

(a) Engineered Safety System Approach and Configuration

The Reactor Protection System is based on two independent systems for reactor shutdown.

The first shutdown system is constituted by the same control devices outside of the core. The second shutdown system, not disclosed yet, is also positioned outside of the core and is characterized by a fast-active actuation and by a slow backup passive actuation.

(b) Decay Heat Removal System

DHR is performed by means of two diverse (DHR1 and DHR2), redundant systems, each consisting of two identical loops. One loops is sufficient to remove the decay heat.

The DHR1 system (not yet disclosed) removes heat through the cold collector of the primary system.

Each loop of the DHR2 system is equipped with spirals of square-cross-sectional tubes wrapped around the reactor safety vessel for transfer to a water-steam system the heat transmitted by radiation from the reactor vessel to the safety vessel. The steam is passively condensed in an air cooler which can be actively actuated or even passively actuated above a certain temperature threshold.

(c) Containment System

The reactor is provided with a concrete containment external-missile-proof. The dimension of the containment is kept small by the very low potential energy stored in the coolant (which operates at atmospheric pressure) and the small inventory of water/steam of the secondary circuit.

A safety vessel eliminates any loss of coolant accident (LOCA) even in the event of a failure of the reactor vessel.

5. Plant Safety and Operational Performances

Potential for operation in load following is under evaluation.

6. Instrumentation and Control Systems

The instrument and control system will be realized by safety related and non-safety related systems. The safety related systems will include the reactor protection system and the remote shutdown system.

7. Plant Arrangement

The key for economics of the LFR-TL-X is based on reactor compactness and associated compactness of the reactor building.

The compactness of the reactor, the absence of intermediate loop, of the fuel handling facilities and spent fuel storage allows to drastically reduce the size of the reactor island civil structures.

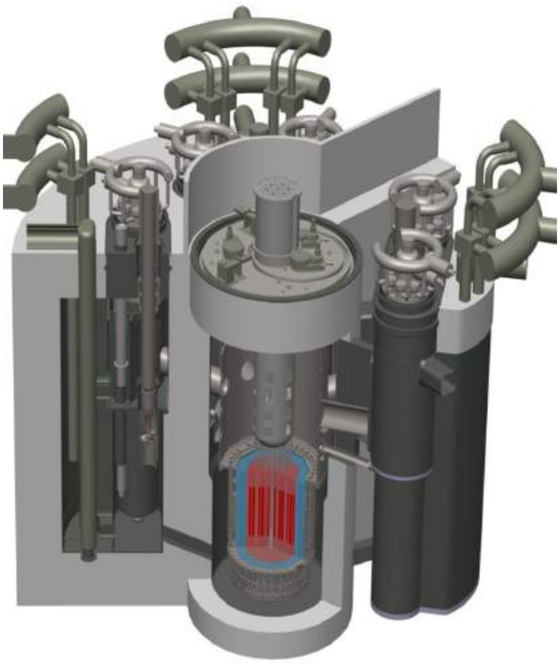
8. Design and Licensing Status

Not yet started.

9. Development Milestones

2017	Initial of the conceptual design
2018	Conceptual design ongoing

BREST-OD-300 (NIKIET, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Liquid metal cooled fast reactor
Coolant/moderator	Lead
Thermal/electrical capacity, MW(t)/MW(e)	700/300
Primary circulation	Forced circulation
System pressure (MPa)	Low pressure
Core inlet/exit temperatures (^{o}C)	420/535
Fuel type/assembly array	Mixed uranium plutonium nitride
Number of fuel assemblies	169
Fuel enrichment (%)	~13.5
Fuel burnup (GWd/ton)	61.45
Fuel cycle (months)	900-1500 eff day
Main reactivity control mechanism	Shim and automatic control rods ($\Delta\rho \approx 12.5\ \beta_{eff}$)
Approach to engineered safety systems	Passive
Design life (years)	30
Plant footprint (m^2)	80x80
RPV height/diameter (m)	17.5/26
Seismic design	VII-MSK 64
Distinguishing features	High level of inherent safety due to natural properties of the lead, fuel, core and cooling design
Design status	Detailed design potential start-up in early 2023

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 subcritical steam turbine and generate electricity of 300 MW(e) 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. 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 and its construction.

2. Target Application

The BREST-OD-300 reactor is designed as a pilot and demonstration power installation intended for studying the reactor facility operation in different modes and optimizing all processes and systems that support the reactor operation. Main goal is practical confirmation of realization of the “inherent safety” concept of the lead-cooled fast reactor, operating in NPP mode in closed nuclear fuel cycle (NFC). After operational tests, the unit will be commissioned for electricity supply to the grid.

3. Specific Design Features

(a) Design Philosophy

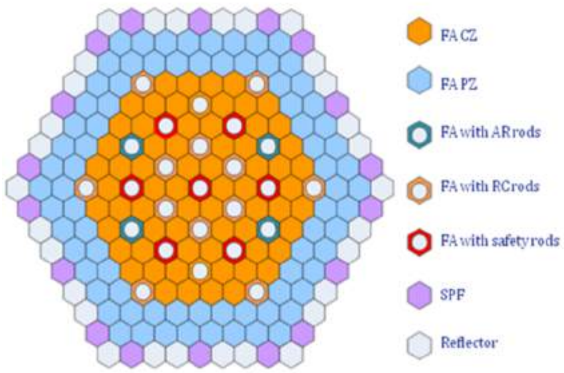
BREST-OD-300 is a pool type reactor design with metal-concrete vessel. The objective of the design is to eliminate severe accidents; complete fuel breeding (equilibrium mode) for self-sustaining and exclusion of accidents caused by reactivity; integral-type arrangement of the first circuit to avoid release of coolant outside the reactor vessel, to eliminate the loss of coolant; to use of low-activated lead coolant with high enough boiling temperature, without violent 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 circulation pumps (MCP), fuel assembly (FA) loading system, control and protection system (CPS), steam-turbine unit, passive decay heat removal system, reactor heat-up system, reactor overpressure protection system, gas purification system and other auxiliary systems.

(b) Nuclear Steam Supply System

Nuclear Steam Supply System has two circuits. The reactor of a pool-type design has an integral lead circuit accommodated in one central and four (4) peripheral cavities of the metal-concrete 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 and MCP, heat exchangers of the emergency and normal cool-down systems, filters of coolant and auxiliary components. The cavities have hydraulic interconnection.

(c) Reactor Core

The lead coolant properties in combination with a dense, high heat-conductivity 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 criticality power excursions. The adopted fuel is mixed mononitride (UN-PuN) that features 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. 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 wrapper can walls. The FA design allows radial coolant flow transfer in the core which prevents overheating of the damaged FA.



(d) Reactivity Control

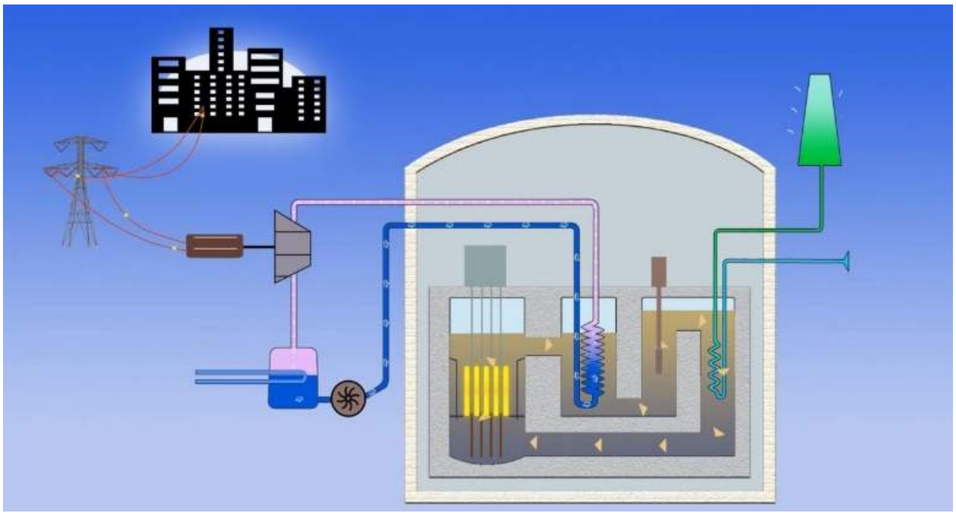
Reactivity control during normal operation is achieved using shim and automatic control rods ($\Delta\rho \approx 12.5$ βeff). Emergency protection rods are also provided ($\Delta\rho \approx 6.5$ βeff).

(e) 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 subcritical steam system as secondary circuit. The secondary circuit is a non-radioactive circuit consisting of one turbine unit with subcritical steam parameters, main steam lines, a feedwater system, secondary side of SGs located in the primary circuit. A standard K-300-15,70-50 turbine unit with two-cylinder (HPC +LPC) steam condensation turbine with intermediate steam superheating and a rotation speed of 3000 rev/min is used. The nominal steam flow rate to the turbine is about 1500 t/h. Oxygen neutral water at subcritical pressure is used in the secondary loop.

(f) Reactor Coolant System

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 535°C, and enters into the shared "hot" coolant drain chamber. Then coolant flows up and enters the SG inlet cavities and inter-tube space via the distributing header nozzles. As flow goes down into 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 it is pumped out back to the pressure chamber. Exclusion of high pressure in the primary lead circuit and a relatively high lead freezing temperature contribute to crack self-healing, which eliminates the possibility of loss-of-core-cooling accidents and release of radioactive lead from the reactor vessel. Lead circulation through the reactor core and steam generator takes place due to the difference between the levels of cold and hot coolant generated by the pumps. Non-uniformity of lead flow through the steam generators with one of all 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.



Since the coolant in primary system is separated in two volumes with isolated free surfaces, in case of potential steam generator tube depressurization accidents, most of the steam bubbles would be released at the second coolant surface in the annular pressure chamber.

(g) Steam Generator

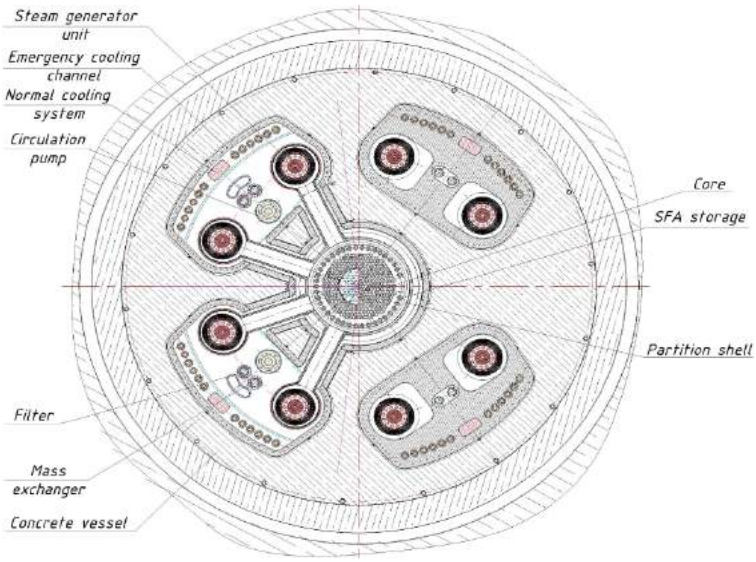
Steam generator is designed with single-walled twisted tubes.

(h) Pressurizer

The BREST-OD-300 reactor does not require a classical pressurizer. Pressure of protective gas in vessel is controlled by gas system.

4. Safety Features

Accidents are avoided due to the inherent safety features of BREST, including small operating reactivity margin ($\Delta\rho < \beta eff$), reactivity fuel temperature coefficient, coolant and core design components, as well as low coolant pressure and temperature at the core inlet and outlet, passive emergency core cooling system, etc. Thanks to these feature, BREST can be considered as a reactor with inherent safety. 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 hazardous steam entrainment into the core and reactor overpressure.



(a) Emergency Core Cooling System

The emergency core cooling system (ECCS) uses pipes, immersed directly in lead of the primary circuit, which may be used to cool down reactor under normal conditions. The system coolant circulation in emergency heat removal mode is provided by natural circulation, with the system coolant under atmospheric pressure. The system consists of four (4) loops. The ECCS air circuit inlet air temperature operates at a minimum and maximum temperature of $-55^{o}C$ and $37^{o}C$ respectively. The system is passive.

5. 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 is 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 80×80 m.

(a) Reactor Building

The reactor building and fuel storage area are equipped with a full monitoring system with closed circuit monitoring system (CCTV).

(b) Balance of Plant

i. Turbine Generator Building

The reference concept of the turbine plant has been developed.

ii. 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.

6. Design and Licensing Status

The development and construction of the BREST-OD-300 reactor is included in the framework of tasks in:

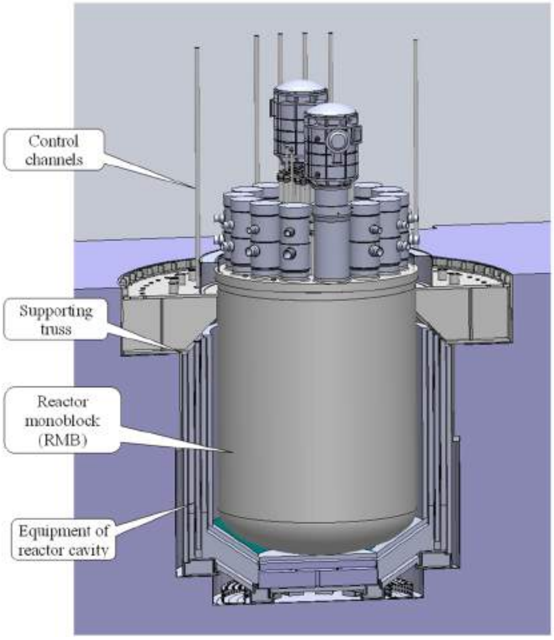
- Development Strategy of Nuclear Power in the Russian Federation in the first half of the 21st century approved by the Russian Government in 2000;
- The Federal Target program “Nuclear Power Technologies of a new generation for the period of 2010-2015 and up to the year 2020” approved by the Russian 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 design is being licensed.

7. 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)
2016	A design study of the BREST-OD-300 NPP with an on-site nuclear fuel cycle (OSNFC) at the Tomsk Site
2020s	First of a kind engineering demonstration plant

SVBR-100 (JSC "AKME-engineering", Russian Federation)



MAJOR TECHNICAL PARAMETERS FOR PILOT PLANT	
Parameter	**Value**
Technology developer, country of origin	JSC Institute for Physics and Power Engineering and JSC EDB "Gidropress", Russian Federation
Reactor type	Liquid metal cooled fast reactor
Coolant/moderator	Lead-bismuth eutectic alloy
Thermal/electrical capacity, MW(t)/MW(e)	280/100
Primary circulation	Forced circulation
System pressure (MPa)	Low pressure
Core inlet/exit temperatures (°C)	340/485
Fuel type/assembly array	UO_2/hex
Number of fuel assemblies	61
Fuel enrichment (%)	< 19.3
Fuel burnup (GWd/ton)	60 (average)
Fuel cycle (months)	7-8
Main reactivity control mechanism	Control rod drive mechanism
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	150,000
RPV height/diameter (m)	7.55/4.53
Seismic design	0.5 (g)
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
Design status	Detailed design for potential construction in 2021

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 ^{210}Po 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(e)) 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: 1) creation of regional NPP and nuclear co-generation plant (NCGP) of low and medium capacity, 2) utilization as part of floating NPPs, 3) 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; and
- Nuclear desalination systems.

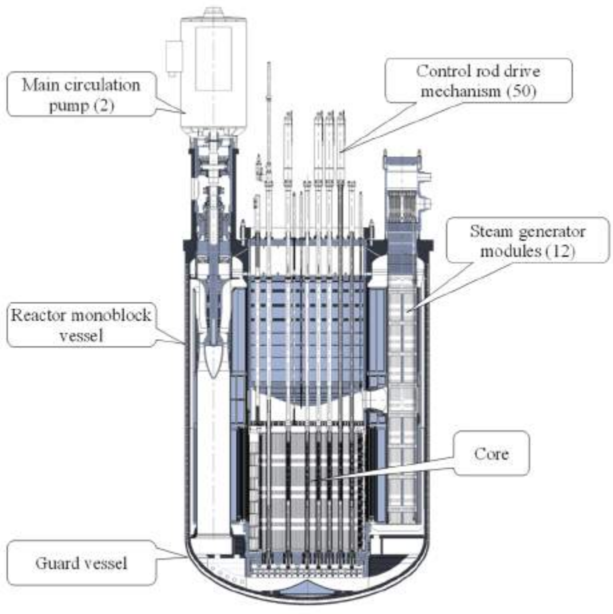
3. Specific Design Features

(a) 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 and it transportability, include railway;
- Possibility of creation of module based structured NPP with power multiplying by adding the reactors.

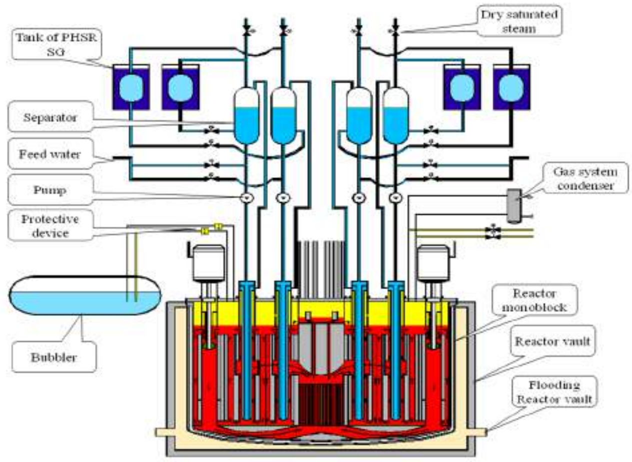
(b) Reactor Coolant System



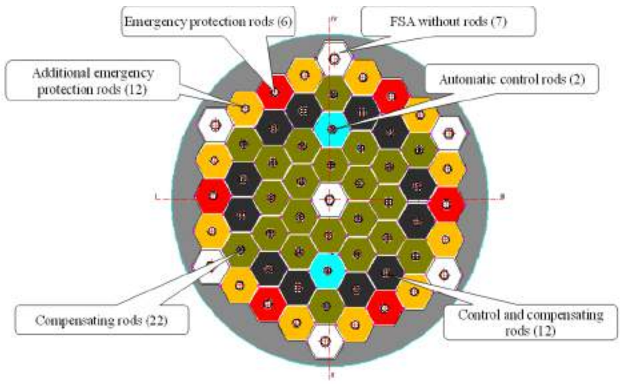
The entire primary equipment circuit of SVBR-100 is contained within a robust single reactor vessel. A protective casing surrounds the single-unit reactor vessel. The reactor transfers generated heat into a two-circuit heat-removal system and SG with forced multiple-circulation secondary coolant system. 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. Being heated in the core, the coolant flows to the inlet of the medium part of the inter-tube chamber of twelve SG modules connected in parallel to each other. 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 surface 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 flows 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 flows 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 the coolant moves upwards via the gap near RMB vessel wall, cooling it and goes into the peripheral buffer chamber.

(c) 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 cool down mode. The gas system condensers are installed in the upper part of the box in the separate concrete compartment.



Reactor and secondary system



Core layout

(d) 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 0.83; MOX fuel can also be used, leading to a CBR just about 1, which provides fuel self-sufficient mode in the closed fuel cycle. 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.

4. Safety Features

Physical basement for high level of inherent self-protection and passive safety:

- First, this is potential energy contained in coolant. At atmospheric pressure, LBE does not store potential energy, which in an event of accident occurrence can cause destruction of defence barriers, core damage and disastrous release of radioactivity. For other reactor types and coolants, there are potential energy of coolant compression and potential chemical energy of coolant's interaction with structural materials (zirconium) (for water coolant), and with water and air (for sodium coolant);
- 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 the possibility 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, and negative void reactivity effect. Efficiency of the strongest absorbing rod does not exceed 0.5 $, that being coupled with technical performance of the control and protection system exclude an opportunity of prompt neutrons criticality in the reactor.
- Elimination of radioactivity release into the environment is insured by the system of disposed defence-in-depth barriers.

The performed analysis has revealed that reactor facility (RF) SVBR-100 is not an amplifier of external impacts and, therefore, the scale of damages will be only determined by the energy of the external impacts. Those type RFs assure their high resistance not only in events of single failures of the equipment and personnel errors but in events of intentional malicious actions when all special safety systems operating in a standby mode can be intentionally disabled. At LBE cooled reactors such catastrophic accidents as Chernobyl or Fukushima disasters as well as fires similar to that occurred at reactor "Monju" are physically impossible or can be easily localized with a purpose to prevent population's exposure to irradiation beyond the NPP site (LOHS type accidents). This is extremely viable for radiophobia elimination and realization of NPP construction in developing countries where the level of terroristic threat is high.

5. Instrumentation and Control Systems

The main principles of the design include:

- Distributed I&C system with several levels of hierarchy and defence 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; and
- Self-diagnostic of I&C programmable devices.

Design specificities are:

- Control 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; and
- 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 %).

6. Pilot plant

General view of pilot plant is shown.

7. 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.

8. Development Milestones

2015	License for placement
2021	License for constructing (planned)
2025	License for operation and commissioning (planned)
2030	Serial production and supply of packaged equipment (planned)

SEALER (LeadCold, Sweden)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	LeadCold, Sweden
Reactor type	Small lead cooled
Coolant/moderator	Lead
Thermal/electrical capacity, MW(t)/MW(e)	8/3
Primary circulation	Forced
System pressure (MPa)	Non-pressurised
Core inlet/exit temperatures (^oC)	390/432
Fuel type/assembly array	UO_2/Hex-can
Number of fuel assemblies	19
Fuel enrichment (%)	19.75
Fuel burnup (GWd/ton)	33
Fuel cycle (full power years)	27
Main reactivity control mechanism	Boron carbide, $(W,Re)^{10}B_2$
Approach to engineered safety systems	Passive
Design life (years)	30
Plant footprint (m^2)	600/10000 (buildings/fence)
RPV height/diameter (m)	6.0/2.748
Seismic design	None
Distinguishing features	Nuclear battery
Design status	Conceptual

1. Introduction

SEALER (Swedish Advanced Lead Reactor) derives from research made at Kungliga Tekniska Högskolan (KTH) and is designed by LeadCold Reactors to meet the demands for commercial power production in Arctic regions of Canada. In Nunavut and the North-West Territories, about ten off-grid communities are of the size that power supply from a stand-alone SEALER unit can be made commercially viable. Moreover, as power constitutes 30-50% of the cost for producing a commodity in the Arctic mining industry, a set of SEALER units can be installed at each mining site, allowing to reduce expenses and to make lower grade ores profitable.

2. Target Application

The primary market for SEALER is constituted by Arctic communities and mining operations which today depend on diesel generators for production of power and heat. The locations where SEALER can be made competitive are not accessible by road, meaning that diesel fuel is transported to the sites either during the short summer period when they are accessible by sea-lift (coastal communities), or during the few winter months when an ice-road can be built and maintained for transport of diesel by truck (mining operations). The cost for transporting and storing the diesel fuel typically exceeds the cost for the commodity itself. The remote nature of these sites means that evacuation of residents and personnel is excluded as means of emergency response, since weather conditions may not allow for air transport to take place for periods lasting up to a week. Therefore, a nuclear reactor to be deployed under such conditions should provide completely passive safety.

3. Specific Design Features

(a) Design Philosophy

Since small reactors in general have high specific costs for hardware and personnel, LeadCold has adopted the following provisions to reduce costs for licensing, construction and operation:

- Elimination of on-site fuel cycle operations by designing a long-life core
- Primary vessel dimensions and weight allowing transport to site by air
- The use of uranium dioxide pellet fuel with an enrichment lower than 20%

The first item, combined with sealing of the inner vessel and locating the reactor under-ground, reduces the cost for security. The second provision allows to prepare an Arctic site for loading of fuel and coolant during the time-window when these can be delivered by secure transport on open sea or by ice-road. A survey of potential air-freighters capable of landing on gravel tracks indicates that a maximum vessel weight of less than 30 tons and a diameter of less than 2.8 meters are desired design objectives. Within this diameter, a long-life shielded core, control and shut-down rods, and heat transport equipment need to be accommodated. The combined requirement of long-life and passive safety is best met by selecting a liquid metal coolant.

(b) Reactor Core

SEALER is designed with the smallest possible core that can achieve criticality in a fast spectrum (taking into account reactivity losses during burn-up) using 19.75% enriched uranium oxide fuel and lead coolant. The core consists of 19 fuel assemblies, 12 control assemblies, 6 shutdown assemblies, 24 reflector assemblies and 24 shield assemblies. The number of fuel assemblies was determined by the requirement that, with an imposed size of the control assemblies identical to that of the fuel assemblies, each of the control assemblies located at the periphery of the core should have a reactivity worth of less than 0.5 dollars, while the combined worth of shut-down elements should be large enough to ensure a safe shut-down state. This may be achieved by a combination of 12 control assemblies and 6 shut-down assemblies. The geometry of the fuel rods, the fuel assemblies and the core was obtained using the multi-variable fast reactor core design and optimisation code ADOPT, in conjunction with the Monte-Carlo code Serpent, applying boundary conditions for temperature and pressure gradients over the core, lead velocity and peak stress in fuel cladding tubes and ducts.

(c) Reactivity Control

The absorbing material of the control rod assemblies is natural boron carbide, the absorber material intended for use in shut-down rods is $(W,Re)^{10}B_2$, a compound having a density significantly higher than lead. The use of this novel material permits passive insertion of the shut-down elements by means of gravity, without introduction of tungsten ballast. Reflector assemblies contain rods with yttria stabilized zirconia (YSZ) pellets, and shielding of the core barrel is provided by 96% enriched boron carbide absorbers.

(d) Reactor Coolant System

The dimensions of the primary system are determined by the desire to make the main vessel transportable by aircraft and by requiring an elevation of the steam generator sufficient for removing decay heat by natural convection of the primary coolant. During normal operation conditions, forced circulation of the lead coolant is provided by eight variable speed reactor coolant pumps. Eight steam generators transfer the heat to the secondary system. The pumps and steam generators are located symmetrically between the core barrel and the reactor vessel, forming eight symmetrical flow paths. Lead coolant enters the hot leg after passing the core where it is driven through the pump suction side and delivered into the annular hot plenum above the steam generator. Hydrostatic head in the hot plenum then directs the coolant downwards and radially through the steam generator tube bundles before discharging to the annular cold leg. Here the coolant moves downwards and enters the cold pool which forms the inlet to the core. The flow is then distributed through individual fuel, control, shield and reflector assemblies by orificing. Upon loss of a single pump, non-symmetrical operation is possible. Loss of a pump during normal operation is compensated by increasing the capacity of the remaining pumps. The primary system is designed to provide significant margins for natural convection which is sufficient to ensure adequate core cooling in all off-normal operating conditions (e.g. after complete loss of forced circulation).

(e) Secondary System

The SEALER plant will employ a Rankine steam cycle power conversion system. The detailed engineering design will be performed at a later stage in cooperation with the selected balance of plant (BOP) supplier. The main conceptual feature of the cycle is feedwater pre-heating using live steam extraction. An auxiliary heater is provided for feedwater pre-heating during start-up conditions. The design objectives are reliability and robustness rather than performance. Based on parameters of small turbo generators currently available on the market, the thermal to electrical power conversion efficiency is estimated at 36%.

(f) Steam Generator

The steam generator of SEALER should be as compact as possible. Incentives include a desire to reduce the diameter of the vessel, as well as to maximize the vertical separation of the tubes from the core. The latter feature would reduce the probability for steam generator tube rupture to result in voids entering the core. It also reduces the damage dose and activation of the steam generator. To this end, the spiral heat exchanger tube design patented by Cinotti is adapted to the available geometry and required performance in SEALER. The resulting bean shape allows to take maximum advantage of the space available between inner and outer vessels and to reduce the number of steam generator tubes. Ten staggered layers of planar spiral tubes with four turns are stacked on top of each other. An analysis of thermal displacements and corresponding stresses during operational and accidental conditions remains to be performed. Whereas bending radii, diameter and length of the steam generator tubes are such that inspection by probe is possible, it should be assessed if inspection, and if necessary, tube plugging, can be carried out at hot-standby temperature.

4. Safety Features

The application of lead-coolant allows the removal of decay heat from the core by natural convection. The same decay-heat can be removed from the primary system by radiation through the primary vessel to the reactor pit. In case of a core disruptive event, the lead coolant forms stable compounds with iodine and caesium having low vapour pressure, reducing the release rate of the latter by more than 99.99%. Hence, the only radiologically significant nuclides are the noble gases, the full release of which does not lead to an exposure requiring evacuation for a population residing at the site boundary ($r > 100$ m from the source of the release).

(a) Engineered Safety System Approach and Configuration

The shut-down assemblies are parked above the core during nominal operation, and are designed to be passively inserted by gravity. Insertion is initiated by cutting current to electromagnets. Back-up batteries are only required for post-accident monitoring.

(b) Decay Heat Removal System

Decay heat is removed entirely by inherent mechanisms, including natural convection of the primary coolant and radiation from the vessel to the reactor pit.

(c) Emergency Core Cooling System

Thanks to the above listed inherent safety measures, no other dedicated emergency core cooling system is foreseen.

(d) Containment System

The SEALER reactor unit is located underground with a concrete top plug for airplane crash protection, and as such does not require a conventional containment in form of a biological concrete shield. The steel confinement of the nuclear island is designed for an overpressure of 0.4 MPa.

5. Plant Safety and Operational Performances

The SEALER is designed to provide a passively safe and secure power source for remote areas in the Canadian Arctic. The reactor is able to produce 3 MW of electric power for 27 full power years without reloading nor reshuffling of its UO_2 fuel. The application of alumina alloyed steels, in conjunction with nominal operational temperatures limited to less than 820 K, provides corrosion protection that is deemed sufficient over the life of the reactor. The here presented safety analysis shows that as designed, SEALER can survive unprotected loss of flow and transient overpower accidents with no consequences for fuel and clad integrity. Moreover, the source term is sufficiently small that a full release of volatile fission products into the coolant at EOL does not require permanent relocation from housing residing beyond 1.0 km from the point of release. Hence, the reactor can be located in the vicinity of the communities to which it provides power, reducing transmission losses on 11 kV grids and simplifying maintenance procedures during difficult weather conditions.

6. Instrumentation and Control Systems

The primary system is instrumented in order to measure neutron flux, temperature, oxygen concentration and lead flow rate.

7. Plant Arrangement

(a) Reactor Building

The Reactor Building is located below grade and contains the primary system (reactor vessel) and all primary auxiliary systems that can contain radioactive material. The systems are located in the confinement structure, which consists of a steel wall enclosing the reactor hall and the reactor pit. In addition, areas are provided for storage of used activated components such as steam generators and pumps. The confinement system is designed for 0.4 MPa overpressure. The concrete building structure does not serve a confinement function for radioactive materials. It contains a top plug designed to provide protection against external hazards, such as aircraft impact.

(b) Balance of Plant

i. Turbine Generator Building

The Turbine Generator Building is not safety classified and is located above grade.

ii. Electric Power Systems

The SEALER reactor will generate electrical power with a 11 kV generator that is connected to the external 11 kV-grid via a conventional medium voltage switchgear, located indoors in the electrical building. Battery back-up is only required for post-accident monitoring.

8. Design and Licensing Status

The design is in its conceptual stage. LeadCold has entered Phase 1 of the Canadian Nuclear Safety Commissions vendor's pre-licensing review. The review is currently suspended.

9. Development Milestones

2014	Pre-conceptual Design
2016	Conceptual Design
2018	Primary system design accepted for publication in Nuclear Engineering and Design.

EM^2 (General Atomics, USA)

Elevation view of EM^2 modular building element employing two modules on a single seismically isolated platform

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	General Atomics (GA), USA
Reactor type	Modular high temperature gas-cooled fast reactor
Coolant/moderator	Helium/none
Thermal/electrical capacity, MW(t)/MW(e)	500/265
Primary circulation	Forced cooling
System pressure (MPa)	13.3 (Peak)
Core inlet/exit temperatures (°C)	550/850
Fuel type/assembly array	UC pellet/hexagon
Number of fuel assemblies	85
Fuel enrichment (%)	~14.5 (LEU)
Fuel burnup (GWd/ton)	~130 (average)
Fuel cycle (months)	~360
Main reactivity control mechanism	Control rod drive mechanism
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	~90,000 (4 modules)
RPV height/diameter (m)	12.5/4.6
Seismic design	N/A
Distinguishing features	Silicon carbide composite cladding and fission gas collection system
Design status	Conceptual design

1. Introduction

The 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). The reactor 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 a Rankine bottoming cycle for 53% net power conversion efficiency assuming evaporative cooling. EM^2 is multi-fuel capable, but the reference design uses low-enriched uranium (LEU) with depleted uranium (DU) carbide fuel material with accident tolerant cladding material, i.e. SiGA^{TM} (silicon carbide technology developed by GA).

2. Target Application

The EM^2 is being developed for the electricity generation and high temperature use.

3. Specific Design Features

The EM2 core was specifically designed to extend the fuel burnup to maximize the fuel utilization with a reasonable amount of initial uranium loading. From this perspective, a fast neutron spectrum was chosen. For the thermal efficiency of the plant, high temperature operation was chosen. These design choices require use of high temperature material for the fuel and core structure. To accommodate high fuel burnup, the fission gases are removed from the fuel and stored in a collection system, which maintains the pressure in the fuel slightly lower than the primary system pressure.

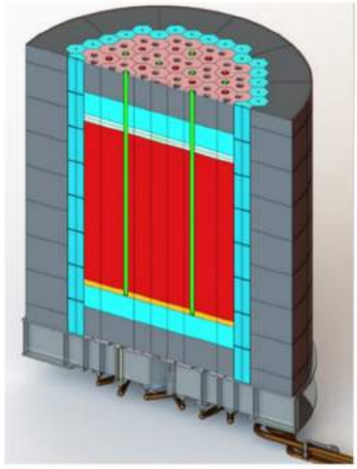
(a) Design Philosophy

EM^2 design philosophy 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; and
- Improved fuel resource utilization, reduced nuclear waste and high proliferation-resistance

(b) Reactor Core

The core is supported by the support floor through the core barrel, which is attached 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 fissile and fertile sections. 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.



EM^2 core arrangement composed of fuel and reflector.

The basic building block of the EM^2 fuel system is the hexagonal assembly, of which there are 85 in the core. 81 assemblies are joined into 27 tri-bundles and 4 remain as individual assemblies. The tri-bundle is located between separate upper and lower reflector blocks. It has a bottom alignment grid, an upper manifold, 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 SiGA.

(c) Fuel Characteristics

Uranium carbide (UC) is used to meet the high uranium loading requirement; UC has a very high thermal conductivity; is compatible with the SiGA 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. Silicon carbide (SiC)-based material, e.g. a SiC-SiC composite, 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.

(d) 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.

(e) Reactivity Control

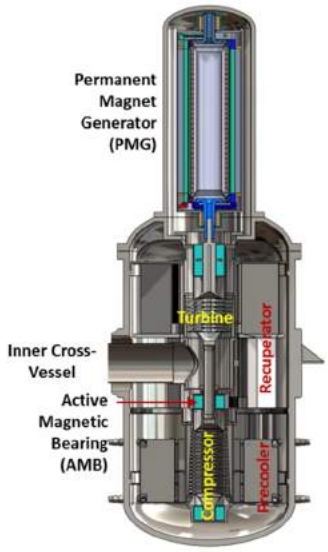
Reactivity control is provided by the 18 control rods and 12 shutdown rods. Both control rod system and shutdown rod system each have sufficient negative reactivity to render the core cold subcritical. The control and shutdown drives are located at the top of the reactor vessel. The control rod drives utilize a ball-screw drive while the shutdown rods use linear motors.

(f) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) is constructed from welded ring-forgings or rolled plate. The vessel contains large penetrations for the two cross-vessels and a flanged hatch. 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 371ºC during normal operation and design basis accidents, which allows the use of SA-533 grade B material.

(g) 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 provided by located in the PCU, while Rankine cycle is implemented in the facility 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. The cycle also incorporates two heat exchangers (HX), a recuperator and a precooler. The generator uses a permanent magnet (PM) rotor to eliminate losses associated with a wound rotor and exciter.



Power conversion unit and generator cutaway

4. Safety Features

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 SiGA cladding. The second barrier is the primary vessel system, which encompasses the reactor, PCU, and Direct reactor auxiliary cooling system (DRACS). The third barrier is the free-standing, below-grade containment. Because the fuel is vented, the fission gas collection system (FGCS) is an extension of the first barrier.

(a) Fuel Cladding

The SiGA cladding is a superior material being developed for accident tolerant fuels. SiGA maintains its strength up to 1700ºC and retains greater than 75% of its strength at 2000ºC.

(b) Fission Gas Collection System

The FGCS protects fuel cladding by capturing vented gaseous fission products in high temperature adsorber. The venting effectively reduces the volatile fission product inventory in the fuel rods and reduces the radioactive source term for accidents.

(c) Direct Reactor Auxiliary Cooling system

The DRACS safely removes core decay heat 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.

(d) 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 (0.14 MPa). The peak pressure rating is 90 psig (0.62 MPa). The design leakage rate is less than 0.2% per day.

(e) Protection from External Events

Protection from external events is provided by locating the reactor containments and the spent fuel storage facility below grade. The roof of the maintenance hall above the reactors and spent fuel storage is an aircraft crash shield per U.S. Nuclear Regulatory Commission (USNRC) 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.

5. Plant Safety and Operational Performances

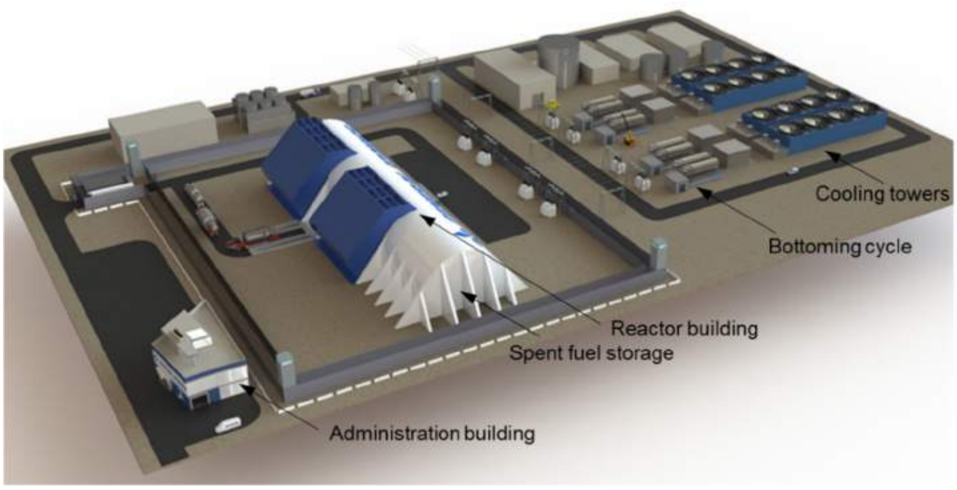
Unlike large power reactors, each EM^2 module utilizes a unique, non-synchronous, variable speed generator with a frequency inverter and generator load commutator to follow load demand. In the automatic load following mode, the generator speed is set by modulating the generator speed. A field-oriented 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.

6. Instrumentation and Control Systems

EM^2 deploys advanced sensors for high operating temperature condition such as solid-state and 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.

7. 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. The plant layout 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.



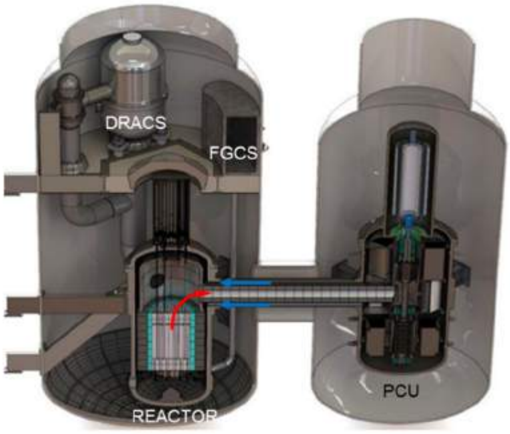
EM^2 plant layout on 9 hectares of land

(a) Reactor Building

The reactor building 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.

(b) Balance of Plant

The primary system is enclosed within a sealed, containment 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. 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.



EM^2 primary heat transport system enclosed in two-chamber sealed containment

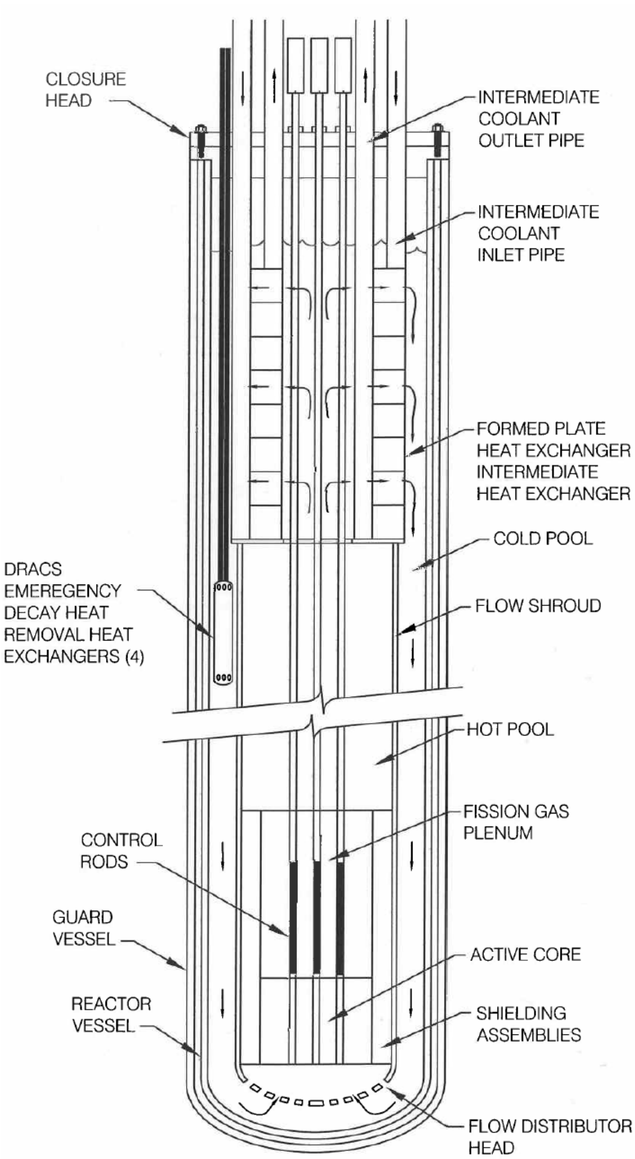
8. Design and Licensing Status

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. The technology-inclusive, risk-informed, performance-based licensing framework developed by the Licensing Modernization Project will be used to develop the NRC license applications.

9. Development Milestones

Phase I	High risk R&D
	Conceptual design of reactor core and power plant
Phase II	Fuel design and demonstration
	High temperature material development
	Demonstration/prototype plant
	Qualification of fuel
	Qualification of plant operation

SUPERSTAR (Argonne National Laboratory, USA)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Argonne National Laboratory, United States
Reactor type	Liquid metal cooled fast reactor (Pool type)
Coolant/moderator	Lead (Pb)/none
Thermal/electrical capacity, MW(t)/MW(e)	300/≈120
Primary circulation	Natural circulation
System pressure (MPa)	0.1
Core inlet/exit temperatures (°C)	400/480
Fuel type/assembly array	Particulate-based U-Pu-Zr metallic fuel with weapons Pu
Number of fuel assemblies	188
Fuel enrichment (%)	< 12
Fuel burnup (GWd/tonne)	55 mean/84 peak
Fuel cycle (months)	15 years
Main reactivity control mechanism	W-Re control rods containing B_4C enhanced in ^{10}B. Control rods are denser than lead coolant
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	TBD
RPV height/diameter (m)	22.86/4.57
Seismic design	Seismic isolation
Distinguishing features	No refuelling equipment on site except during refuelling, heavy liquid metal coolant intermediate circuit, supercritical CO_2 Brayton cycle power conversion, autonomous load following
Design status	Concept

1. Introduction

The Sustainable Proliferation-resistance Enhanced Refined Secure Transportable Autonomous Reactor (SUPERSTAR) is a natural circulation lead-cooled fast reactor (LFR) being developed at Argonne National Laboratory for international or remote deployment on growing electrical grids, integrating lessons learned, innovations and best features from the previous STAR-LM, SSTAR, and ELSY LFR concepts. Initial efforts are focused on a near-term deployable SUPERSTAR demonstrator (demo) representing a first-of-a-kind deployment. Key new features of a SUPERSTAR demo concept include a long lifetime fissile self-sufficient converter core incorporating particulate-based metallic fuel, low pressure drop intermediate heat exchangers (IHXs) incorporating compact diffusion-bonded formed plate heat exchanger (FPHE) technology exposing the reactor vessel to only cold lead coolant, and a natural circulation heavy liquid metal coolant (HLMC) intermediate heat transport circuit. SUPERSTAR development seeks to achieve the largest thermal power limited by primary lead coolant natural circulation heat transport at greater than 100% nominal power inside of a reactor vessel and guard vessel having dimensions limited by the requirement of transportability by rail. The objective is to maximize the economic performance of a transportable natural circulation LFR as measured by the capital cost per unit electrical power.

2. Target Application

International deployment in developing nations with growing economies and infrastructures that cannot support classical economy-of-scale plants. Deployment in remote regions of developed nations.

3. Specific Design Features

(a) Design Philosophy

The goal is to attempt to achieve better economics than SFRs or other NPPs by taking advantage of the unique properties of lead (Pb) coolant together with design innovations and simplifications.

(b) Nuclear Steam Supply System

SUPERSTAR incorporates a supercritical carbon dioxide (sCO_2) Brayton cycle power converter with dry air cooling directly rejecting heat to the ambient air atmosphere. An intermediate HLMC (lead or lead-bismuth eutectic) circuit transports heat from IHXs inside of the reactor vessel to HLMC-to-CO_2 heat exchangers.

(c) Reactor Core Design Approach

SUPERSTAR incorporates a long lifetime core with a 15-year or greater core life and a goal to maximize the capacity factor. There is no downtime for refuelling. The goal is to eliminate downtime due to the need to shut down the reactor for maintenance and repair or due to spurious events resulting in scrams or trips. The initial target lifetime of 15 years is shorter than the 30-year lifetime calculated to be optimal for transitioning to a mainly fast reactor nuclear architecture over an interval of 100 years while capping greenhouse gas emissions at the current level or reducing greenhouse gas emissions to 50% of the current level.

The use of lead-bonded nitride fuel is envisioned for deployment of SUPERSTAR in the future. However, significant testing will be required to qualify nitride fuel and gain regulatory acceptance for its use. The timeframe for nitride fuel qualification is too long for near-term demonstration of a first-of-a-kind SUPERSTAR demo. For this reason, the SUPERSTAR demo utilizes the innovative particulate-based metallic fuel (U-Pu-Zr) fuel form proposed by Walters, Wade, and Hoffman that doesn't require a sodium bond eliminating the need to otherwise incorporate sodium inside of a LFR. The approach in incorporating this particulate-based metallic fuel form is to facilitate near-term deployment by taking advantage of the existing experience and database for sodium-bonded cast metallic fuel together with testing and analyses investigating effects of the particulate-based metallic fuel form. Relative to lead-bonded nitride fuel, the particulate-based metallic fuel form may introduce some drawbacks. Nitride fuel has a high thermal conductivity which limits the peak fuel temperature reducing the positive reactivity contribution resulting from a decrease in fuel temperature during specific transients and postulated accidents. Unlike sodium-bonded cast metallic fuel that also has a high thermal conductivity, the thermal conductivity of particulate-based metallic fuel is expected to be lower due to the effect of the helium-filled internal porosity.

The peak cladding temperature exposed to the lead coolant is limited to 550°C with the goal of enabling the use of existing codified materials in the near term. The lead coolant velocity is limited to 1 m/s.

(d) Reactivity Control

SUPERSTAR incorporates control rods having a net density greater than that of the surrounding lead coolant. This retains the benefit that the rods can be dropped should the control rod drives be deenergized or if the temperature above the core exceeds a threshold temperature thus adding a passive shutdown mechanism. Tungsten-rhenium, W – 25 wt % Re or W- 26 wt % Re, are refractory alloys having melting points of 3120 and 3130°C, respectively; the density of both alloys is 19,700 kg/m^3. The alloy ductile-to-brittle transition occurs over a range of -100 to 25 °C which is well below the reactor system operating temperatures. Both W and Re contain isotopes with large neutron cross sections resulting in transmutation reactions. In SUPERSTAR, the W-Re would be utilized as a neutron absorber without a structural function. Tungsten oxidizes above 400°C such that it must be protected from contacting the Pb coolant containing dissolved oxygen. T91 or T91 with the GESA treatment should be used as cladding for the control rods. Initial calculations investigating the effectiveness of tungsten-rhenium control rods for a preliminary core design indicate a negative answer for the use of the refractory alloy alone as the absorber material for neutrons. However, the use of B_4C absorbers enriched in ^{10}B together with refractory alloy followers above them to increase the control rod net density is a viable approach.

(e) Reactor Pressure Vessel and Internals

SUPERSTAR incorporates eight lead-to-lead IHXs inside of the reactor vessel that cool down the primary lead coolant as it flows outward from the hot pool inside of the cylindrical shroud above the core into the cold pool in the annulus between the shroud and reactor vessel. This feature enables the reactor vessel to be exposed to only the cold lead inside of the cold pool during normal operation. Consequently, the reactor vessel can be fabricated of austenitic stainless steel such as type 316 for which no corrosion protection measures are needed for temperatures below about 425°C. It also eliminates heatup and cooldown transients and thermal stresses in the portion of the reactor vessel that would otherwise be exposed to hotter primary coolant during startup and shutdown operations. There is no fuel handling equipment located inside of the reactor vessel.

(f) Reactor Coolant System

SUPERSTAR is a natural circulation reactor. The benefits of natural circulation are: 1) eliminating the capital cost of primary coolant pumps; 2) eliminating loss-of-flow accident initiators due to pump coastdown; 3) eliminating downtime due to failures of mechanical pumps; and 4) eliminating the potential problem of erosion of the pump impeller due to high velocities in the HLMC. While the use of Ti_3SiC_2 has been identified for pump impellers to handle the last item, it remains to be demonstrated through testing whether this is a practical approach. Babcock & Wilcox stated that their mPower reactor is rail shippable to any point in North America in an envelope of 15 feet (4.6 m) by 75 feet (23 m) and 500 tons (450 tonnes). These limitations on diameter and height are assumed for SUPERSTAR; they limit the maximum core power that can be removed by natural circulation. The development of the SUPERSTAR preconceptual design also incorporates the use of hot channel factors as well as margins for other uncertainties further limiting the reactor power level.

SUPERSTAR incorporates IHXs consisting of formed plate heat exchanger (FPHE, Heatric Division of Meggitt (UK), Ltd.) compact diffusion-bonded heat exchangers. The FPHEs provide a large surface area for interfacial heat transfer. Each IHX incorporates eight FPHE blocks welded together. The primary lead coolant flows outward through straight rectilinear channels while the intermediate lead coolant flows inward through straight rectilinear channels in a Z configuration entering the channels through a nozzle and header on one side of each block near the outer radius and exiting the channels through a header and nozzle on the other side of the block near the inner radius. Each FPHE thus incorporates what is referred to as a Z-I configuration.

A means of setting and maintaining the dissolved oxygen concentration inside of the lead coolant in the pool configuration of the SUPERSTAR vessel shall need to be identified or developed. Iron and other steel constituents are removed from the cladding and structures over time and enter the lead coolant. An approach for filtering them from the coolant inside of the vessel also needs to be developed.

(g) Secondary System

SUPERSTAR incorporates an intermediate heat transport circuit utilizing lead intermediate coolant to exclude the CO_2 working fluid from inside of the containment. Because CO_2 is a molecule and decomposes in a radiation field, it is not possible to have CO_2 inside of a primary Pb-to-CO_2 heat exchanger immersed in primary lead inside of the reactor vessel. Decomposition of CO_2 could give rise to species that more aggressively corrosively attack stainless steel and turbomachinery alloys as well as long chain molecules having a composition similar to CO that can gum up rotating machinery. Activation of the Pb primary coolant through the formation of excited states of lead isotopes, predominantly $m^{207}Pb$ which is a metastable/excited state of ^{207}Pb having a half-life of 0.806 second created by neutron inelastic scattering, would result in gammas irradiating the CO_2 causing CO_2 decomposition. Incorporation of an intermediate heat transport circuit isolates the CO_2 from the primary coolant. However, the intermediate lead coolant must be suitably shielded from neutrons from the core as the intermediate coolant flows through the IHXs, headers, and piping inside of the reactor vessel to prevent its activation as well.

Ideally, Pb is the first choice for the intermediate coolant because the postulated rupture of the boundary between the primary and intermediate coolants would simply introduce lead into the primary coolant. However, if the primary coolant core inlet temperature is 400°C, then the intermediate coolant inlet temperature to the IHX will be lower (e.g., 380°C) which may not provide sufficient margin above the lead freezing temperature of 327°C to exclude concerns about freezing of the intermediate coolant. Consequently, the use of lead-bismuth eutectic (LBE) for the intermediate coolant is also an option; a negative for LBE is that postulated rupture of the primary-to-intermediate coolant boundary would introduce LBE into the primary lead coolant where the Bi would undergo transmutation reactions forming ^{210}Po.

(h) Steam Generator

Compact diffusion-bonded HLMC-to-CO_2 heat exchangers are utilized.

(i) Pressurizer

The argon cover gas above the primary lead pool maintains the lead at a low near atmospheric pressure. An expansion vessel with argon cover gas on each intermediate cooling loop maintains the intermediate coolant at the desired low pressure.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

The combination of lead coolant, metallic or nitride fuel, and a fast neutron spectrum results in large negative reactivity feedbacks with increasing temperature providing passive shutdown. Natural circulation at greater than 100% nominal power and elimination of coolant pumps eliminates loss of flow accident initiators.

(b) Decay Heat Removal System

The sCO_2 Brayton cycle power converter incorporates a separate sCO_2 shutdown heat removal system with its own HLMC-to-CO_2 heat exchangers, CO_2 pumps, and air coolers. Four Direct Reactor Auxiliary Cooling System (DRACS) heat exchangers (HXs) for emergency decay heat removal are installed inside of the reactor vessel. By locating the DRACS HXs inside of the downcomer, they are normally immersed inside of the cold lead exiting the IHXs. A passive means of initiating heat removal through the DRACSs when needed is to design the louvers on the air heat exchanger to require electrical power to close such that they passively open upon the loss of electrical power. To always maintain a small natural circulation flow of the DRACS circuit intermediate coolant, the louvers incorporate an orifice that results in a small amount of heat rejection to air and natural circulation of the DRACS circuit intermediate coolant. In this case, locating the DRACS inside of the cold lead further reduces the heat removed from the lead coolant during normal operation. Optionally, the DRACS circuits could be designed such that the louvers are always open and accept a greater loss of core power. A key feature of the DRACS circuits is that the DRACS intermediate coolant flow is driven by natural circulation. This imposes a requirement on the design of the DRACS HXs and piping to limit the frictional pressure drop.

(c) Emergency Core Cooling System

Lead is a low pressure coolant and does not undergo flashing. There is a guard vessel surrounding the reactor vessel to maintain a sufficiently high faulted level of lead inside of the reactor vessel to maintain natural circulation heat transport in the event of a reactor vessel leak.

(d) Containment System

SUPERSTAR incorporates an intermediate heat transport circuit utilizing lead intermediate coolant to exclude the CO_2 working fluid from inside of the containment. This eliminates the need to include a CO_2 line break in the containment design basis. Thus, the containment does not need to have a significant pressure retention capability simplifying the containment structural design and reducing the cost associated with the containment. The containment must withstand the thermal loads resulting from the postulated release of either lead primary or intermediate coolant as well as burning of any combustibles inside of the containment. The containment only needs to have a small pressure retention capability to retain the postulated release of radionuclides. It is expected that the cost savings to be realized from a containment having a low pressure retention capability will outweigh the costs associated with the intermediate heat transport system.

5. Plant Safety and Operational Performances

SUPERSTAR incorporates autonomous load following to simplify plant load following operations and reduce operator workload. Autonomous load following is a feature of the combination of lead coolant, metallic or nitride fuel, and a fast neutron spectrum. SUPERSTAR incorporates passive safety.

6. Instrumentation and Control Systems

These systems will be similar to those installed in current SFR designs.

7. Plant Arrangement

(a) Reactor Building

The SUPERSTAR reactor and containment can be located underground as protection against aircraft crash. A removable berm might also be erected at grade level providing additional protection. Refuelling of SUPERSTAR requires access to the containment which requires removal of any berm or soil atop the containment structure for an underground containment. Access to the core requires removal of control rod drives and control rod drivelines and/or IHXs to make room for temporary installation and operation of an in-vessel fuel handling machine. Refuelling equipment is brought on site by an itinerant refuelling crew only during refuelling enhancing proliferation resistance.

(b) Balance of Plant

The BOP includes the sCO_2 Brayton cycle and other auxiliary systems.

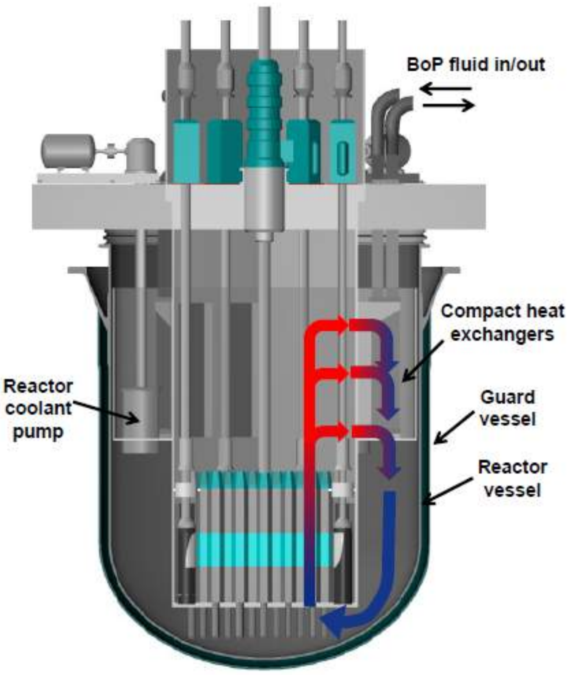
8. Design and Licensing Status

Licensing activities have not been initiated.

9. Development Milestones

The SUPERSTAR concept was developed in 2010 and 2011. Work was suspended thereafter due to lack of funding.

Westinghouse Lead Fast Reactor (Westinghouse Electric Company LLC-USA)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse Electric Company, LLC, U.S.A.
Reactor type	Pool-type, liquid metal cooled fast reactor
Coolant/moderator	Lead / fast-spectrum
Thermal/electrical capacity, MW(t)/MW(e)	950 / >450 (Net)
Primary circulation	Forced circulation
System pressure (MPa)	Nearly atmospheric
Core inlet/exit temperatures (ºC)	420/600 (or higher, pending materials demonstration)
Fuel type/assembly array	Oxide, with provision for transition to UN
Number of fuel assemblies	--
Fuel enrichment (%)	≤19.75%
Fuel burnup (GWd/ton)	≥100
Fuel cycle (months)	≥24
Approach to engineered safety systems	Passive: IAEA passive safety category B goal
Design life (years)	60 components, 100 structures
Plant footprint (m^2)	--
RPV height/diameter (m)	Approx. 8.0 / 7.5
Seismic design	Isolation of primary systems
Distinguishing features	Low pressure reactor vessel; high pressure containment vessel not required; High boiling point coolant; compact configuration with hybrid microchannel heat exchangers; no moving parts or actuation for decay heat removal; high efficiency; non-reactor based load follow
Design status	Conceptual design

1. Introduction

The Westinghouse lead fast reactor (LFR) is a medium-output, modular plant harnessing a lead-cooled, fast spectrum core operating at high temperatures in a pool configuration reactor. High temperature operation and unique configuration of the compact reactor vessel (RV) and guard vessel (GV) present an opportunity for automatically actuated passive cooling without the need for instrumentation and control signals or moving parts. The simplicity of its safety systems, high-efficiency supercritical CO_2 (sCO_2) balance of plant (BoP), compact reactor building, absence of high-pressure containment, streamlined modular construction, and integrated non-reactor-based load following capability results in unparalleled economic potential, allowing the Westinghouse LFR to supply clean energy in the most challenging of market conditions.

2. Target Application

The Westinghouse LFR is designed to be a versatile plant, with baseload electricity production and load levelling as the primary design focus, but with the capability to fulfil a range of non-electricity applications such as process heat, desalination, and hydrogen production needs according to market demand. Its output is sufficiently small to integrate into lower-capacity grids while also being substantial enough to be used in standard baseload plant applications. High temperatures permit extremely high BoP efficiencies and use for many process heat applications. Furthermore, a low cost per MW(e) permits further electrical boosting of process temperatures while still being competitive at any temperature. Similarly, high temperature and BoP efficiency, combined with projected low plant cost, permit the use of hybrid heat/electricity methods providing the potential for cost-effective hydrogen generation.

Integrated thermal energy storage using low-cost materials, coupled to standard BoP equipment, allows for non-reactor-based load follow to complement non-dispatchable energy forms while maximizing energy production. These capabilities could allow for the increased use of renewable technologies, making nuclear power and renewables complementary. The proposed sCO_2 power cycle is air-cooled and has high turbine outlet temperatures allowing for a greatly-expanded variety of plant siting options and applications, including combined heat and electricity in captive markets.

The use of a fast-spectrum core also permits a wide variety of fuelling options and strategies. These range from once-through high-burnup cores, breed/burn extended life cores, MOX-fuelling for most effective plutonium utilization, and actinide burning closed cycle applications, to satisfy market demand, customer preference, and nuclear energy policy in the country of deployment.

3. Specific Design Features

(a) Design Philosophy

The Westinghouse LFR was designed to harness the outstanding safety, neutronic, and thermal characteristics of molten lead coolant to simplify safety, reduce overall plant size, and maximize the BoP efficiency. This, when coupled to advanced fast-spectrum fuel cycles and non-reactor based load follow, drives towards the ultimate goals of being economically competitive against any competing energy form in free markets while maintaining mission flexibility to customers worldwide.

(b) Reactor Coolant System

The Westinghouse LFR features a novel reactor design configuration utilizing high power density hybrid microchannel heat exchangers (MCHE) integral to the upper part of the core barrel. The compactness of the MCHE design reduces the overall height and volume of the RV. This arrangement allows the reactor coolant pump (RCP) impellers to be placed in the lower temperature coolant discharging from the MCHEs, reducing service temperature of the rotating components and related material design challenges. The figure above depicts the primary coolant flow path. After exiting from the MCHE, primary coolant at cold pool temperature flows through the RCPs and is sent to the core, where it is heated and discharged to the upper plenum. The primary coolant is then allowed to flow radially through the MCHE and return to the RCP inlet plenum. The configuration also ensures the entire RV is in contact with lower temperature coolant, easing material requirements for this component.

(c) Primary Heat Exchangers

Located in the reactor vessel pool and integrated into the core barrel internal structure are six hybrid microchannel primary heat exchangers (PHEs) to transfer heat to the secondary side working fluid; sCO_2 at 250-300 bar of pressure. With no welds in the main body, very small CO_2 channels within diffusion bonded plates, and sCO_2 headers located outside of the RV, a robust structure capable of maintaining extreme pressure differentials is created. When combined with the lack of exothermic reaction between primary lead coolant and BoP fluids, these elements allow PHE's placement directly into the RV pool with limited risk of a significant RV pressurization event, eliminating the need for an intermediate heat transport loop present in other advanced reactor technologies, resulting in a more cost competitive plant.

(d) Reactor Core

The core employs a conventional configuration, featuring solid fuel in cylindrical cladding. Various options are being investigated for the commercial fleet, including uranium nitride, while considering higher technology readiness oxide fuels (UO_2 and MOX) for the nearer-term prototype plant. An average discharge burnup of approximately 100 GWd/ton HM is envisaged in the prototype plant, which employs a 15-15Ti-type austenitic steel cladding, such as D9, with an increase in burnup and operational temperature in the follow-on commercial plant following the adoption of higher-temperature and more irradiation resistant materials as they become available for use. The core design uses ^{235}U fuel with enrichment <20% while maintaining the option to burn Pu-containing fuel.

(e) Balance of Plant

While supercritical water remains a high-technology-readiness, high-performance option for the LFR, a significant advantage can be captured through the use of sCO_2. sCO_2 offers significant efficiency and size benefits. At 600°C, efficiency is expected to surpass 48% Net, with 700°C variants capable of efficiency in the mid 50% range. Not only is sCO_2 efficient, but the associated turbomachinery size is much smaller, allowing for reduced building and foundation sizes, reducing the overall plant cost. Additionally, the characteristics of the sCO_2 cycle being developed for this application are such that air-cooling is not only possible, but preferable. The cycle being proposed delivers exceptional efficiency, reduced building volume, competitive capital cost economics, and greatly-reduced water usage.

(f) Non-Reactor Load Levelling

Westinghouse is currently developing thermal energy storage systems capable of providing load-levelling for thermal power plants. The Westinghouse LFR is designed to accommodate such systems, recognizing the important role in fulfilling needs of future and diverse energy markets. The storage system maximizes economic advantage by being integrated with the same turbine and generator as would be used for power generation, managing supply fluctuations produced by renewable sources by storing heat energy when electricity demand is low and selling produced electricity from that stored heat when it is high, all while maintaining the reactor core at full-power. For supercritical water systems, this is accomplished through manipulation of feedwater and turbine extraction flows to either increase or decrease the mass flow through the turbine. For sCO_2, the abilities of CO_2 to work in a heat cycle are harnessed, allowing storing of energy at lower temperatures as compared to those intended for use within the cycle. Manipulation of existing process flows within the sCO_2 power conversion cycle is used to store and deliver this heat, and ultimately electrical power. For both cycles, the storage of heat is in a modular assembly of low-cost, high-performance concrete plates. The combination of these technologies shows promise in providing a simplified solution which incorporates nuclear base-load, low-cost renewable energy, and variable-output grid support into a single, economic package.

The same systems and components used in the energy storage system appear capable to be integrated with solar-thermal boosting, with the potential to reduce the amount of necessary storage while marginally increasing the plant's effective size. As much of the equipment necessary is already employed by the plant and energy storage, the additional marginal cost of the collectors also appears to be economical.

4. Safety Features

The Westinghouse LFR harnesses the inherent favourable safety characteristics of the lead coolant to simplify the reactor design and lower plant cost while allowing for the highest level of safety.

(a) Robust, Inherently Safe Design Characteristics

The following characteristics of the design enhance its safety inherently:

- Thermophysical properties of lead, including its high boiling point (1745°C); atmospheric pressure operation; lack of violent chemical reaction with water, air, and sCO_2; ability to retain some key fission products; shielding capability; high thermal conductivity; and, when combined with a pool-type primary system configuration, high thermal inertia
- Lead's excellent neutronic properties for operation in fast neutron spectrum allows the fuel rod lattice to be opened relative to sodium fast reactors, resulting in a minimum neutronic penalty while providing a significant enhancement in natural circulation capability during accidents, due to the associated reduction in core pressure drop
- Integral, pool-type configuration of the primary system eliminates primary line break, thus eliminating loss of coolant concerns by design
- Favourable reactivity feedback typical of liquid metal fast reactors
- Robust, microchannel hybrid PHE reduces chance of secondary break and also greatly reduces its severity; and
- Underground placement of components important to safety

(b) Passive Heat Removal

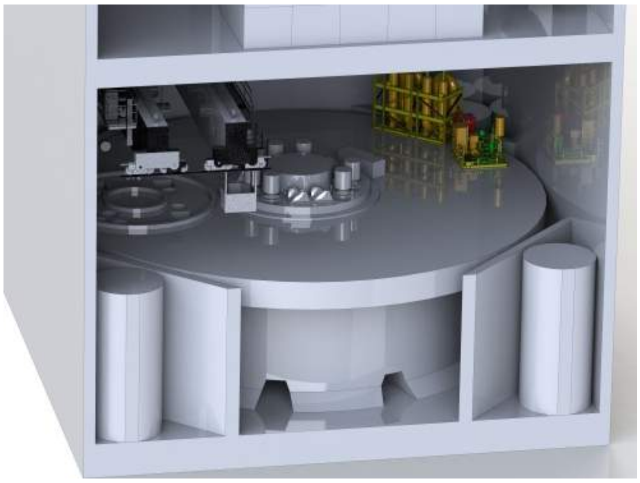
The LFR harnesses its high temperature capability in order to use radiation heat transfer between the RV and GV to remove reactor core decay heat during a safety event. The GV is submerged in a pool of water sufficient to remove heat from the reactor through boiling long enough to ultimately transition to natural circulation air-cooling of the RV through a Reactor Vessel Air Cooling System (RVACS). The system is designed such that parasitic losses during normal plant operation are not significant, but a relatively minor increase in RV wall temperature is sufficient to increase heat loss to match core decay heat, as radiation heat transfer is a function of temperature to the 4^{th} power.

(c) Pressurization Events

In addition to not having any source of credible RV pressurization events originating from the BoP, no BoP plenums, piping, or headers will be located within the primary containment, thus removing the potential for large leaks in nuclear-related areas. The limiting credible break size is reduced to the heat exchanger microchannels in the diffusion bonded block of the PHE. While small in terms of leak size, the pressures resulting from a microchannel break become substantial over time and still require mitigation. Due to the target of IAEA Passive Safety Category B, no isolation valves are credited for use in PHE leaks (although non-safety isolation valves will be present). Instead, filtered venting of the leaked BoP fluid is envisioned. The cleanliness of the BoP fluid prior to break, as well as the low quantity of radionuclides expected to be entrained in the escaping gas, enables the application of this solution to eliminate the need for a high-pressure containment and also prevents a large buildup of stored energy.

(d) Reactivity Transients

A reliable, diverse, redundant reactivity control and shutdown system will ensure protection from this class of events.



5. Instrumentation and Control Systems

A design goal for the development of the Westinghouse LFR's plant safety systems is to not rely on signals from the instrumentation and control (I&C) system. As a result, the majority of components, systems, and software used to control the plant will be commercial grade. To support anticipated licensing requirements, a reduced number of "safety-grade" systems will be incorporated, such as post-accident monitoring.

6. Plant Arrangement

As previously-noted, the Westinghouse LFR design results in a compact nuclear system. The GV and RV will be suspended from a seismically-isolated platform into a safety pool. The safety pool into which the GV is submerged will reside in the lower levels of the plant. All of these areas, as well as an area located above the reactor platform, will be located underground. At grade elevation will be an impact shield sufficient to provide protection from external threats. No nuclear-related systems will be located above grade and all components and systems with safety significance (outside of the safety pool itself) will be placed on the isolated platform, thus truly providing a "nuclear island."

The use of an sCO_2 system on the LFR reduces the size of rotating BoP components significantly and eliminates the need for a large condenser to be located in the turbine building, below the turbine. Recuperators, being large masses of stainless steel, will be located outside at grade. Similarly, the air-cooled condensers will also be located in the yard. This arrangement results in a compact nuclear/turbine island with significant, large-component erection performed outside of the plant, allowing for more parallel construction activities and reduced construction duration.

7. Design and Testing Status

Westinghouse has established an international team of partners dedicated to successfully delivering a high performance commercially viable plant. Harnessing the respective competencies and talents of all organizations, multiple parallel efforts are under way. A number of test facilities are currently being used or planned to be built in order to quantify and test the unique behaviours and properties of lead coolant, material compatibility, and new safety-related phenomena.

8. Development Roadmap

Achieving a plant with the number of innovative features as described herein is a considerable step forward for the commercial nuclear power industry. As a result, a staged approach is planned to demonstrate key aspects of the design while additional testing is performed to validate further enhancements to the plant performance.

The LFR targets operation by 2030 for a ≤300 MW(e) prototype that will demonstrate basic feasibility during an initial phase of operation. As operational experience is gained and higher-performance materials are qualified, subsequently, a higher output ~450-510 MW(e) first of a kind (FOAK) plant representative of the commercial fleet will be licensed and deployed.

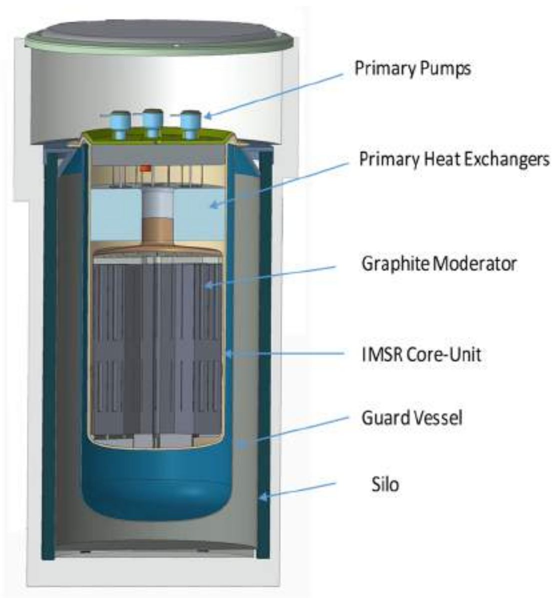
Lower operating temperatures are envisioned for the prototype plant to allow demonstration of key features of the plant at temperatures for which relatively conventional materials have already been extensively tested in liquid lead, e.g. SS316; thereby reducing the design and licensing risks of the prototype plant. Meanwhile, testing of more advanced materials, and subsequently of individual components at higher temperatures, will be performed in a controlled environment to qualify them for use in evolved designs.

Through the use of compact heat exchangers and supercritical water or CO_2 secondary fluid, it is anticipated that the reactor vessel for the FOAK plant will be essentially the same as that used for the prototype plant. The primary heat exchangers in the prototype LFR are anticipated to utilize supercritical water on the secondary side. When a sCO_2 power conversion package and higher operating temperatures are adopted, an increase in plant efficiency of approximately 10% will be realized. Other components are being designed with scalability in mind from the start, such that a minimal amount of re-design and re-licensing efforts are necessary for the transition from the prototype to the FOAK.

While the FOAK LFR will achieve ultimate economic potential thanks to its further performance enhancements, the fundamental advantages of lead as a coolant, as well as innovative design solutions adopted by Westinghouse, give confidence that the prototype reactor's design will be competitive in many markets. LFR serves to answer the call of future energy markets, allowing a multitude of missions, fuel cycles, and operating strategies to be adopted in various locations and nations around the world.

MOLTEN SALT
SMALL MODULAR REACTORS

Integral Molten Salt Reactor, Terrestrial Energy Inc., Canada



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Terrestrial Energy Inc., Canada
Reactor type	Molten salt reactor
Coolant/moderator	Fluoride fuel salt/graphite
Thermal/electrical capacity, MW(t)/MW(e)	400/190
Primary circulation	Forced circulation
System pressure (MPa)	< 0.4 (hydrostatic)
Core inlet/exit temperatures (^{o}C)	625-660/670-700
Fuel type/assembly array	Molten salt fuel
Fuel enrichment (%)	Low enriched uranium (<5%)
Fuel cycle (months)	84 months before Core-unit replacement
Main reactivity control mechanism	Short term: negative temperature coefficient long term: online liquid fuel additions
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	45,000
RPV height/diameter (m)	7.0/3.6
Seismic design	~0.3 (g)
Distinguishing features	Completely sealed reactor vessel with integrated pumps, heat exchangers and shutdown rods; core-unit is replaced completely as a single unit every seven years
Design status	Conceptual design complete – basic engineering in progress

1. Introduction

The integral molten salt reactor (IMSR®) – is a 400 megawatts-thermal (IMSR400) small modular molten salt fuelled reactor. The IMSR® is an integral nuclear reactor design. It features a completely sealed reactor vessel with integrated pumps, heat exchangers and shutdown rods all mounted inside a single vessel; the IMSR® core-unit. The sealed core-unit is replaced completely 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. The IMSR400 achieves the highest levels of inherent safety as 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.

2. Target Application

The IMSR® plant is designed to accommodate various load users, from baseload to load-following. By utilising a simple, modular, and replaceable core-unit, high reliability ratings are achieved. 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, like aircraft jet engine production lines for example.

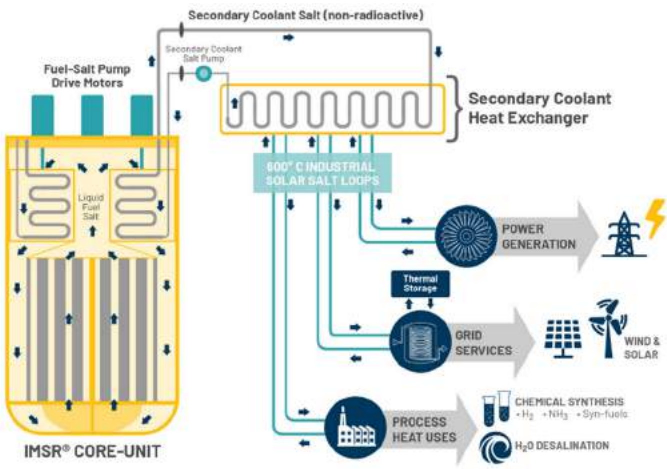
3. Specific Design Features

(a) Design Philosophy

The underlying molten salt technology of the IMSR® design is molten salt reactor technology that was the product of successive and extensive research programs at the Oak Ridge National Laboratory (ORNL) in the 1950's, 1960's and 1970's. During this period, extensive R&D was undertaken to develop molten salt reactor materials, equipment and reactor components. This culminated in the construction and successful operation of a small (less than 10 MW(t)) experimental test reactor, the Molten Salt Reactor Experiment (MSRE). Like the MSRE, the IMSR® employs a molten fluoride fuel salt which also serves as the primary coolant that is circulated between a critical graphite moderated core and primary heat exchangers. However, the IMSR®, although based on the proven molten salt technology of the MSRE, utilizes a unique "integral" reactor plant architecture where all primary pumps and primary heat exchangers are integrated inside a sealed and replaceable reactor vessel. This key innovation together with other proprietary innovations, delivers a reactor of high commercial and industrial value – high reliability and operational utility – unlike the small laboratory scale MSRE test reactor.

(b) Reactor and Core-unit

The Core-unit is manufactured in a controlled factory environment and then brought to the reactor plant site where, following final assembly, is lowered into a surrounding guard vessel located in a below grade reactor silo. The IMSR® primary fuel salt is a thermally stable fluid with excellent coolant heat transfer properties, and high intrinsic radionuclide retention properties. As shown in the adjacent figure, a secondary coolant salt loop, also a fluoride salt (but without fuel), transfers heat from the core-unit fuel salt via primary heat exchangers to a third industrial solar salt loop.



(c) Power Conversion System

The solar salt loop, which is pumped from the nuclear island to a separate building, either supplies a steam generator that generates superheated steam for power generation or is used to drive process heat applications. 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 the hot molten solar salt may be sent directly to a process heat application.

(d) Reactivity Control

Reactor criticality control is assured through negative temperature feedback made possible by the neutronic behaviour of the molten salt fuel in the reactor core. This negative temperature feedback avoids overheating by assuring criticality control, even with loss of all control systems. Molten salt fuel does not degrade by heat or radiation, which gives a high-power limit to the salt fuel. Although shutdown rods are integrated into the IMSR® core-unit, these are for the operational control, and not needed for safety. These shutdown rods will shut down the reactor upon loss of forced circulation and will also insert upon loss of power. A further backup shutdown mechanism is provided with meltable cans, filled with a liquid neutron absorbing material that will permanently shut down the reactor in the unlikely event overheating occurs.

(e) Fission Product Retention

The first barrier for fission product release is the fluoride fuel salt. This fuel salt is chemically stable and binds into the salt with chemical ionic bonds, the large majority of radioactive fission products created during reactor operations. The Core-unit represents the second barrier. The guard vessel containment surrounding the Core-unit is provided as an additional sealed barrier in the extremely unlikely event that the integral Core-unit would experience a failure. Without sources of pressure in the core-unit or in containment, the containment is never challenged by pressure. Overheating of containment is precluded by the balance of heat generation and heat losses to the always operating internal reactor vessel auxiliary cooling system (IRVACS).

(f) 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 any of the in-core fuel during makeup fuelling. All the fuel remains inside the closed IMSR® core-unit during the entire 7-year power operations period of the core-unit. The small volume of additional "makeup" fuel salt is simply accommodated during operation in the upper gas plenum.

Unlike other power reactor systems, the IMSR® core-unit is never opened at the power plant site, either during start-up fuelling, during make-up fuelling, or during switch over to a new core-unit. After ~7 years of power operation, the operating core-unit is shut down and after a cool-down period, the used fuel charge is pumped out to holding tanks located inside the reactor containment.

(g) Cooling System

The IMSR®'s unique cooling system is based on inherent and passive thermodynamic characteristics of the IMSR® – high thermal inertia, large heat capacity and a passive and the continuously operating IRVACS. A large inherent heat capacity is provided by the thermal mass of the fuel salt, reactor vessel metal, and graphite moderator. Furthermore, the reactor vessel is not insulated leading to inherent and continuous parasitic heat loss to the Guard Vessel and ultimately to the IRVACS. Short term Core-unit cooling is assured by the internal natural fluid convective cooling capability of the molten fluoride fuel salt through the natural and passive circulation of fuel salt through the primary heat exchangers. These inherent and passive cooling mechanisms are fully capable of absorbing transient and decay heat generation.

Longer term cooling is provided by heat loss from the uninsulated reactor vessel, to the guard vessel which is cooled by the IRVACS. The Guard Vessel is a closed vessel that envelops the core-unit, providing containment and cooling through its vessel wall. Excessive heating of the core-unit will cause increased heat losses from the core-unit, in turn increasing heat transfer, via thermal radiation, to the guard vessel. The guard vessel in turn is surrounded by the independent, closed-loop inert-gas passive cooling IRVACS annulus. This perpetually functioning IRVACS system transfers heat from the uninsulated guard vessel to its inert gas. The gas in turn circulates via natural convection to radiators in in the reactor auxiliary building where it loses its heat to outside air before returning to the cooling annulus. The IRVACS cooling system operates at atmospheric pressure so it will continue to cool in the event of a leak or damage to the system.

(h) Fuel Characteristics

The IMSR® is a liquid fuel reactor – there are no solid 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 IMSR400 is designed to use a once-through, low enriched uranium (<5% enrichment) fuel cycle, which is viewed as the simplest and most commercially pragmatic option. This fuel salt is diluted with a coolant salt, which consists of other fluorides. Together, this salt eutectic mix forms both fuel and primary coolant. The fuel-coolant salt mix 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.

4. Safety Features

The safety objective with the IMSR® design is to achieve high inherent safety, and a walk-away safe nuclear power plant. No operator action, electricity, or externally-powered mechanical components are needed to assure the primary safety functions of controlling, cooling, and containing.

Fundamental in the IMSR® safety philosophy is the removal of drivers that have potential to push radioactive material out of containment and into the environment. The reactor operates at low pressure, which is a benefit of using a thermally and chemically stable low-volatile fuel-coolant mixture. Additionally, there is no water or steam in the reactor system. This approach eliminates any potential energy sources, both physical and chemical, in the reactor system. The IMSR® further augments this high level of inherent safety with its integrated, pipe-less, fail-safe systems architecture.

IMSR® cooling during abnormal conditions does not depend on depressurizing the reactor or bringing external coolant to the reactor. All required control and heat sink functions are 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, controls, or operator actions for cooling. This is the case in both the short term and the long term. This is achieved as the IMSR® design combines molten salt reactor technology with an integral reactor design surrounded by the guard vessel that is cooled by the continuously operating IRVACS cooling system. The IMSR® design hence provides for the highest levels of inherent safety and promises nuclear safety excellence with high commercial relevance.

5. Plant Performance

The IMSR® core-unit is designed to be simple and safe to replace. This supports a high operational utility factor for the IMSR® power plant and high capital efficiency. The replaceable core-unit ensures the materials' lifetime requirements of all other reactor core components are accommodated by the design; a failure of this condition is often cited as an impediment to immediate commercialization of MSRs. The benefit of the replaceable core-unit is hence a power plant that delivers the combination of high energy output, simplicity and ease of operation, and cost-competitiveness essential for widespread commercial deployment. The IMSR® design promotes reactor-core safety through the intrinsic properties of the fuel salt combined with a strongly negative temperature coefficient of reactivity: as temperature rises, reactivity declines, and vice versa. The IMSR® thus has a substantial load-following capability: a decrease of heat extraction by the steam generator (or by an industrial-heat load) leads to an increased coolant temperature, which lowers reactivity resulting in a reactor power decrease; conversely an increase in heat extraction reduces coolant temperature, which increases reactivity and raises reactor power.

6. Instrumentation and Control Systems

The IMSR® does not depend on electrical or even I&C systems for ultimate safety. The control safety function for reactor power for the IMSR400 is inherently part of the reactor's fundamental physics characteristics and does not require an engineered active safety system to maintain control of power in any design basis accident scenario. A plant investment protection system and a plant control system provide all control functions. Additionally, however, the IMSR400 does include provisions to initiate either of two shutdown mechanisms in case of severe accidents.

The IMSR400 reactor power naturally follows the heat load of the nitrate salt loop (turbine demand, process heat demand, or industrial heat demand). This is, by its inherent reactivity response to changes in fuel/coolant temperature, triggered by temperature changes in the secondary and tertiary coolants. That is, reactivity decreases on fuel salt temperature rise and increases as fuel salt temperature drops. Primary reactivity control is achieved by using the secondary coolant-salt pump to alter the circulation flow rate, which changes the temperature of the fuel salt in the core and thus alters reactivity.

The fuel salt is thermally stable, has a boiling point greatly in excess of the IMSR® operating temperature and is resistant to decomposition; this eliminates concerns about a temperature overshoot on a sudden decrease in heat removal from the reactor coolant. In addition, the passive IRVACS has the capacity to remove all decay heat following a "reactor trip". These inherent characteristics eliminate the need for in-core reactivity control devices for the purposes of shutting down the reactor for safety reasons. In the short term, the functions of the power control system are to enable the reactor to match the power withdrawn by the balance of plant heat load, and to ensure salt temperatures remain within intended limits. In the long term, fuel burn-up is offset by routine manual additions of small amounts of fresh fuel salt.

7. Plant Arrangement

The plant layout is shown in the adjacent figure. An operating IMSR® core-unit is housed in one of two operating silos in the below grade nuclear island.

Prior to core-unit shutdown at end-of-life, a fresh core-unit is installed in the adjacent operating silo in preparation for switchover of the secondary coolant salt loop from the operating, end-of-life core-unit to the new core-unit. After switchover, the spent core-unit remains in its operating silo until it has been de-fuelled to the spent fuel salt storage tanks. The de-fuelled core-unit is then moved with an overhead crane from its operating silo to a long-term storage silo inside the reactor auxiliary building. A third core-unit can then subsequently be installed in the now empty operating silo, ready to begin another 7-year operating cycle once the second operating core-unit unit reaches end-of-life and is shut down.

The secondary (non-radioactive) coolant salt lines transfer heat from the operating core-unit primary heat exchangers to secondary heat exchangers and a third, nitrate salt loop located in the nuclear island. The nitrate salt loop transfers the heat from the nuclear island to either steam generators and re-heaters located in the Balance of Plant or directly to an industrial heat end user which could be located up to 5 km from the nuclear island.

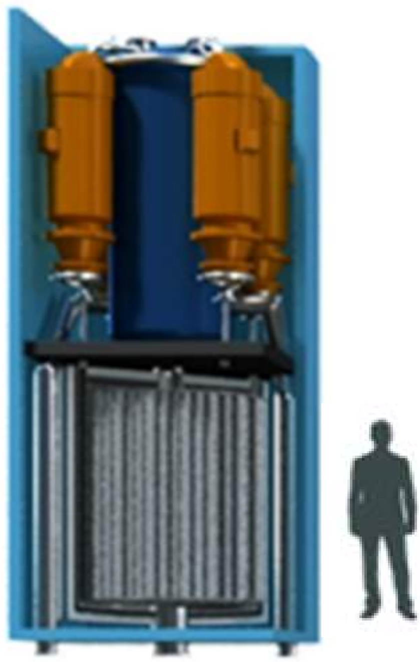
8. Design and Licensing Status

The Canadian Nuclear Safety Commission has completed the first phase of the Canadian nuclear regulator's pre-licensing vendor design review for the IMSR®. This represents the first regulatory opinion by a western nuclear regulator of a commercial reactor power plant design designated as a Generation IV advanced reactor power plant.

9. Development Milestones

2015	Conceptual design completed
2016	Start of basic engineering phase
2017	Completion of CNSC pre-licencing phase-1 vendor design review
Early 2020's	Secure necessary licenses
Early 2020's	Construction planning of a first full-scale IMSR® NPP in Canada

CMSR, (Seaborg Technologies, Denmark)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Seaborg Technologies, Denmark
Reactor type	Molten salt reactor
Coolant/moderator	Proprietary moderator material
Thermal/electrical capacity, MW(t)/MW(e)	250/100-115
Primary circulation	Forced circulation
System pressure (MPa)	1
Core inlet/exit temperatures (°C)	600/700 or 700/900
Fuel salt	Sodium-actinide fluoride (93% Th, 3.5% U 3.5% Pu)
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
Main reactivity control mechanism	Control rods
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	-
RPV height/diameter (m)	2.5/2.1
Distinguishing features	Uniquely compact core; Thermal spectrum wasteburner; Safety-by-physics
Design status	Conceptual

1. Introduction

The Seaborg Technologies compact molten salt reactor (CMSR) is a thermal spectrum, single salt, ultra-compact molten salt reactor (MSR) which can be operated on conventional nuclear fuel, as well as a combination of spent nuclear fuel (SNF) and thorium. It is designed to produce 100 MW(e), or 115 MW(e) with a two-stage turbine, from 250 MW(t). The core outlet temperature is 700°C, but can go as high as 900°C for special use cases such as hydrogen production. Its unrivalled compactness is a result of a novel, patent-pending liquid moderator material. The CMSR is designed around inherent safety features so that no active intervention is required to control and bring the reactor to a safe state.

The fully modularized CMSR is optimized for mass manufacturing. It consists of serially produced modules that are transported to and assembled on site. The ‘core unit module’ contains all the components in regular contact with the fuel salt and moderator. It has a lifetime of 12 years, after which the fuel will be drained, and the module extracted, cooled and returned for recycling in a central facility. The power plant will operate on the same batch of SNF for the 60-year facility lifetime.

The CMSR is in the conceptual design phase with development work focusing on reactor physics, material and corrosion studies, and engineering design. Concurrently, Seaborg is developing next-generation multi-physics codes capable of accurately simulating the dynamic behaviour of MSRs and supporting licensing. Their main source of revenue is from consultancy within neutronics and multi-physics. Additional backing comes from both private investors in addition to European and Danish public funding.

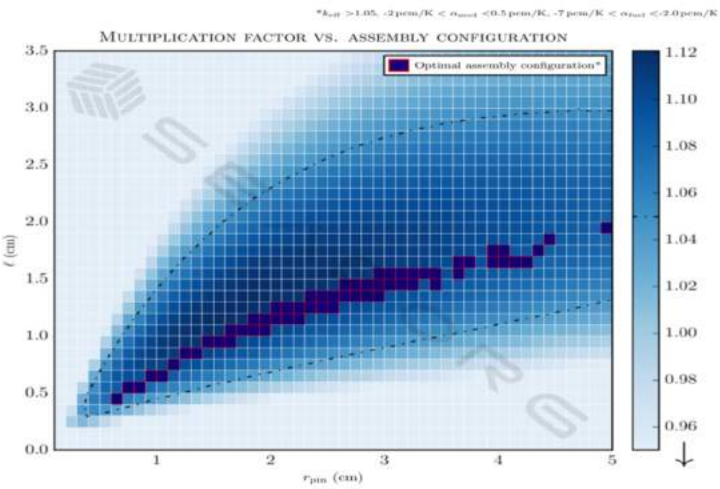
2. Target Application

The CMSR is primarily intended as a power generator, but can also co-produce heat for district heating/cooling or desalination. Due to the high outlet temperature, it is also well-suited as input in the production of a wide range of substances including hydrogen, synthetic fuels, ammonia, cement, and more. Its high burn-up makes it a good candidate for reducing spent nuclear fuel stockpiles. However, with minimal modifications, it can operate on a wide array of different fuels, including conventional, low-enriched uranium.

3. Specific Design Features

(a) Design Philosophy

Seaborg originally set out to address the three main concerns that resulted in the Danish ban on nuclear power, namely safety, sustainability, and proliferation. Today these three central pillars have been complemented with two more; portability and affordability. Together they steer Seaborg's design efforts on a day-to-day basis.



Safety: The reactor is kept in a neutronically optimized configuration so that any perturbation - from human error to major earthquake - automatically reduces power production. Eventually, the reactor shuts down and the fuel cools until it solidifies, unless the situation is actively rectified.

Waste: By combining SNF with thorium, the production of transuranic elements (TRU) is lower than the rate at which they are destroyed. For a single 250 MW(t) module, the net reduction is roughly five tonnes TRU over the 60-year plant lifetime.

Non-proliferation: To ensure the highest proliferation-resistance, the CMSR is designed to be a thermal-spectrum, single-salt reactor with a chemical reprocessing system that is incapable of sorting actinides.

Portability: The CMSR is designed to be deployed in any region of the world, including those that need energy the most. The 35 ton, compact core module and exhaustive modularization enables even road transport by truck.

Affordability: To importantly displace fossil fuels, the CMSR must be cost competitive. By relying on safety-by-physics rather than safety-by-engineering, many of the complex and costly engineered safety systems of conventional nuclear reactors can be exchanged for simple passive safety features.

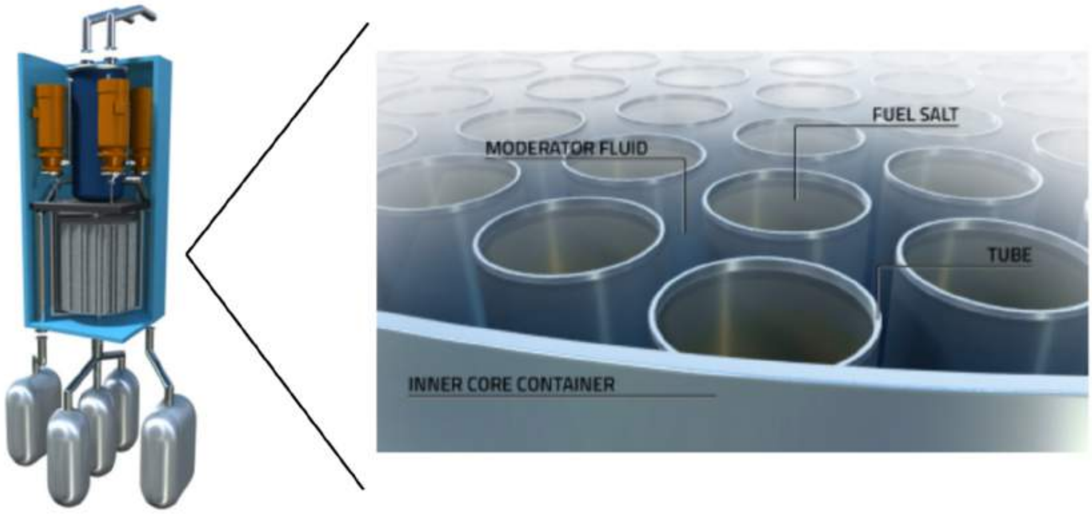
(b) Power Conversion Unit

The CMSR power plant features three flow circuits; the primary fuel salt circuit, a secondary salt circuit, and the tertiary power conversion circuit. The latter can either be a conventional Rankine steam cycle with a single or two-stage turbine that produces 100-115 MW(e), or a direct output for industrial heat applications. The power conversion circuit can also be equipped with co-production capabilities to deliver district heating/cooling or desalinated water in addition to electricity.

(c) Reactor Core

The CMSR core consists of an array of straight fuel pipes that directs fuel salt through the liquid moderator material. It is configured so as to ensure negative void and temperature reactivity coefficients of both the fuel and the moderator. The use of a liquid moderator alleviates concerns related to conventional graphite moderators and their long-term performance under irradiation and in contact with fuel salt, as well as disposal. The reactor operates at 55 kW per litre core, with a power density of around 110 kW per litre fuel. Corrosion and thermal creep of the primary circuit are expected to be the limiting factors for the lifetime of the core module.

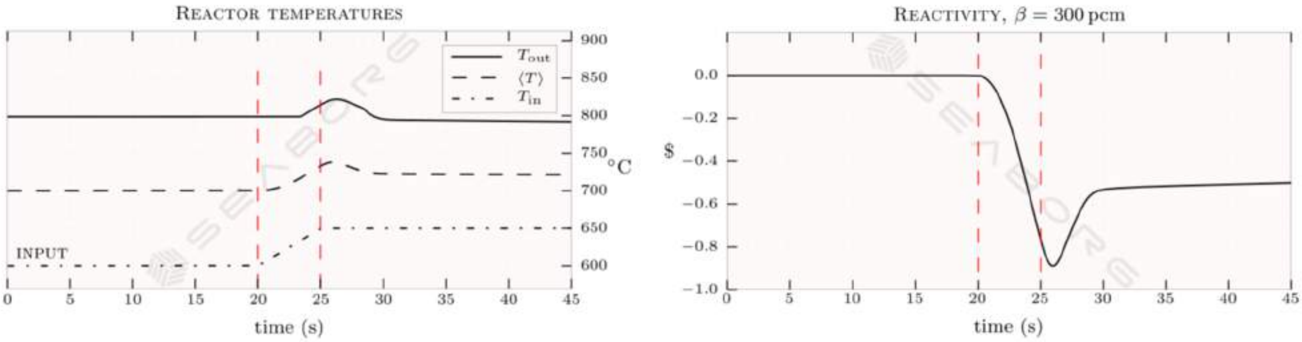
The reactor core will passively shut down if overheated as a result strongly negative fuel temperature coefficients. In case of a continued inability to cool the core, the fuel will eventually drain itself through an actively cooled freeze plug to a passively cooled dump tank.



(d) Reactivity Control

The primary reactivity control mechanism in the CMSR FOAK consists of control and shutdown rods that are inserted into the moderator volume of the reactor core. While the CMSR features negative fuel and moderator temperature coefficients, control rods are favoured for primary reactivity control so as to maintain a constant fuel temperature as far as practically achievable. This, in turn, reduces thermal cycling of the structural materials. The diverse shutdown systems function by injecting neutron poisons into the fuel or moderator. A third, inherent reactivity control mechanism results from the uniformly negative temperature coefficients.

Long-term depletion of reactivity is counteracted with the online refuelling system that feeds fresh SNF into the fuel salt. Moreover, some fission products, notably xenon, are continuously extracted from the fuel salt, immobilized and cooled in order to improve the neutron economy and reduce the source term and decay heat.



(e) Fuel Characteristics

The fuel salt is an eutectic sodium-actinide fluoride mixture with a melting point of about 490°C, although several candidates are being explored. The benefits of this fuel salt include avoiding enriched litium-7 and toxic beryllium, as well as minimizing tritium production and costs. Initially the actinide fraction is 87% thorium and 13% pre-processed SNF (reactor-grade plutonium), but over time the fissile fraction of the plutonium decreases, and so only SNF is added while the thorium fraction decreases.

(f) Reactor Pressure Vessel and Internals

The reactor core sits inside the core vessel that contains the liquid moderator and features penetration for the primary fuel circuit, the control and shutdown rods, and moderator control systems. Since the reactor operates below atmospheric pressure above the fluid surface, and there are no large temperature differentials that could cause flash boiling, the vessel is not required to sustain large pressures. Inside the vessel are the fuel pipes, guide thimbles to ensure rod insert ability under all conditions, and various instrumentation to monitor flows, temperatures, pressures, corrosion, and more.

4. Safety Features

In the CMSR, safety is guaranteed by physics rather than engineering. First, the power output is passively regulated by large negative fuel temperature coefficients. Second, reactor cool ability is ensured by gravity and natural convection. Third, actinides and most of the fission products are chemically retained within the fuel salt, notably including iodine and caesium. All active safety systems are redundant and primarily to ensure asset protection.

(a) Emergency Core Cooling System and Decay Heat Removal System

Residual heat produced in the fuel salt is initially removed through the ordinary heat transfer path and dissipated in the ultimate heat sink. In case of isolation of the primary circuit, or when the residual heat production is sufficiently low, the moderator temperature control system is used to maintain the fuel temperature within design limits.

The emergency heat removal system provides asset protection and passive cooling of the fuel salt under accident conditions. It consists of two separate trains of valves and pipes, one with control valves and one with actively-cooled freeze plugs, that can drain the fuel salt from the primary circuit to passively-cooled dump tanks located below the reactor.

The CMSR does not feature a conventional, injection-based emergency core cooling system.

(b) Containment System

The CMSR is designed so that temperatures approaching the fuel boiling point are unreachable due to negative temperature coefficients, the temperature-dependent ratio between residual heat production and radiative heat transfer, and passively-cooled dump tanks. With the exception of noble gasses, noble metals, and cadmium, fission products are then chemically retained in fluoride salts, including the traditionally hazardous caesium and iodine. The remaining volatile fission products are continuously extracted and immobilized, and are therefore never present in significant amounts in the fuel salt. For these reasons, the source term, even in severe accidents, will be small and pose a minimal off-site risk.

To adhere to existing regulatory paradigms, and to control the atmosphere around the primary circuit, the reactor is designed with additional barriers. This includes a sealed, sub-atmospheric compartment containing the primary circuit, dump tanks, and fuel processing systems.

5. Plant Safety and Operational Performances

The neutronic optimum design philosophy provides outstanding inherent safety features and excellent transient behaviour, including great load following capabilities. By relying on both control rods and temperature coefficients to accommodate load variations, substantial power transients can be performed with only modest thermal cycling. Furthermore, due to the continuous removal of noble gasses and the relatively large fuel inventory, the reactor will not experience xenon poison-outs after reducing power. By attaching the secondary circuit to a thermal salt storage, these load following capabilities can be further enhanced.

6. Instrumentation and Control Systems

The CMSR will feature both conventional and novel I&C systems, including an automatic power control system that synchronizes turbine load variations and control rod movement to minimize thermal cycling. The fuel chemistry control system monitors the chemical state of the fuel salt to predict and control corrosion with online fuel reprocessing. A conventional SCRAM system will be installed with inputs from detectors that monitor the neutron flux, primary circuit flow, temperatures and pressures, as well as the primary pumps and turbine, among other things. The automatized draining system takes input from temperature sensors and actuates at specified set points to protect the integrity of the primary circuit and core vessel.

7. Plant Arrangement

All irradiated components, fuel, and coolant salts are contained in an approximately 8 m deep and 12 m wide underground area, covered with 2 m of concrete during operation. The underground area contains two reactor bays that hold one core unit module each, allowing for installation and commissioning of a fresh module in tandem with the shutdown of the other. The rest of the plant is placed above ground.

8. Design and Licensing Status

The CMSR design is in the conceptual design phase, with development focusing on reactor physics, engineering and systems design, and materials and corrosion studies. Licensing is actively considered in the development work, and will begin when the conceptual design is matured and key hypotheses are validated.

9. Development Milestones

2014	Thermal wasteburner concept coined.
2015	Thermal wasteburner neutronic benchmark.
2016	First major design revision: Ultra-compact MSR. Pre-conceptual design initiated.
2017	Core patents filed. Danish funding granted. Private investment secured.
2018	Conceptual design phase initiated. EU funding granted. First experiments. Seaborg becomes largest reactor design start-up in Europe with 17 employees.

Copenhagen Atomics Waste Burner (Copenhagen Atomics, Denmark)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Copenhagen Atomics, Denmark
Reactor type	Molten salt reactor
Coolant	Fuel salt
Moderator	Heavy water
Thermal/electrical capacity, MW(t)/MW(e)	50/20
Primary circulation	Forced circulation
Initial fissile inventory	Transuranic
Fuel salt	$LiF\text{-}ThF_4$
Online reprocessing	Vacuum spraying
Power conversion process	Open air Brayton cycle
Main reactivity control mechanism	Heavy water level adjustment
Approach to engineered safety systems	Passive
Design life (years)	3-5
Plant footprint (m^2)	2500
Operation	Firmware, no human operators.
Design status	Conceptual

1. Introduction

The Copenhagen Atomics waste burner version 0.2.3 is a 50 MW(t) heavy water moderated, single fluid, fluoride salt based, thermal spectrum, molten salt reactor. Copenhagen Atomics Waste Burner 2.0 and above versions are expected to be breeder and converter type designs, breeding more fissile material than consumed while converting fissile transuranic from existing uranium cycle waste to start a thorium based cycle. The core, fission product extraction and separation systems, dump tank, primary heat exchanger, pumps, valves, and compressors are all contained in a leak tight 40-foot shipping container surrounded by a shielding blanket of frozen thorium salt, see the figure below.

2. Target Application

The following applications are foreseen:

- Addon units at existing nuclear sites, coupled with a spent nuclear fuel reprocessing unit.
- Ship or barge based power systems.
- Biofuel production and desalination plants.
- Baseload power in Asia and Africa.

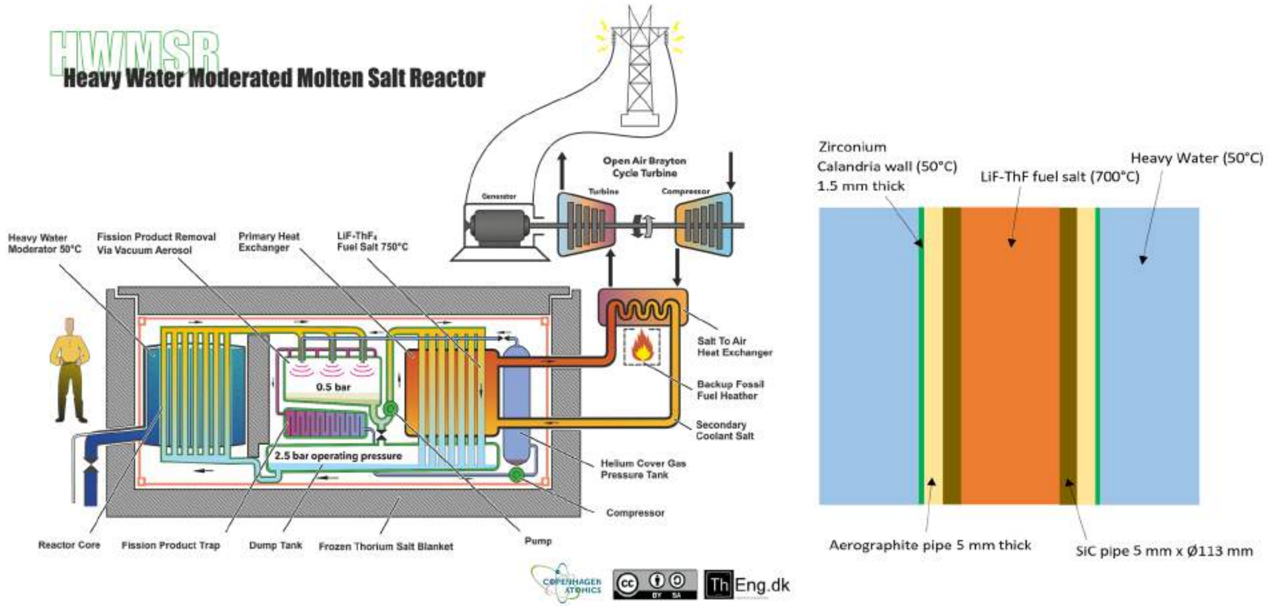
The reactor design effort is focussed on a small modular thorium breeder thermal spectrum fluoride molten salt reactor, made to fit inside a 40-foot shipping container and with an initial fissile inventory made up of spent nuclear fuel transuranic.

3. Specific Design Features

(a) Design Philosophy

The Copenhagen Atomics Waste Burner, more than being a paper reactor design is really a thought experiment used by the Copenhagen Atomics team to explore new ideas and play around with concepts, leading to constant revisions and forking. These thought experiments have led to a number of design pillars:

i. Fast Technology Cycles and Rapid Prototyping:
Copenhagen Atomics doesn't believe that we should try to build the first generation of molten salt reactors to last for 40, 50, 60 years, or more. Neither do we think that we should try to make deaths per unit energy produced much lower than other competing energy forms. We're instead focus on designs that are able to scale rapidly to meet the world's vastly growing demand for energy in all forms. Thus, making the first generation of molten salt reactors to only last a couple of years allows for experience gained to be implemented much sooner and making for a much faster development turnaround. This is necessary for building reliable, durable, and rugged machines, optimized for breeding and utilization of fissile inventory. Only then do we see long operational lifetimes as appropriate.



Schematic drawing of a heavy water moderated molten salt reactor (left) and possible core fuel channel insulation configuration (right).

ii. 40' Shipping Container:
Shipping containers are central to Copenhagen Atomics long-term goal of achieving mass production and deployment, and rapid development cycles. For utilizing the economies of scale, it is a necessity that molten salt reactors be constructed on an assembly line and are easily shipped anywhere in the world. This is why the Copenhagen Atomics Waste Burner is made to fit inside of a gas tight stainless steel 40' shipping container that serves as the secondary containment.

iii. Walkaway Safe and Prime Minister Safe:
Each Copenhagen Atomics Waste Burner unit will operate fully autonomously, with experience and learnings being synchronized through firmware updates to the whole reactor fleet. This, along with cleaver design will ensure that the reactors are not only walkway safe but also safe from any authority with malicious intent. The term *"prime minister safe"* refers to the philosophy that no one, even not people of high authority, can prevent the reactor from shutting down when needed or operate the reactor in manner deemed unsafe.

(b) Power Conversion Unit

An open-air Brayton cycle turbine, connected to a secondary coolant salt system, is envisaged.

(c) Reactor Core and Fuel Characteristics

The Copenhagen Atomics Waste Burner 0.2.3 is a heavy water moderated, single fluid, fluoride salt based, thermal spectrum, molten salt reactor. The fuel salt is $LiF\text{-}ThF_4$ with an initial fissile inventory of spent nuclear fuel transuranic. The fuel salt is also the heat transport medium, and the fuel-salt flow rate also determines the core thermal power. The fuel-salt flow through thermally insulated pipes in the core, surrounded by the heavy water moderator. The figure shows a possible configuration leading to manageable heat loss in the zeroth-order kilowatt range per pipe, not including the considerable heating from thermalization of neutrons and other particle interactions.

Heavy water is used to achieve a thermal spectrum. While most of the transuranic suffers from poor reproduction factor in the thermal spectrum, the bred uranium 233 does not. It is expected that the Copenhagen Atomics Waste Burner 0.2.3 will start off with a breeding ratio below one but increase well above one as the initial fissile inventory of transuranic is converted to a fissile inventory of uranium 233. Future version of the Copenhagen Atomics Waste Burner might employ a dual fluid design if protactinium separation is deemed too cumbersome.

(d) Heavy Water Moderator

Unpressurized heavy water is chosen as the moderator due to its superior neutron economy and increased power to fissile inventory over graphite moderation, its lack of long-term degradation under irradiation, and ease of reactivity control. It should be stressed that the heavy water is not used for heat extraction, and that the water is thermally insulated from the molten fuel salt (as shown in the fuel channel configuration figure) and continuously circulated and cooled to around 50□.

Besides radiolysis of the heavy water, which will be managed with a passive recombiner, one of the main reasons why Copenhagen Atomics is pursuing a heavy water moderated design is that it suffers no long-term degradation like swelling, cracking, or ingress of fuel salt and fission products, unlike graphite, whose integrity will probably end up setting the lifetime of the units. Thus, the issue of moderator decommissioning is also circumvented, since we're now in the paradigm of infinite moderator and fuel lifetime with the core vessel being the only expendable part.

(e) Reactivity Control

The liquid moderator allows core reactivity control through a simple water level adjustment. E.g. gradual build-up of fissile inventory from breeding can be offset through a simple lowering of the water level. Similar for compensating the startup temperature defect.

(f) Reactor Pressure Vessel and Internals

The small reactor core tank also operates at low pressures and have a short design lifetime. Copenhagen Atomics is also pursuing the possibility of automated resmelting of whole 40 foot reactor units to new units. So that the salt, heavy water, and metals are all reused, vastly decreasing the amount of material that has to be decommissioned by orders of magnitude.

(g) Online Fission Product and Tritium Separation

The Copenhagen Atomics Waste Burner 0.2.3 doesn't employ any online wet chemistry but rather aims to achieve online removal of up to half of the generated fission products using a closed hot helium recirculation loop, passing through a vacuum spraying chamber, gas centrifuges, fission product and tritium traps, and high temperature compressors.

Vacuum spraying, developed by Copenhagen Atomics, is an advancement of the traditional helium bubbling method, which allows for a much larger fraction of the fission products to be extracted from the fuel salt than is possible with helium bubbling. Both methods rely on evaporation of noble gases and boiling of volatile fission product compounds for the extraction of fission products.

While far from all fission products are neither noble gasses nor volatile at the operating temperature and pressure of the primary salt, a majority of the fission products decay through multiple elements that are. Thus, most fission products can technically be extracted through short lived intermediate states along their decay chain. This also entails spraying the entire salt stream just after it leaves the core and before passing through the heat exchanger.

The enhanced fission product extraction of vacuum spraying is achieved by increasing the surface to volume ratio and hence increasing the evaporation rate of noble gasses and volatile compounds, and decreasing the ambient pressure to a fraction of atmospheric pressure here by increasing the volatility of the majority of the fission products. It's this combination of spraying tiny droplets into a low-pressure region, promptly upon leaving the reactor core, that allows for vacuum spraying technology to theoretically extract up to half of the fission products without the need for any online wet chemical reprocessing. Wet chemistry batch reprocessing to remove the remaining fission products at the end of each units expected 3-5 year operational lifetime will then be carried out at reprocessing plants.

The Copenhagen Atomics vacuum spraying system also serves the purpose of extracting the tritium before it has time to migrate through the heat exchanger or other metal surfaces, coupled with a trap for reacting the tritium into a stable form. Separation of fission products are achieved through gas centrifuges, membranes, filters, catalysers, and reagents. Similar approaches are being considered for possible online separation of Protactinium.

(h) Lithium Breeding

Since it's unlikely that highly enriched lithium 7 (>99.995%) will be available for the early deployment of molten salt reactors, Copenhagen Atomics expects to start the first couple generations of Copenhagen Atomics Waste Burners on market available medium enriched lithium 7 (~99.9%) and breeding highly enriched lithium 7 in the fuel salt through parasitic neutron losses to lithium 6.

4. Safety Features

(a) Reactivity Control by the Heavy Water Moderator

The heavy water moderator is continuously and passively drained from the core at a high rate and passively cooled before being actively pumped back into the core. This is to secure passive shutdown of the reactor in events such as total power loss where the moderator would simply passively drain from the core within seconds, removing the need for a fuel salt drain plug, control rods, or neutron poison injection systems. A similar concept is employed for passive decay heat removal in case of a shutdown, where the fuel salt is always draining into the dump tank while continuously being pumped back out (see the schematic drawing of the reactor systems). The dump tank is then passively cooled above a certain level leading to minimal losses during operation.

(b) Emergency Core Cooling System and Decay Heat Removal System

The design does not need a freeze plug (as used in many other designs to drain the core for sub-criticality and decay heat removal) since the dump tank is integrated in the primary salt loop. On power loss the fuel-salt will drain to the dump tank where passive decay heat removal is achieved by means of heat transfer via long oil-filled pipes installed into the surrounding earth. This design reduces the possibility of sabotage. Fission product traps are also passively cooled in the same manner.

(c) Containment System

The containment function is supported by the fuel-salt properties (to contain most fission products), the primary cycle boundary and the 40-foot container (with its thorium salt blanket). In addition, fission products and tritium is actively removed (as explained above) thus reducing the potential source term from the fuel-salt.

5. Plant Safety and Operational Performances

Since the heavy water is unpressurized and held below 50□, there's no chance of a superheated steam explosion. Even if the salt and water were somehow to contact and in an unlikely event lead to a Coulomb explosion and the rapid unscheduled disassembly of the core pipes as the worst possible scenario, such an event is easily contained by just the secondary containment alone.

6. Plant Arrangement

The reactor is installed in a 40-foot container, surrounded in a frozen thorium salt blanket, and can be installed underground inside a reactor building, which functions as the third barrier against radioactive release.

7. Current Developments, Design and Licensing Status

The design is still in conceptual stage. Several forks of the Copenhagen Atomics Waste Burner have or are still exploring novel design ideas, either through laboratory scale experiments or purely academically, and maybe merged in the future:

- Molten lithium hydroxide (LiOH) or lithium deuteroxide (LiOD) moderated and cooled with the fuel salt only being recirculated for fission product removal.
- Pure graphite pebble bed moderated fuel salt with the purpose of being able to do online replacement of the moderator.
- Synthetic diamond moderated fuel salt, either configured as a bed or a slurry.
- Helium moderated and cooled molten salt with helium contained in cooling channels surrounded by the molten fuel salt.
- Helium moderated and cooled molten salt mist or froth.

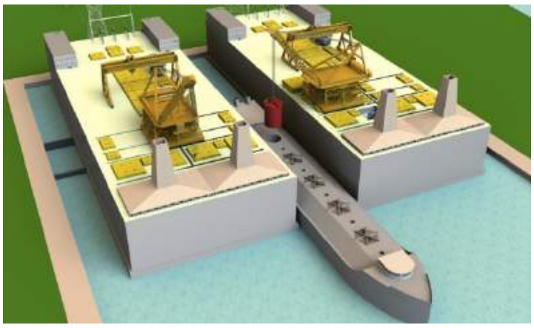
Copenhagen Atomics approach towards achieving this fast technology cycle and rapid prototyping of molten salt reactors is not by developing fully fletched paper reactor designs but rather to focus on the chemistry, measurement technology, simulation and control software, and components that makes up the reactor. Gaining experience with actual molten salts is invaluable and constantly shapes the design process.

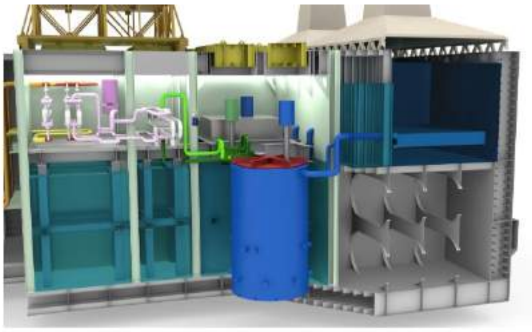
To this end, Copenhagen Atomics employs regular stainless steels and does as much development as possible in common commercially available non-radioactive materials and salts. The Copenhagen Atomics Loop is a forced circulation molten salt loop that's constructed from 316L stainless steel and fits on top of a steel EUR-pallet for transport. It is with these loops that we test everything from materials, perform corrosion studies, and develop LIBS (laser induced breakdown spectroscopy) interfaces for remote measurement of the isotopic composition of the salt, detection of corrosion products, redox potential, etc.

8. Development Milestones

2015	Copenhagen Atomics was founded in 2015 by a group of passionate engineers and scientists meeting up since 2013 and based on an open source model, where results and findings are shared with the thorium molten salt reactor community.
2015	First simulations of neutron economy and online fission product removal.
2016	First static molten salt test.
2017	First pressure driven circulation molten salt loop.
2018	First pumped circulation molten salt loop.

ThorCon (ThorCon International, Indonesia)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ThorCon International, first deployment in Indonesia
Reactor type	Thermal molten salt reactor
Coolant/moderator	NaF, BeF_2 molten salt/graphite
Thermal/electrical capacity, MW(t)/MW(e)	557/250 (per module)
Primary circulation	Forced circulation
System pressure (MPa)	1.2
Core inlet/exit temperatures (^oC)	565/704
Fuel salt	12% heavy metal in NaBe salt.
Fuel enrichment (%)	19.7
Fuel burnup (GWd/ton)	220 GWd/tonne LEU20
Fuel cycle (months)	96
Main reactivity control mechanism	Negative temp coeff, salt flow rate; fissile/fertile additions.
Approach to engineered safety systems	Intrinsic, passive, using natural circulation, water evaporation.
Design life (years)	80
Plant footprint (m^2)	20,000 for 500 MW(e)
RPV height/diameter (m)	12/8
Seismic design	SSE 1.0 pga, OBE > coal plant
Distinguishing features	Low cost, full passive safety, short construction time
Design status	Complete basic design

1. Introduction

ThorCon is a molten salt fission reactor. 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. The ThorCon reactor operates at garden hose pressures using normal pipe thicknesses and easily automated, ship-style steel plate construction methods. The top picture shows two hull-mounted 500 MW(e) ThorConIsle power plants. A CanShip is changing the Can containing the reactor vessel and radioactive primary loop. Decay heat cooling towers are in the foreground. The yellow rectangles are hatches for access by gantry cranes. The middle cutaway graphic shows the cooling towers, fission island, heat exchangers, steam turbine-generator, and switchgear. Bottom graphic shows basement water used as a third, backup decay heat sink.

2. Target Application

The first planned application of ThorCon reactors is to generate electric power in developing nations with fragile grids, so ThorCon is capable of demand discontinuities and black start without grid power. Capital cost and generated electricity costs are critical in these markets. ThorCon is cheaper than coal and deployable as rapidly. Indonesia completed a ThorCon pre-feasibility study in 2017.

3. Specific Design Features

(a) Design Philosophy

i. ThorCon is Walkaway Safe

If the reactor overheats for any reason, it will automatically shut itself down, drain the fuel from the primary loop, and passively remove the decay heat. There is no need for any operator intervention; the operators cannot prevent the draining and cooling. ThorCon has three gas tight barriers between the fuel salt and the environment. In a primary loop rupture, there is no coolant phase change and no dispersal energy. Spilled fuel merely flows to the drain tank where it is passively cooled. The most troublesome fission products, including I-131, Sr-90, and Cs-137 are chemically bound to the salt. They will end up in the drain tank as well.

ii. ThorCon is Ready to Go

The ThorCon design needs no new technology. ThorCon is a scale-up of the successful molten salt reactor experiment (MSRE). A full-scale dual 250 MW(e) ThorConIsle prototype can be operating under test within four years. This prototype will be subject to the failures and problems that the designers claim the plant can handle. As soon as the prototype passes these tests, commercial production can begin.

iii. ThorCon is Rapidly Deployable

The entire ThorConIsle plant 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 assembled into a hull containing the complete power plant, to be towed to a customer site and firmly settled in 5-10 m of water. A 500 MW(e) power station will require fewer than 100 blocks. Compared to traditional on-site nuclear power plant construction, this improves productivity, quality control, and build time. A single large reactor yard can turn out thirty 500 MW(e) ThorConIsles per year. Alternatively, the blocks can be barged to the site and assembled in an excavation, creating an underground fission island linked to a standard above-grade turbine hall. ThorCon is much more than a power plant; it is a system for building power plants.

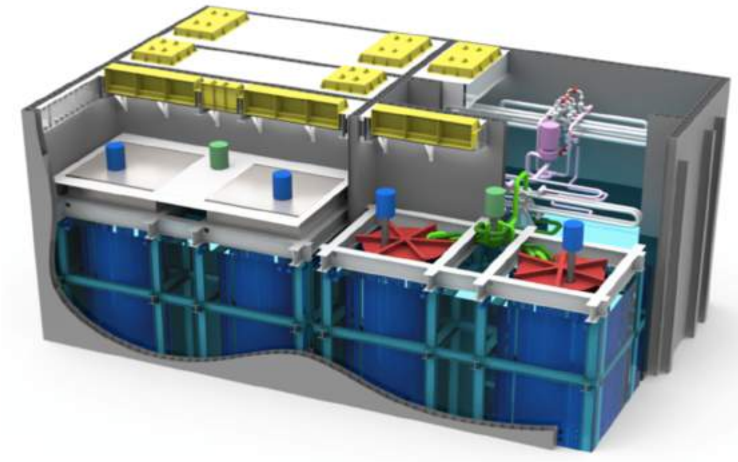
iv. ThorCon is Fixable

No complex repairs will be attempted on site. Hatches and cranes permit everything in the fission island to be replaced with little interruption in power output. The primary loop is totally contained within a Can. Every four years the Can 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. A fission power plant following such a change-out strategy can in principle operate indefinitely. Decommissioning should be little more than removing the Cans without replacing them, then towing the hull away.

v. ThorCon is Cheaper than Coal

ThorCon requires far fewer resources than a coal plant. Assuming efficient, evidence-based regulation, ThorCon will produce clean, reliable, carbon-free electricity at less than the cost of coal.

(b) Nuclear Steam Supply System



ThorCon is divided into 250 MW(e) power modules. Each module contains two replaceable reactors in sealed Cans. The Cans, depicted in red, sit in silos. Just one of the Cans of each module produces power at a time, while the other is in cooldown mode. After four years the cooled Can is replaced with a fresh Can, the fuel salt transferred to it, and used Can starts its 4-year cool down.

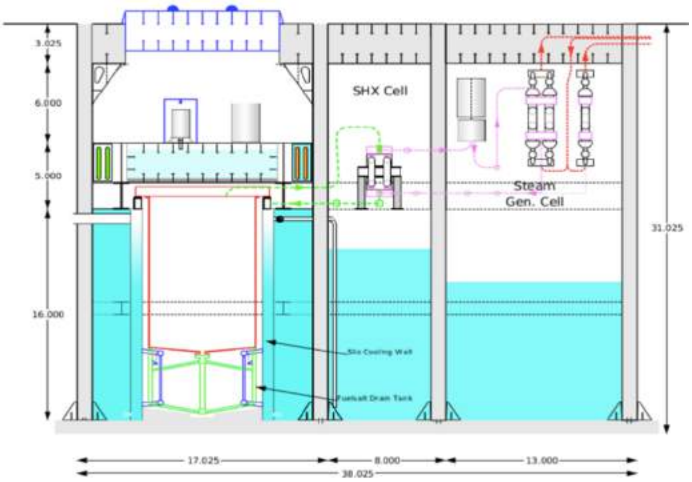
The fuel salt is a mixture of sodium, beryllium, uranium and thorium fluorides at 704°C. The (red) Can contains the (orange) reactor called the Pot. Hidden behind the (blue) header tank, the primary loop pump pushes the fuel salt at 3000 kg/s through the (red) piping down through the (blue) primary loop heat exchanger (PHX).

The PHX transfers heat to secondary salt in (green) piping. The fuel salt at 565°C is then piped into the Pot. There the graphite moderator slows neutrons, which fission uranium in the fuel salt as it rises through the Pot, heating the salt. Neutron absorption also converts some fertile thorium and U-238 to fissile fuel.

The Pot pressure is 3 bar gage at the maximum stress point. The outlet temperature of 704°C results in an overall plant efficiency of 46% with net electric power output of 250 MW per Can. Consumption of fissile uranium is 112 kg per year. The Can is 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 (green) 32-segment fuel salt drain tank (FDT). In the bottom of the Can is a freeze valve. At normal operating temperatures, the fuel salt in the freeze valve is kept frozen by cold flowing helium creating a plug. If the Can overheats for any reason, the helium flow stops, the plug will thaw, and the fuel salt will drain to the FDT. This drain is totally passive. There is nothing an operator can do to prevent it. Fission in the Pot stops as the drain begins. The 32 segmented vertical drain tanks have no moderator, and re-criticality is impossible in all events, including flooding.

An important feature of ThorCon is the silo cold wall (blue). The silo wall is made up of two concentric steel cylinders, shown in blue. The annulus between these two cylinders is filled with water. The top 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 to flow into the bottom of the annulus. The Can is cooled by thermal radiation to the silo cold 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.



The silo cold wall also cools the FDT. The drain tank is a circle of (green) vertical tanks to contain hot fuel salt drained from the Pot. This 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 diagram 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 turn transfers its heat to a supercritical steam loop, shown in red.

(c) Reactor Core

The reactor core is inside the pot. The core is 90% filled with hexagonal graphite logs which moderate neutron energies. The core is 5 m diameter, and 5.7 m high.

(d) Reactivity Control

The primary reactivity control is temperature and fuel salt flow rate. Makeup fissile uranium fuel salt additions increase reactivity slowly. Adding fertile thorium fuel salt decreases reactivity.

(e) Fuel Characteristics

The fuel salt is NaF-BeF_2-ThF_4-UF_4 76/12/9.5/2.5 where the uranium is 19.7% enriched. As fissile is consumed more fissile U-233 and Pu-239 is generated, but not enough to replace the fuel burned. The reactor has no excess reactivity, no burnable poisons, no poison control rods. Makeup fuel must be added daily.

(f) Reactor Pressure Vessel

The ThorCon Pot reactor vessel is never under high pressure. Since no high pressure is present that can act as a driving force to disperse radioactive content into the environment, ThorCon's reactor pressure vessel does not have the central safety importance that it does in a LWR.

4. Safety Features

(a) Approach to Engineered Safety Systems

Passive reconfiguration into positive shutdown and passive, infinite grace time decay heat removal with no requirement for electricity or operator actions to initiate or continue any safety systems.

The ThorCon negative temperature coefficient provides passive temperature stability. The large margin between the operating temperature of 700ºC and the fuel salt boiling temperature of 1430ºC exceeds any possible temperature excursions, so radioactive material can never be vaporized. If the temperature of the fuel salt rises much above the operating level, physical principles decrease reactivity and shut fission down. Additionally, if high temperature somehow persists, the freeze valve will thaw and drain the fuel salt from the primary loop to the drain tank, which radiates heat to the cold wall to passively remove the decay heat. No operator intervention is needed at any time. No valves need be realigned by operators nor control systems. In fact, operators can do nothing to prevent the shutdown, drain, and cooling. The decay heat is transferred to the external pond which has sufficient water for 145 days cooling. After evaporation exposes the condenser at the pond bottom, natural air flow suffices for cooling indefinitely. If the pond cooling line is lost, there is enough water in the basements to handle 1.5 years of decay heat.

(b) Release Resistance

ThorCon has three gas tight barriers between the fuel salt and the environment. ThorCon reactor operates at near-ambient pressure. In the event of a primary loop rupture, there is little 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 radioactive material barriers are somehow breached, few of these salt-soluble fission products could disperse.

(c) Spent Fuel Salt

ThorCon uses an eight-year fuel salt processing cycle, after which the used salt is drained to the fuel drain tank cooled by the cold wall. Within 4 weeks the liquid fuel salt is pumped to a shipping cask in its own silo alongside the Can, within the power module, within the hull. The silo is passively cooled by basement water. The fuel salt cooling in its shipping cask is as well protected as the fuel salt being burned. After 4 years the cooled fuel cask is transferred to the visiting CanShip and shipped to a fuel salt handling facility for future uranium re-enrichment and fuel salt recycling.

(d) Four Loop Separation of Steam and Fuel Salt

ThorCon employs four loops to transfer heat from the reactor to the steam turbine - the fuel salt loop, the secondary salt loop, the solar salt loop, and the steam loop. Oxygen in the solar salt captures any created tritium that may have penetrated hot heat exchangers. A pressure limiting standpipe in the solar salt loop ensures that a rupture in the high-pressure steam generator just vents harmlessly into the Steam Generating Cell.

5. Plant Safety and Operational Performances

Load following is accomplished by changing primary loop pump speed while keeping the temperatures relatively constant. ThorCon meets EU utility requirements for light water reactors: 5% full power per minute operating at 50% to 100% power. Since the off-gases are continuously removed xenon poisoning and power oscillations are not an issue. No neutron poisons are used in the control of the reactor, reducing fuel consumption.

6. Instrumentation and Control Systems

Instrumentation and control systems are not safety-critical for ThorCon. Argonne National Lab is adapting its isotopic concentration sensors to monitor ThorCon fuel salt components. Commercial instrumentation and sensors will record and report the condition of power generation. Statistical process control will track trend lines and detect incipient failures such as bearing wear. A common control centre for all modules in a power plant minimizes staffing requirements. A central engineering facility will monitor conditions at all plants, allowing fleet wide analysis, detect unusual activity, and provide expert advice for any plant experiencing unusual conditions.

7. Plant Arrangement

The control room is shared by all 250 MW(e) power modules in a power plant. Commonly two power modules will drive a single 500 MW(e) turbine/generator. This allows using competitively-priced, efficient supercritical steam turbine-generators, while also remaining suitable for smaller 250 MW(e) power plants.

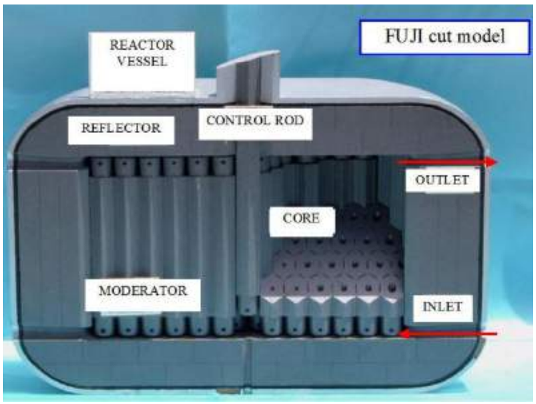
8. Design and Licensing Status

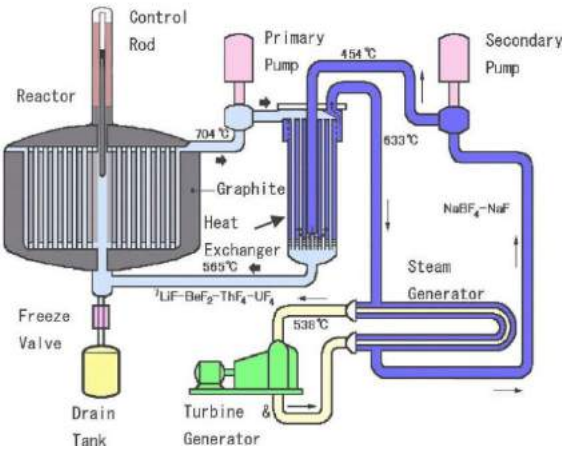
Basic design is complete. Some detailed designs are being discussed with specialty component suppliers. License discussions have started with the Indonesian regulator, Bapeten.

9. Development Milestones

2011	Conceptual design development
2018	Complete basic design documented
2020	Pre-fission testing starts
2022	ThorConIsle prototype complete; fission testing starts
2024	Indonesia regulator issues type license; commercial production begins

FUJI (International Thorium Molten-Salt Forum, Japan)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ITMSF, Japan
Reactor type	Molten salt reactor
Coolant/moderator	Molten fluoride/graphite
Thermal/electrical capacity, MW(t)/MW(e)	450/200
Primary circulation	Forced circulation
System pressure (MPa)	0.5
Core inlet/exit temperatures (°C)	565/704
Fuel salt	Molten salt with Th and U
Number of fuel assemblies	--
Fuel enrichment (%)	2.0 (0.24%^{233}U + 12.0%Th). Pu or LEU can be used.
Fuel burnup (GWd/ton)	--
Fuel cycle (months)	Continues operation is possible
Main reactivity control mechanism	Control rod, or pump speed, or fuel concentration
Approach to engineered safety systems	Passive
Design life (years)	30
Plant footprint (m^2)	<5000 (R.B. + SG.B. + TG.B.)
RPV height/diameter (m)	5.4/5.34
Seismic design	Same as LWRs
Distinguishing features	High safety, high economic performance, contribution to non-proliferation, and fuel cycle flexibility
Design status	3 experimental MSRs were built. Detailed design is not started

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.

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 MW(e) to 1,000 MW(e). But, a latest and typical design (FUJI-U3) is 200 MW(e), which can be categorized as small-sized reactors with modular designs (SMR). The thermal output of FUJI-U3 is 450 MW(t) and thus a 44% thermal efficiency can be attained. In addition, the simple core structure and high fuel efficiency should facilitate a favourable economic performance.

Molten fuel salt can contain thorium (Th) as fertile material and ^{233}U as fissile material, and 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. Specific Design Features

(a) 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.

(b) 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 the figure above. Only one loop is shown but a loop can be redundant depending on a plant size or a need for flexibility.

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.

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.

(c) Reactor Core

A reactor vessel is cylindrical in shape. The core structure is made of 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 be required as in any power plant.

(d) 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.

(e) Fuel Characteristics

The molten fuel salt is a liquid form of fluoride (LiF-BeF_2) with ThF_4 and a small amount of $^{233}UF_4$. A typical composition is LiF-BeF_2-ThF_4-$^{233}UF_4$ (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.

(f) Reactor Pressure 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.

(g) 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.

(h) 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.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

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.

In order to protect against a freeze accident in a molten fuel salt loop, a high temperature containment is equipped with heaters.

(b) Decay Heat Removal System

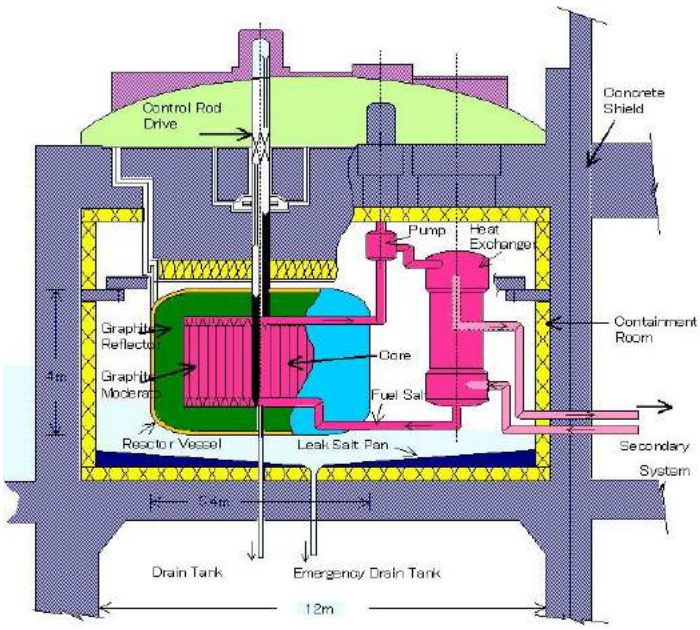
In normal shutdown condition, decay heat is transferred to a secondary loop and a steam-line loop, and disposed to the ultimate heat sink (seawater for example). If all pumps in a primary or secondary loop stop, fuel salt is drained to a drain tank through a freeze valve. Decay heat at the drain tank 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.

(c) 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.

(d) 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. In order to avoid a freeze 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.



5. 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.

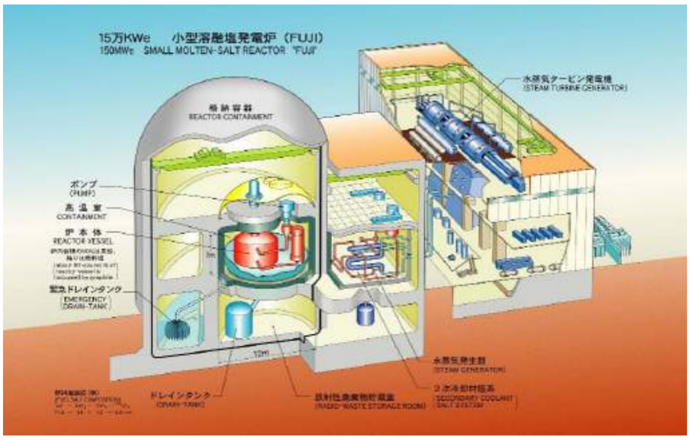
In normal daily operation, the power level can be controlled by core flow or by core temperature control. That is, load following is easily performed without using control rods. Control rods are withdrawn in normal operation, and are inserted by gravity in case of emergency shutdown.

6. 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.

7. 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.



(a) 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.

(b) 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.

(c) Balance of Plant

i. 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.

ii. 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, it can be shut down and cooled without electricity.

8. Design and Licensing Status

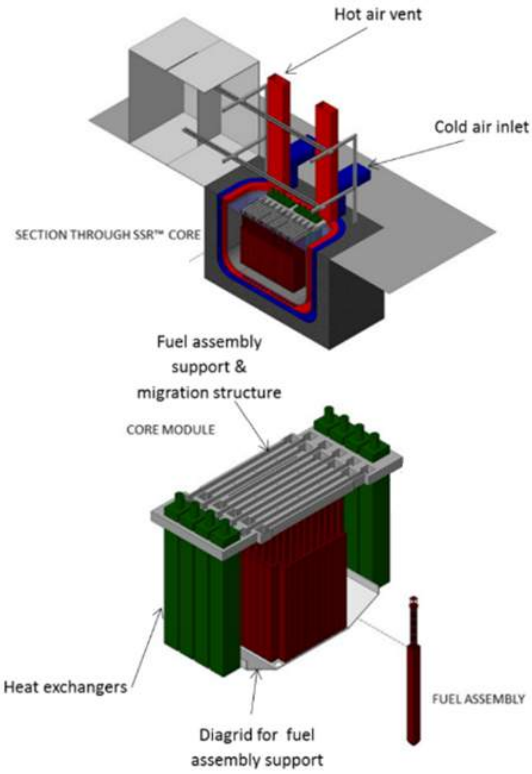
Preliminary designs for various applications have been completed [1]. Three experimental MSRs were constructed, and one of them was operated for 4 years without severe problems. Although the detailed design is not yet started, safety criteria and guidelines for MSR licensing are proposed with numerical results for major accident analysis.

9. 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-fuelled plant (FUJI-Pu),
Recent	The latest SMR plant (FUJI-U3),

[1] Yoshioka, R., Kinoshita, M. "Liquid Fuel, Thermal Neutron Spectrum Reactors", Chapter-11 of the book "*Molten Salt Reactor and Thorium Energy*", Elsevier Inc., USA, 2017

Stable Salt Reactor - Wasteburner (Moltex Energy, UK)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Moltex Energy, United Kingdom
Reactor type	Static fuelled molten salt fast reactor
Coolant/moderator	No moderator. Coolant is molten salt $ZrF_4/KF/NaF$
Thermal/electrical capacity, MW(t)/MW(e)	750/300 continuous as baseload 750/900 as 8 hour peaking plant
Primary circulation	Forced circulation
System pressure (MPa)	Atmospheric
Core inlet/exit temperatures (°C)	500/630
Fuel salt	Molten salt fuel within vented fuel tubes in a conventional style fuel assembly
Number of fuel assemblies	200
Fuel enrichment (%)	Reactor grade plutonium
Fuel burnup (GWd/ton)	>300
Fuel cycle (months)	~150
Main reactivity control mechanism	Boron carbide control assemblies
Approach to engineered safety systems	Hierarchy is eliminate hazard, then passive engineering then active engineering
Design life (years)	60
Plant footprint (m^2)	22500
Tank height/length/width (m)	5/6/5/
Seismic design	Yes
Distinguishing features	Molten salt fuel in conventional fuel assemblies; thermal energy storage to allow operation as peaking plant; very low cost conversion of spent LWR fuel into SSR fuel
Design status	Concept, Canadian vendor design review in progress

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. The reactor is fuelled with very low purity, reactor grade plutonium recycled from stocks of spent uranium oxide fuel and produced by a low cost process called WATSS (Waste to Stable Salt). The reactor outputs its heat as a stream of molten nitrate salts which can be stored in large volume at low cost making the reactor a low-cost peaking power plant rather than being restricted to baseload operation. This same system permits the entire steam cycle to be identical to the low-cost steam system in CCGT power stations and for it to be operated completely independently of the nuclear plant and not subject to nuclear regulations.

2. Target Application

The SSR Wasteburner is designed for countries with significant stocks of spent nuclear fuel. The reactor burns the full higher actinide component of that fuel leaving a relatively short lived, fission product only, waste stream. The fuel cost is expected to be negative, net of the reduced liability cost for disposal of the original spent fuel. It is designed to be capable of economically efficient electrical power peaking but with the reactor itself running at constant power. It therefore fills the need in national power systems for a low carbon complement to intermittent renewable energy sources.

3. Specific Design Features

(a) 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.
- 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.
- Economic competitiveness to have a capital cost below $1000 per kW electrical output so that it can produce electricity cheaper than fossil fuels without relying on subsidy.
- Modular design with the reactor assembled from road transportable, factory produced modules creating a single reactor unit of 300MW(e).
- Fuel assembly form is compatible with IAEA Safeguards procedures used in reactors today.

(b) Power Conversion Unit

A molten salt to steam boiler is proposed to generate steam. The turbine currently being specified for a 300MW(e) plant is the Siemens SST5-5000. The air-cooled generator is Siemens SGen5-1200A-2P 118-55. Steam output temperature is 550˚C. There will either be a larger turbine of up to 900 MW(e) installed, to be coupled with the solar salt storage or multiple smaller turbines. This will depend on the local electricity grid needs and economics. 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.

(c) 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 with adjacent rows travelling in opposite directions to provide a counterflow pattern.

The assemblies are constructed from HT9 steel, similar in materials and structures to sodium fast reactor fuel assemblies. The tubes are 2 m long with 1.6 m of fuel and 400mm of gas space at the top which has a venting mechanism 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). Fission product gases have time to decay to more stable isotopes before entering the pool of coolant salt.

(d) 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 by step by step addition of the final fuel assemblies with the coolant at operating temperature

Shutdown is achieved with boron control mechanisms which drop into the core on demand or on power failure. A second level of shutdown control is provided by the high negative reactivity coefficient of the core which causes the core to shutdown simply due to rising temperature.

(e) Fuel System

The fuel is 60% NaCl/~20% reactor grade $PuCl_3$ /20% UCl_3 and lanthanide trichlorides. It is redox stabilised to render it non- corrosive to steel by inclusion of metal zirconium in each tube which maintains the salt in a strongly reducing state incapable of dissolving chromium from steel.

(f) 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. Defence in depth is provided by sequential layers of containment – the fuel tube wall, the primary coolant which is miscible with the fuel salt, the tank wall, the lined concrete reactor pit.

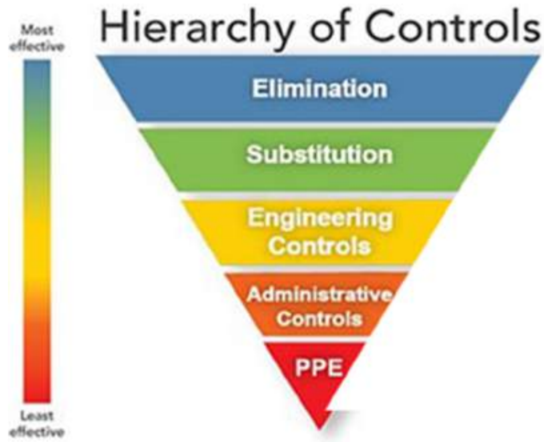
(g) Reactor Coolant System

The reactor is a pool type reactor so the primary coolant fills the tank. The coolant salt is 42% ZrF_4 /48%KF/10%NaF melting point 385°C

The secondary coolant is identical and transfers heat from the primary coolant via heat exchangers immersed in the pool of primary coolant. The secondary coolant passes heat to a tertiary coolant via heat exchangers outside the reactor containment. The tertiary coolant is nitrate "solar" salt and is stored in large steel tanks outside the nuclear island. These tanks of solar salt are drawn on by steam generators to power turbines. The turbogenerator sets have capacity up to three times that of the reactor so that the power plant can output up to 900 MW(e) for 8 hours a day rather than a baseload output of 300 MW(e).

4. Safety Features

The design philosophy adopted is to follow the internationally accepted principle of the risk mitigation pyramid shown right. The focus is to eliminate hazards wherever possible and only to rely on engineering or administrative controls when that cannot be achieved.



(a) Hazards Eliminated or Radically Reduced

The molten salts used in the SSR are chemically stable with minimal reactions with air or water. Redox control of fuel and coolant salts prevents corrosion of the fuel clad 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. The use of molten salt coolant eliminates the need for high pressures in the nuclear island.

(b) 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 indefinitely in this accident scenario.

(c) 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 with no power requirement.

(d) Containment System

There are no significant internal pressures. The primary containment is the tube wall, secondary is the coolant salt, which absorbs fission products in the event of tube failure. 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 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 ~300 mm 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.

5. 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. The presence of multiple steam generators and turbines will allow steam side maintenance to be carried out independently of reactor operation. Since the reactor is continuously fuelled, an exceptionally high capacity factor >95% is anticipated.

6. Instrumentation and Control Systems

Primary reactivity control will be by the reactivity coefficient of the coolant and fuel. 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 and replaced remotely by visual or mechanical means.

7. Plant Arrangement

The functional separation of the nuclear island from the steam generation and turbine system by the large "solar salt" stores entirely eliminates any role in nuclear safety from the steam system. In safety terms, the functioning of the steam system is no more important to reactor safety than is the presence of demand on the power grid with a conventional reactor.

Decay heat removal is handled by a combination of a small reservoir of solar salt on the regulated site and the passive heat transfer to atmosphere by ducts around the reactor tank. Neither the large solar salt stores nor the turbine system have any role.

The regulated nuclear site is therefore rather small, with the large solar salt tanks and steam turbine systems outside the regulated area.

8. Design and Licensing Status

A complete conceptual design has been completed and is currently progressing through the Canadian Nuclear Safety Commission Vendor Design Review.

9. Development Milestones

2014	UK patent granted for use of unpumped molten salt fuel in any reactor
	Independent capital cost estimate complete
2015	Pre- conceptual design complete and key claims validated by the UK's National Nuclear Laboratories
	Conceptual design completed and CNSC VDR commenced
2017	Master patent on static fuelled molten salt reactors granted in major geographies. Several other patents progressing through PCT process.
2018	Moltex successful in UK government Advanced Modular Reactor competition

Stable Salt Reactor – Thermal Spectrum (Moltex Energy, UK)

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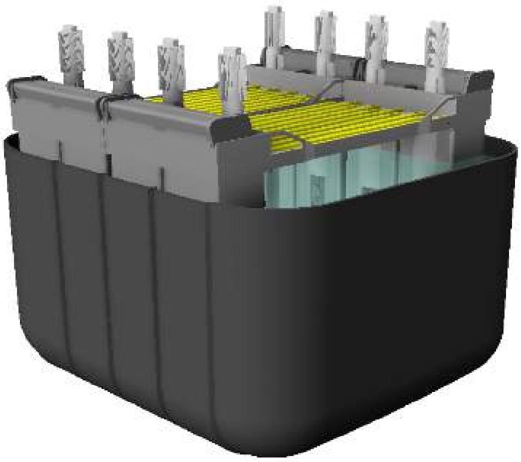


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Moltex Energy, United Kingdom
Reactor type	Static fuelled molten salt thermal reactor (SSR-U) with optional thorium breeding blanket (SSR-Th)
Coolant/moderator	Graphite moderator. Coolant/breeding blanket ThF_4 /NaF
Thermal/electrical capacity, MW(t)/MW(e)	750/300 baseload 750/900 as peaking plant
Primary circulation	Natural convection for fuel forced circulation for coolant
System pressure (MPa)	Atmospheric
Core inlet/exit temperatures (^oC)	650/750
Fuel salt	Molten salt fuel within vented fuel tubes in a conventional style fuel assembly
Number of fuel assemblies	200
Fuel enrichment (%)	~5% enriched U_{235}
Fuel burnup (GWd/ton)	~70
Fuel cycle (months)	~24
Main reactivity control mechanism	Operational control via temperature /reactivity feedback and rate of refuelling. Shutdown is by boron shutdown assemblies.
Approach to engineered safety systems	Hierarchy is first eliminate hazards, then seek passive engineering mitigation of remaining hazards then active engineering mitigation.
Design life (years)	60
Plant footprint (m^2)	22500
Tank height/length/width (m)	6/12/6
Seismic design	Yes
Distinguishing features	Molten salt fuel in conventional fuel assemblies thermal energy storage to allow operation as peaking plant. continuous refuelling thorium breeding
Design status	Pre-concept

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. The reactor comes in two basic variants, the fast spectrum SSR-W (wasteburner) and the thermal spectrum SSR-U which has the option of adding a thorium breeding blanket replacing the zirconium based primary coolant. It is the latter variant that is described in this section.

The reactor is fuelled with low enriched U_{235} initially but is refuelled with ^{233}U in the form of ^{233}U enriched ^{238}U in the variant with thorium breeding. At no time is either pure ^{233}U or ^{233}Pa separated from excess ^{238}U. It thus avoids the significant safeguards issues associated with such materials.

The reactor outputs its heat as a stream of molten nitrate salts which can be stored in large volume at low cost making the reactor a low-cost peaking power plant rather than being restricted to baseload operation. This same system permits the entire steam cycle to be identical to the low-cost steam system in CCGT power stations and for it to be operated completely independently of the nuclear plant and not subject to nuclear regulations. Outside of the small nuclear island the SSR-U/Th and the SSR-W are essentially identical and within the nuclear island they share substantial common components. The SSR-U/Th has a substantially larger core however reflecting the presence of large volumes of moderating graphite.

2. Target Application

The SSR-U/Th is designed for any country with access to low enriched uranium fuel. It is designed to be capable of economically efficient electrical power peaking but with the reactor itself running at constant power. It therefore fills the need in national power systems for a low carbon complement to intermittent renewable energy sources.

3. Specific Design Features

(a) 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.
- 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.
- Economic competitiveness to have a capital cost below $1500 per kW electrical output so that it can produce electricity cheaper than fossil fuels without relying on subsidy.
- Modular design with the reactor assembled from road transportable, factory produced modules creating a single reactor unit of up to 2500 MW(t) output.
- Fuel assembly form is compatible with IAEA Safeguards procedures used in reactors today.

(b) Power Conversion Unit

Three coolant loops are used to take the heat from the fuel pins to a molten salt to steam generator. The turbogenerator sets driven by these boilers will depend on the local electricity grid needs and economics. All coolant loops are operated with forced circulation by centrifugal pumps during normal operation. The system also has a natural circulation capability in emergency conditions.

(c) 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 with adjacent rows travelling in opposite directions to provide a counter-flow pattern.

The major part of each fuel assembly is graphite which serves as moderator. The assemblies are constructed from stainless steel with the preferred alloy being 20/20 Nb as used in the UK AGR fuel assemblies. The tubes are 3 m long with 2.5 m of fuel and 500 mm of gas space at the top which has a venting mechanism to allow some gaseous fission products to be released into the coolant salt (radiologically significant fission products are retained in a non-gaseous form in the fuel salt). Short lived xenon isotopes have time to decay to caesium before entering the pool of coolant salt.

(d) Reactivity Control

The reactor is brought to initial criticality by addition of fuel assemblies to the core. It is then raised to operating power and temperature by adding further assemblies. During operation falls in reactivity due to burn up and accumulation of fission products are compensated via the refuelling frequency and by adjustment of the coolant inlet temperature.

Shutdown is achieved with boron control mechanisms which drop into the core on demand or on power failure. These mechanisms are mechanically diverse and under diverse control systems. An emergency shutdown system is available whereby a boron salt poison is dropped into the coolant pool.

(e) Fuel System

The fuel is a 50/50 mixture of UF_3 and UF_4 diluted in an NaF/RbF salt. It is redox stabilised by a proprietary method to minimise steel corrosion.

(f) 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 ~1 m blast resistant concrete wall which also acts as a biological shield. Defence in depth is provided by sequential layers of containment – the fuel tube wall, the primary coolant which is miscible with the fuel salt, the tank wall, the lined concrete reactor pit.

(g) Reactor Coolant System

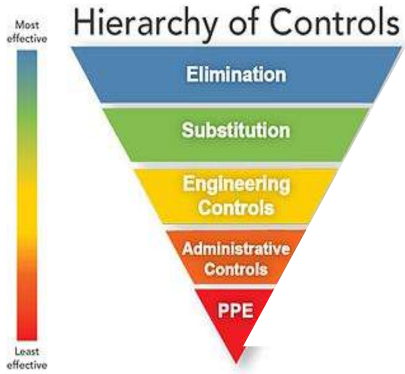
The reactor is a pool type reactor so the primary coolant fills the tank. The coolant salt is 22% ThF_4 /78% NaF melting point 618°C

The coolant circulates through a molten metal/molten salt exchanger containing molten uranium/thorium/ bismuth alloy. This extracts U_{233} bred in the coolant into the alloy, exchanging with thorium so that the coolant blanket thorium content is continually replenished. The ^{233}U is instantly denatured by mixing into the ^{238}U in the alloy. The entire exchanger unit is removable during operation of the reactor as a single unit which can be IAEA sealed prior to transfer to a fuel production centre.

The secondary coolant is 42% ZrF_4 /10% NaF/48% KF melting point 385ºC. It transfers heat from the primary coolant via heat exchangers immersed in the pool of primary coolant. The secondary coolant passes heat to a tertiary coolant via heat exchangers outside the reactor containment. The tertiary coolant is nitrate "solar" salt and is stored in large steel tanks outside the nuclear island. These tanks of solar salt are drawn on by steam generators to power turbines. The turbogenerator sets have capacity up to three times that of the reactor so that the power plant can output up to 900 MW(e) for 8 hours a day rather than a baseload output of 300 MW(e).

4. Safety Features

The design philosophy adopted is to follow the internationally accepted principle of the risk mitigation pyramid shown right. The focus is to eliminate hazards wherever possible and only to rely on engineering or administrative controls when that cannot be achieved.



(a) Hazards Eliminated or Radically Reduced

The molten salts used in the SSR are chemically stable with minimal reactions with air or water. Redox control of fuel and coolant salts prevents corrosion of the fuel clad 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. The use of molten salt coolant eliminates the need for high pressures in the nuclear island.

(b) 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 indefinitely in this accident scenario.

(c) 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 with no power requirement.

(d) Containment System

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 ~1 m concrete wall which serves as the biological shield. The concrete above and below grade is designed to be blast resistant. 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 ~300 mm 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.

5. 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.

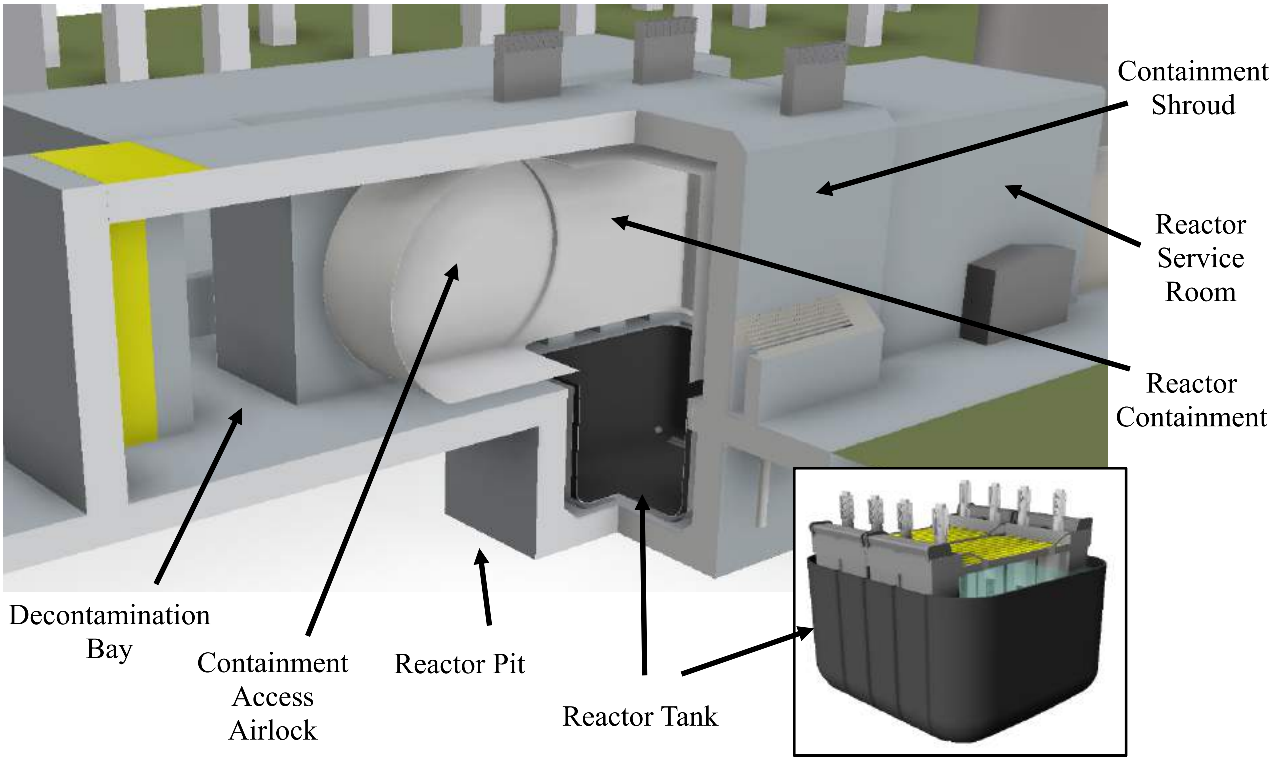
The ramp rates of the plant will be driven by the steam side, not the nuclear side. The presence of multiple steam generators and turbines will allow steam side maintenance to be carried out independently of reactor operation. Since the reactor is continuously fuelled, an exceptionally high capacity factor ~95% is anticipated.

6. Instrumentation and Control Systems

Primary reactivity control will be by the reactivity coefficient of the coolant and fuel. 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 and replaced remotely by visual or mechanical means.

7. Plant Arrangement

The functional separation of the nuclear island from the steam generation and turbine system by the large "solar salt" stores entirely eliminates any role in nuclear safety from the steam system. In safety terms, the functioning of the steam system is exactly as important to reactor safety as is the presence of demand on the power grid with a conventional reactor.



Decay heat removal is handled by a combination of a small reservoir of solar salt on the regulated site and the passive heat transfer to atmosphere by ducts around the reactor tank. Neither the large solar salt stores nor the turbine system have any role.

The regulated nuclear site is therefore quite small, with the large solar salt tanks and steam turbine systems outside the regulated area.

8. Design and Licensing Status

The design is at pre-concept level and the intention is to commence licensing a few years after the licensing of the SSR-Wasteburner.

9. Development Milestones

2014	UK patent granted for use of unpumped molten salt fuel in any reactor
2017	Pre- conceptual design complete
2018	Moltex successful in UK government Advanced Modular Reactor competition with submission including the thermal spectrum SSR

Liquid Fluoride Thorium Reactor (Flibe Energy, USA)

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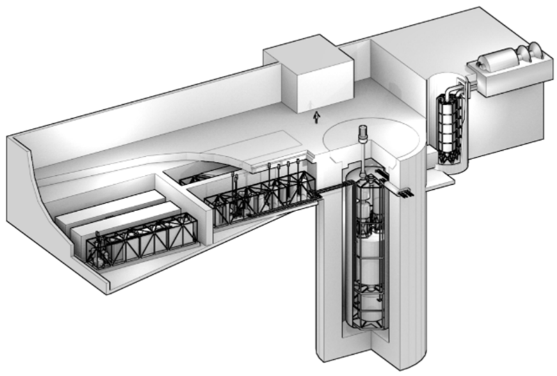


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Flibe Energy, Inc., USA
Reactor type	Molten salt reactor
Coolant/moderator	LiF-BeF_2-UF_4 fuel salt /Graphite
Thermal/electrical capacity, MW(t)/MW(e)	600/250
Primary circulation	Forced circulation
System pressure (MPa)	Ambient
Core inlet/exit temperatures (ºC)	500/650
Fuel salt	LiF-BeF_2-UF_4
Fuel enrichment (%)	Not applicable, uses uranium-233 fuel derived from Th
Fuel cycle (months)	Continuous refueling from ^{233}U produced in blanket
Main reactivity control mechanism	Negative temperature coefficient; control rod insertion
Approach to engineered safety systems	Passive
Distinguishing features	Complete consumption of thorium resource for energy generation
Design status	Concept

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.

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.

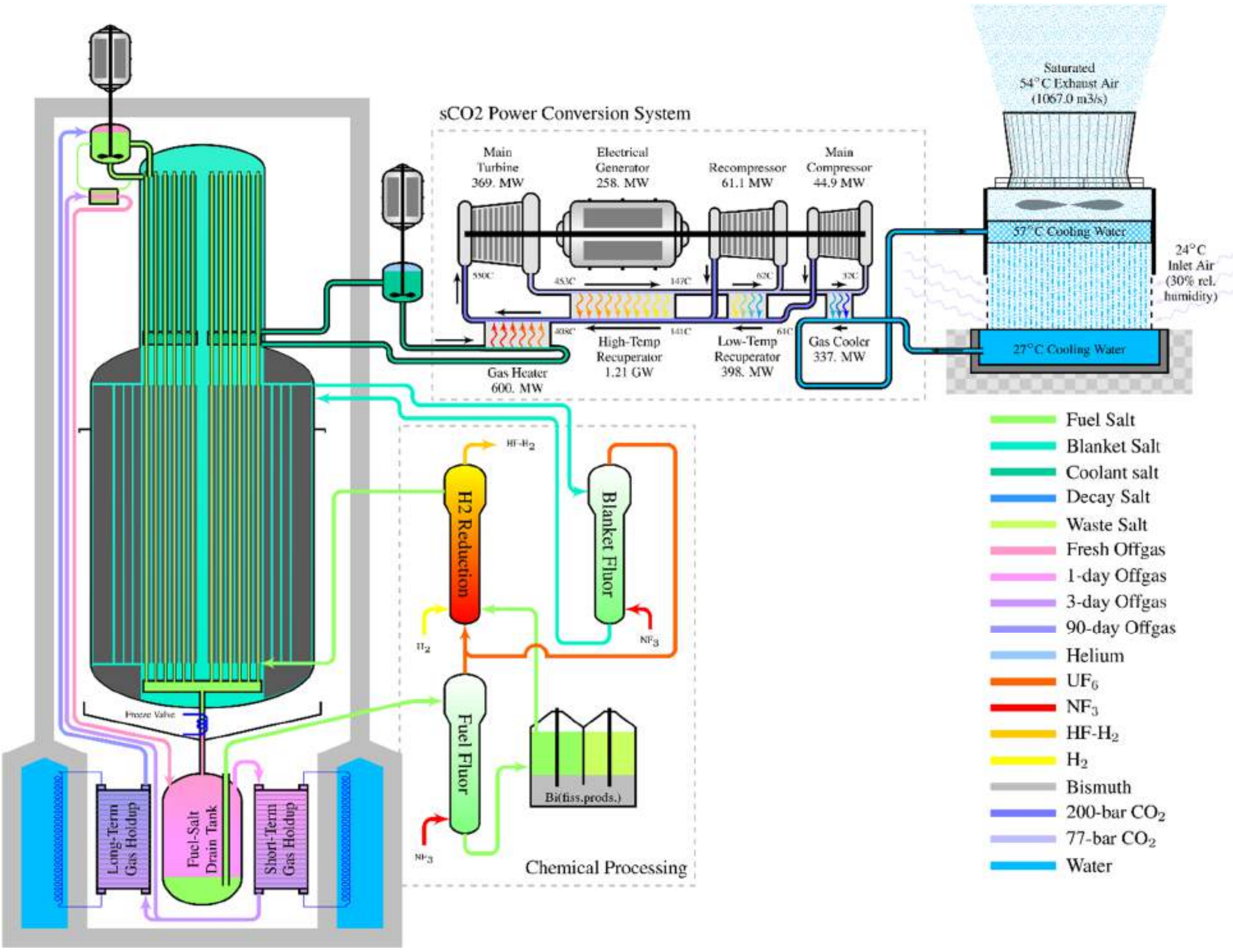
2. Target Application

Develop a power-generating nuclear reactor that will produce electrical energy at low cost by efficiently consuming thorium.

3. Specific Design Features

(a) Design Philosophy

The objective of the liquid-fluoride thorium reactor (LFTR) design proposed by Flibe Energy 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 can utilize the energy content of the thorium at a very high efficiency, and to a point where the Earth's thorium resources practically becomes unlimited. The main design principles 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.



(b) Nuclear Steam Supply System

The nuclear heat supply and power conversion system is included in the simplified flow diagram above. It includes the reactor and primary loop, intermediate loop, power conversion system, and external cooling system. The individual systems are described in more details below followed by other system design descriptions.

(c) 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 need to be replaceable since they are 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.

(d) 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 centre of the core and provide finer control over reactivity levels. Other possible designs are also considered.

(e) 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.

(f) 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.

(g) 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 transfers its heat to the coolant salt. The primary pump provides the necessary forced circulation. The primary loop system includes the primary pump, the PHX (integrated in the reactor vessel), the bubble injection system, and the fuel salt drain tank and its associated external cooling system.

(h) 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 use of a coolant salt in this loop leads to a compact PHX and also reduces the fuel salt inventory, and thus the amount of fissile material needed for a given power rating.

(i) 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.

(j) 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.

4. Safety Features

(a) 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.

(b) 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.

(c) 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.

5. Plant Arrangement

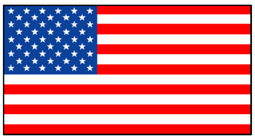
The reactor cavity or silo is below grade and contains the primary circuit.

6. Design and Licensing Status

The design is in an early stage of development and licensing activities have not yet been undertaken.

7. Development Milestones

October 2015	EPRI-funded study of LFTR design published
July 2018	DOE award announcement for advanced fluorination development work



Mk1 PB-FHR (UC Berkeley, USA)

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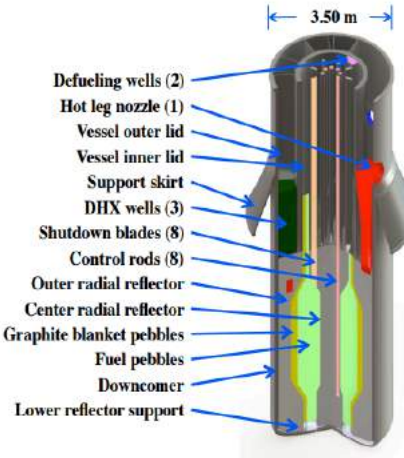
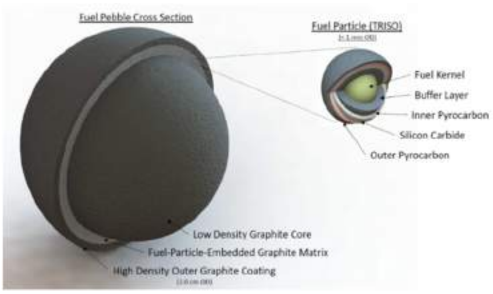




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	University of California, Berkeley, USA
Reactor type	Fluoride-salt-cooled high temperature reactor (FHR)
Coolant/moderator	Li_2BeF_4 / graphite
Thermal/electrical capacity, MW(t)/MW(e)	236/100
Primary circulation	Forced circulation
System pressure (MPa)	0.3
Core inlet/exit temperatures (°C)	600/700
Fuel type	TRISO particles in graphite pebble matrix / pebble bed
Number of fuel pebbles	470000
Fuel enrichment (%)	19.9
Fuel burnup (GWd/ton)	180
Fuel core residence time (months)	2.1, average of 8 passes to achieve full burn up
Main reactivity control mechanism	Negative temperature coefficient; control rod insertion
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (MW(e)/acre)	16.1
RPV height/diameter (m)	12/3.5
Distinguishing features	Large fuel and coolant thermal margin to damage, high temperature operation, NACC with peaking, passive safety
Design status	Pre-conceptual design

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 ($^7Li_2BeF_4$). 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.

2. Target Application

The Mk1 PB-FHR 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. 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. Specific Design Features

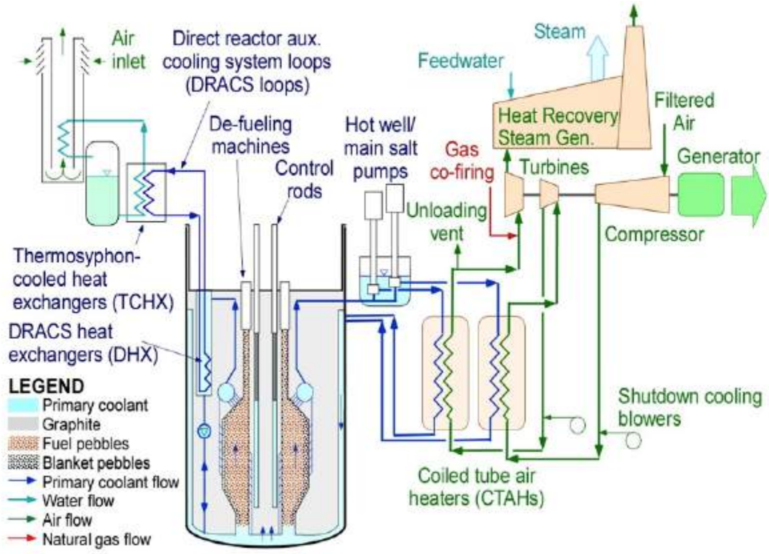
(a) Design Philosophy

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 (when using enriched ^{7}Li), and good neutron moderation capability.

(b) Power Conversion Unit

The 236 MW(t) 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 MW(e) under base-load operation, and with natural-gas co-firing to rapidly boost the net power output to 242 MW(e) 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.



(c) Reactor Core

The core design incorporates an annular pebble-bed core geometry composed of a homogeneous mix of fuel pebbles adjacent to the centre 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.

The centre 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 centre graphite reflector internals need to be replaced periodically due to radiation damage. Its size is much shorter and smaller than centre reflectors proposed for helium-cooled pebble-bed reactors, making it simpler to design for replacement and for seismic qualification.

(d) 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.

(e) Fuel Characteristics

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 MW(e)-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.

(f) 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.

(g) 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.

4. Safety Features

(a) 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.

(b) 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 thermosyphons 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 defence in depth in assuring adequate core heat removal.

(c) 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.

(d) 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.

5. Plant Safety and Operational Performances

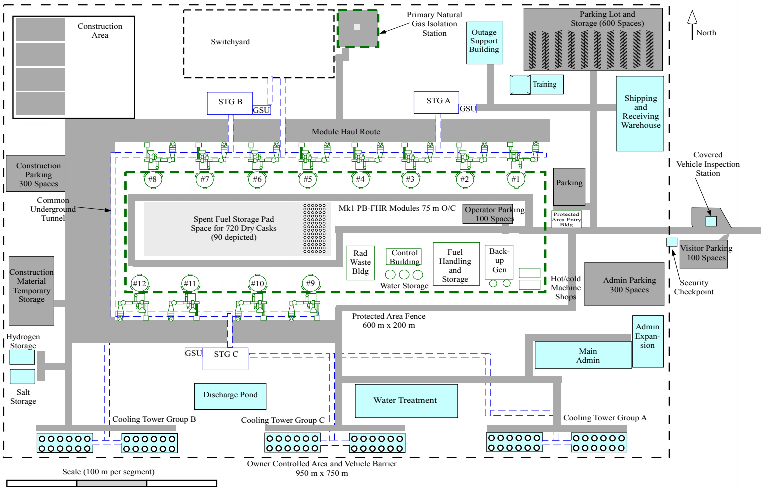
Due to the high thermal efficiency of the NACC system, the steam-bottoming condenser requires only 40% of the cooling water supply that is required for a conventional LWR, for each MWh of base-load generation. As with conventional natural-gas combined cycle (NGCC) plants, this makes the efficiency penalty 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.

6. 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.

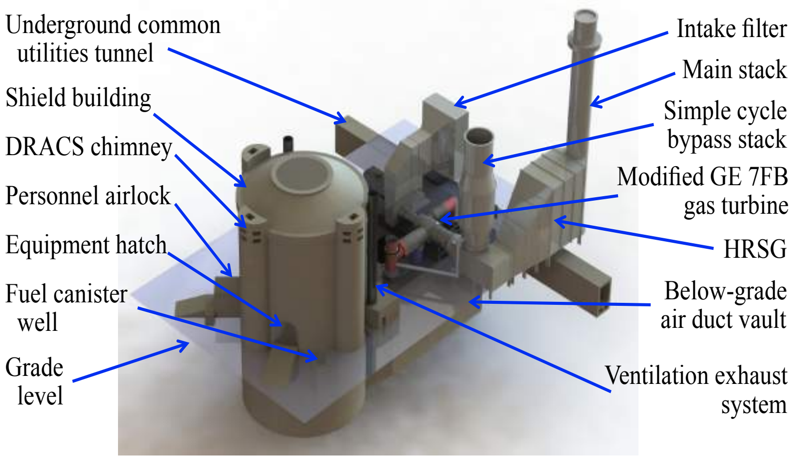
Except during startup and low-power conditions, the PB-FHR operates with constant core inlet and outlet temperatures. Load-following capability is made possible by air bypass flow to respond to rapid load-change transients and turbine inlet temperature control (by bypassing air around the CTAHs) for 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%.

7. Plant Arrangement



The figure on the right presents a notional 180-acre site arrangement for a 12-unit Mk1 PB-FHR power plant capable of producing 1,200 MW(e) base load and 2,900 MW(e) peak power output. Due to the much smaller cooling requirements they do not need to be sited near bodies of water. Population centres 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.

(a) Reactor Building



The Mk1 reactor building and NACC system arrangements supports a multi-module plant configuration, shown in the figure on the right, 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.).

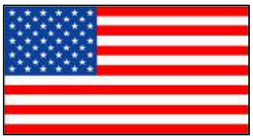
The Mk1 reactor building is partially embedded below grade, with the reactor deck located slightly above grade, shortening the air duct lengths and the depth of the air-duct vault. The baseline Mk1 reactor building design uses a cylindrical shield building fabricated from steel-plate/concrete composite (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-MW(e) AP1000, so the Mk1 shield building volume is 2.2 times greater than the AP1000, per MW(e) baseload.

8. 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.

9. Development Milestones

2018	Pebble bed FHR technology, with significant similarities to the Mk1 PB-FHR, is being developed by Kairos Power



Molten Chloride Salt Fast Reactor (Elysium Industries, USA and Canada)

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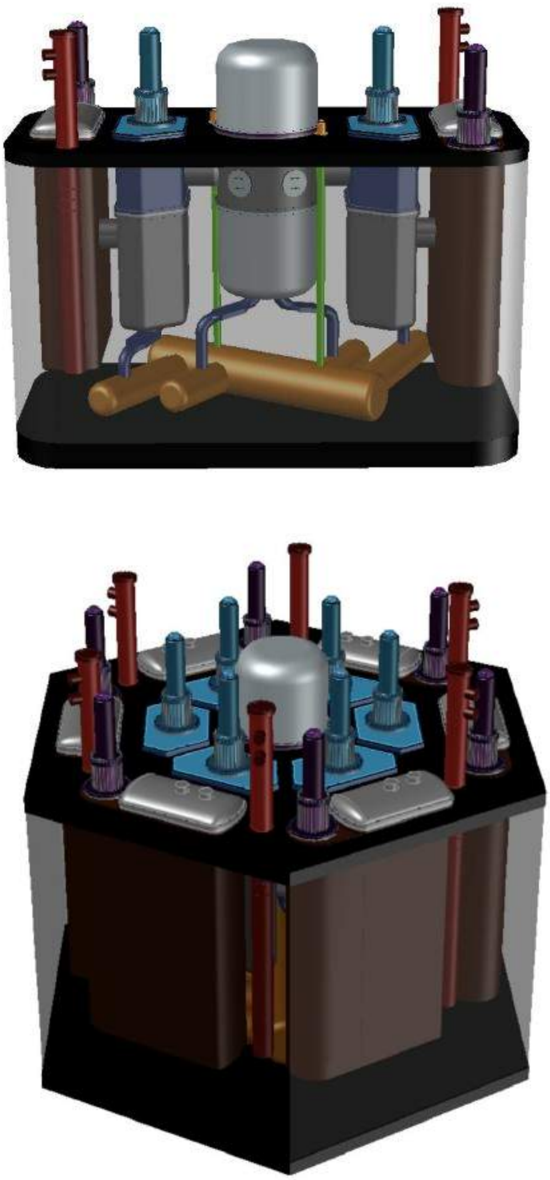


Elysium's modular reactor(s) (notional)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Elysium Industries, United States of America
Reactor type	Molten salt reactor
Coolant/moderator	NaCl-XCl$_z$-UCl$_{3/4}$-PuCl$_3$-FPCl$_y$ fuel salt / None
Thermal/electrical capacity, MW(t)/MW(e)	100/50 (Various options for 50, 200, 400 and up to 1200 MW(e) using same RPV)
Primary circulation	Forced circulation
System pressure (MPa)	0.1 + pump head + hydrostatic
Core inlet/exit temperatures (°C)	510/610 for the prototype; 600/750 as commercial and 650/950 as the ultimate goal
Fuel type/assembly array	Molten salt fuel
Fuel enrichment (%)	Various options (see fuel description)
Fuel cycle (months)	Online fuelling. Very long with on-line soluble purification and 40-100 years without it.
Main reactivity control mechanism	Short term: Negative temperature & void coefficients Long term: Fertile fuel additions
Approach to engineered safety systems	Passive and active (see description)
Design life (years)	20-40 for components, unlimited core, 100 for plant
Plant footprint (m^2)	1/3 size of LWR
RPV height/diameter (m)	9/3.7
Distinguishing features	Fast spectrum, no in-core structure, SNF/Pu fuelled
Design status	Conceptual design in progress, fuel development in testing

1. Introduction

The Elysium Industries molten chloride salt fast reactor (MCSFR) is being developed as a modular (configuration and construction) reactor. The MCSFR aims to enable fuel-efficient, reliable, passively-safe, proliferation-resistant, and environmentally-friendly energy generation that closes the fuel cycle. Unlike solid fuel in a traditional reactor, the fuel is part of the liquid heat transport eutectic fluid in the MCSFR. Heat is produced directly in the fuel coolant. The fuel salt exchanges heat to a clean, non-fissile and non-fertile salt that delivers heat to a super-heater/process heat exchanger, which powers a conventional ultra-super-critical steam turbine-generator set, i.e., a modern off-the-shelf coal power plant or process heat systems.

2. Target Application

The MCSFR supports modular construction and is designed for mass production to address global markets for cost-competitive, low-emission electricity and process heat. The MCSFR will enable economic growth, enhanced standards of living, and improved public health and social stability. The modular design allows low cost construction and great flexibility Therefore, the reactor can be deployed in both new and existing nuclear energy markets. The preferred initial fuel is spent nuclear fuel (SNF) and/or any type of Pu wastes that need to be consumed. Therefore, primary initial target countries are US, Canada, UK, Japan, South Korea, etc., because they have significant quantities of SNF or Pu that need to be consumed.

3. Specific Design Features

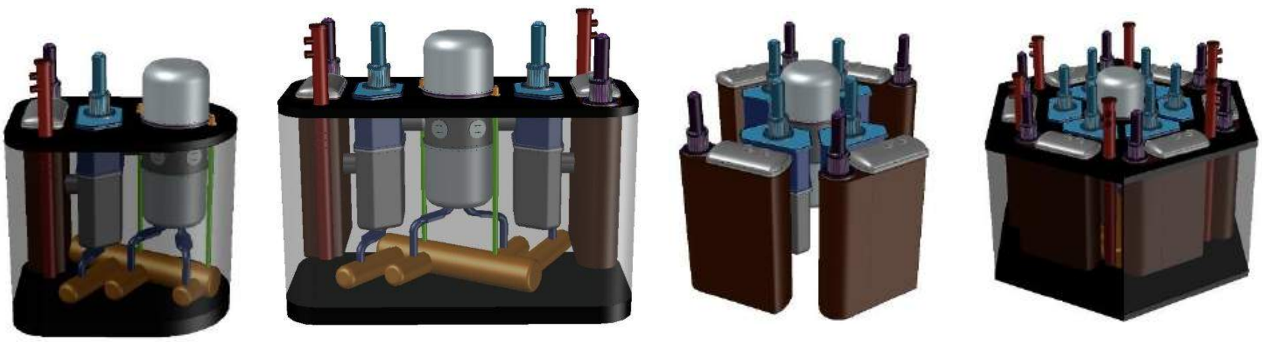
(a) Design Philosophy

The Elysium MCSFR and Elysium's experienced team designed the MCSFR with the aim to have the best and most practical features taken from many other reactor types. For example, from water reactors, common, low cost coolant (table salt), fuels (nuclear waste), and code qualified materials. From liquid metal reactors, the low pressure and materials without corrosion concerns. And from gas reactors, the very high temperatures. From heat-pipe reactors, passive temperature dependent on/off heat-pipe decay heat removal.

These features are used as opposed to expensive solid fuel, high pressure, and low power density/large size and the resultant higher cost features. In addition, the MCSFR aims to eliminate some of nuclear power's public concerns with the focus on improved safety, waste elimination, increased fuel supply, elimination of the Zirconium-water and sodium to hydrogen reactions, and significantly reduced capital construction costs and operational costs.

(b) Reactor and Core-unit

The Elysium MCSFR adopts a modular approach and the reactor size is minimized to just achieve criticality and designed with no in-core structures, that could be subjected to damage. So, utilities can start small and add components to the same reactor vessel to allow for greater power production without buying a new reactor module. Smaller heat exchangers and pumps are used for the lower power SMR deployments, whereas the larger configurations all use the same but larger modular heat exchanger configuration. This flexibility is portrayed in the figure below (not all possible lower & intermediate power configurations are included).



10-200MW(e) **200-400MW(e)** **600-800MW(e)** **1000 – 1200MW(e)**

Flexible power features: Add heat exchanger and pumps to extract more power from the reactor

(c) Power Conversion System and Cogeneration

The power conversion proposed at the initial operating outlet temperatures use an ultra-super-critical super-heater and steam turbine (Rankine cycle) making use of an improved Loeffler boiler, i.e., to achieve minimal water in containment. Elysium will start the prototype with maximum T_{hot} temperatures below 650°C, then up-rating it in power and temperature with experience to 750°C, and for commercial plants, allowing 40-50% electrical efficiencies. At the higher outlet temperature, proposed later in the project, the combination of process heat, the use of a combined open cycle gas turbine (OCGT), and a Rankine bottoming cycle, is proposed.

With the high outlet temperatures, many high-temperature process heat applications can be performed including hydrogen and synthetic fuel production, oil/gas recovery and refining, industrial process heat, and cement manufacturing (base heat or total heat with thermal booster). Other applications such as combined cooling heating and power (CCHP/Tri-gen + thermal storage) and desalination is of course also possible.

Use of super-heaters, rather than steam generators in the containment dramatically reduces water in containment, but also allows the same heat exchanger to be used for super-heating, or process heat gas, if steam is not desired. Different loops can be used for electricity, process heat for petroleum or other resource recovery and refining, H_2 or synthetic-fuel, district heating, desalination, etc., or just used for electricity in a large market.

(d) Reactivity Control

Reactivity control in the short term is via the negative temperature and void coefficients. As the fuel-salt temperature increases the fuel-salt expands and some of the fissile containing fuel salt is removed (squeezed out) from the active core, thus reducing fissions. The long-term reactivity adjustments are made by on-line fertile fuel additions. Fuel can also be moved (actively or passively) to the dump tank where it is kept in a sub-critical configuration.

(e) Fuel Characteristics

The fuel-salt is $NaCl-XCl_z-UCl_{3/4}-PuCl_3-FPCl_y$ and facilitates a flexible fuel source, allowing it to contain 10-20% fissile actinide fraction and consume 100% of actinides.

Fissile options include:

1. dispositioned reactor grade and weapons grade plutonium (RGPu and WGPu), which is the preferred fissile due to lower cost,
2. high assay low enriched uranium (HALEU),
3. high enriched uranium (HEU),
4. unusable HEU fuel, with options 1, 3 and 4 denatured with SNF during fuel production.

Fertile options include:

a. LWR/CANDU SNF,
b. Depleted Uranium (DU), thus consuming SNF and DU fuel stockpiles,
c. natural UO_2,
d. as mined UO_2/U_3O_8 ore,
e. residual U from other mining, coal ash, or seawater extraction, and
f. thorium (Th) combined with >88% U_{238} for denaturing.

The fertile material is added for startup only for Pu, i.e., fertile is already in HALEU. If HALEU is used, it requires feed-in of HALEU declining in enrichment over 5-10 years, until Pu builds up.

(f) Fuel Cycle Length and Approach

On-line refuelling is possible and the different options are described above. Different options are available for the soluble purification of the fuel-salt. With it no limits are foreseen for the fuel cycle length but if no soluble purification is performed the cycle will be ~40-100 years. The reactor components would require replacement sooner, so this may be combined with fuel treatment. Fuel can be reused after 90-95% fission product extractions and dilution to 10% Pu with $U_{238}Cl_x$ from SNF, DU, NU. The fuel consumption rates is ~ 1 ton U / GWe-yr and fuel is added every ~1-7 days (depending on the power level and fuel volume).

(g) Cooling System

Four different cooling loops are included: the fuel salt, secondary salt, the power/process heat, and the heat sink loop. The secondary salt is at higher pressure than the fuel salt to ensure in-leakage to the fuel salt while also preventing water ingress to the fuel salt (by burst disks etc.) if the super-heaters leak. Use of only super heaters against the salt limits water mass ingress. Depending on the power level a varying number (1-6) of fuel heat exchangers (to secondary salt loop) and secondary heat exchangers (to saturated steam or process heat gas) can be used. With the high temperature, dry cooling could be used.

(h) Proliferation Considerations

Actinides are never separated from fuel produced from SNF, thus maintaining excellent proliferation resistance. No actinides enter the waste stream, nor are Pu, U, and fission products (FPs) separated completely. This poor separation of FP's also reduces cost. Only relatively short-lived (100-300 years to below U background) FPs are removed as waste. The reactor reduces proliferation risk by denaturing plutonium with SNF, protecting it further by mixing it with FP's. Proliferation concerns associated with the back-end of the fuel cycle are minimized by never removing actinides from the fuel. Two options are available for FP removal, a simple system in the reactor removing only FP's, or operating the MCSFR for 40-100 years, then discharging and charging partially cleaned fuel back in, and shipping the used fuel to a facility for partial FP removal and reuse, thus reducing utility cost by reducing the purification system. The MCSFR converts the fertile actinides to fissile as needed by iso-breeding plus FP absorption buildup compensation. The utilization of all the actinides eliminates the need for storage of long-lived waste actinides, thus closing the fuel cycle.

4. Safety Features

(a) Engineered Safety Approach and Configuration

Passive features include the large mass and heat capacity of the fuel salt, pump coast down by use of flywheels, two independent drain activation systems to dump the fuel salt to the dump tank, multiple neutronically separated, passively cooled dump tanks, passive on/off cooling of dump tank, secondary salt higher pressure than fuel salt (leak shuts down), low fuel salt volume above the core drains core sooner, and cold salt does not drain through the core. The active safety features include operator trip of the freeze seal cooling and alternate dump system power draining of the reactor.

Fuel salt expanding squeezes fissile out of the core to control power, and shut down the reactor if temperatures rise further. Fail-safe operation is addressed by quick-melting freeze plugs and an independent faster drain start system. Enormous heat capacity is available in the salt surrounding the reactor and components to buffer temperature rise, and pumps have coast down fly-wheels.

The super-heater keeps the water content low in containment with two barriers to the fuel salt, and blowdown discs and tanks in the secondary salt system prevent secondary salt pressurization or water ingress near the fuel salt, if the super-heaters leak. The secondary salt is at higher pressure than the fuel salt to ensure in-leakage to the fuel salt, vice fuel leaking out, with secondary salt diluting fuel salt shutting down the reactor. However, secondary salt temperature is maintained above fuel salt freezing temperatures to prevent freezing, and allow passive dumping of the fuel salt, if temperatures start dropping or rising. Modular, non-cylindrical heat exchangers, ensure high neutron leakage and separation from the core to eliminate criticality concerns.

(b) Heat Removal System

Passive on/off decay heat exchangers remove decay heat indefinitely to air, while also preventing freezing of the fuel salt in the dump tanks.

(c) Containment Function

The Elysium design ensures public safety and a clean environment with intrinsic chemical inertness, design of thermal management, three physical barriers, in addition to chemical binding, physical freezing if fuel salt leaks, and continuous FP gas removal.

Because fuel and fission products are liquid and a chloride salt is used, selected fission products can be removed from the reactor during operation, minimizing the inventory of radioactive material in the reactor at any given time and reducing the effects of fission product absorptions. FP's removed from the reactor are converted to a redundantly safe waste form, while chlorine is recycled back into the fuel production process, and zirconium from SNF is recycled for LWR use, etc.

The chloride fuel salt in the Elysium MCSFR does not boil until ~1400°C, resulting in low pressures for safety, and allowing a very high temperature reactor containment temperature up to 650°C allowing use of existing qualified structural materials. All containment temperatures remain below ~650°C regardless of T_{hot} fuel temperatures, until new non-nickel structural materials can be qualified.

5. Plant Performance

The MCSFR is ideally suited to load follow or to support renewables because the core cannot be damaged by many or frequent transients, and is passively load following. No high metal mass steam generator is required that reacts slowly to load changes to prevent damage. Also, with a closed fuel cycle, and no solid fuel manufacturing costs, and extremely low fertile fuel costs, fuel consumption is not a serious cost issue, due to high fuel efficiency and liquid fuel, so turbine bypass is economically viable. For many, consuming more fuel via bypass operation, i.e., more SNF consumption, is considered a large benefit.

6. Plant Arrangement

The plant deployment can be performed in a modular fashion using the same reactor that is sized to just achieve criticality but allow for greater power production if additional or larger components (pumps and heat exchangers) are added. Locating the reactor underground protects the facility from airborne hazards like airplanes and tornadoes and, coupled with chemical ionic salt bonding of fission products and the natural freezing of leaked salt, reduces the likelihood of airborne radioactive release in the case of an unlikely event, and reduces security costs.

7. Design and Licensing Status

The pre-conceptual design is near completion.

8. Development Milestones

2018-2020	Pre-Conceptual design is near completion. Small scale tests of key concepts are underway with large scale testing estimated to start in 2020, followed by an Integrated Systems Test
2020-2025	Feasibility study, building a very low power prototype, pre-fission testing in 2023, fission testing starts in 2025.
2021-2027	Licensing, including prototype through power up-rating to ~200 MW(e)
2030	Commercial operation

OTHER
SMALL MODULAR REACTORS

Westinghouse eVinci® Micro Reactor (Westinghouse Electric Company LLC, USA)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse Electric Company LLC, USA
Reactor type	Monolithic core with heat-pipe technology
Coolant/moderator	Heat pipes/ metal hydride moderator
Thermal/electrical capacity, MW(t)/MW(e)	0.6-4 -40 / 0.2-15
Primary circulation	Heat pipes
System pressure (MPa)	NA
Core inlet/exit temperatures (°C)	NA
Fuel type/assembly array	UO_2 or UN
Number of fuel assemblies	Monolith Core
Fuel enrichment (%)	19.5
Fuel burnup (GWd/ton)	Less than 10 for a 1.2 MW(t) reactor
Fuel cycle (months)	120
Main reactivity control mechanism	Ex-core control drums
Approach to engineered safety systems	Inherent and passive safety; IAEA safety classification category B
Design Life (years)	10 Years
Plant Footprint (m^2)	< 1500
RPV Height/Diameter (m)	NA
Seismic design	IBC Zone 4 Category
Distinguishing features	Transportable reactor that operates semi-autonomously
Design status	TRL 5

1. Introduction

The Westinghouse eVinci® micro reactor is a semi-autonomous, very small modular reactor (vSMR) that is based on heat pipe technology. The eVinci micro reactor is a transportable energy generator that provides combined heat and power of 200 kW(e) – 15 MW(e). The design allows for it to be fully factory built, fuelled and assembled with a life expectancy of 10 years.

2. Target Application

The eVinci Micro Reactor's target application is the near-term generation of clean, safe and cost-competitive heat and electricity for remote communities, mines and military installations.

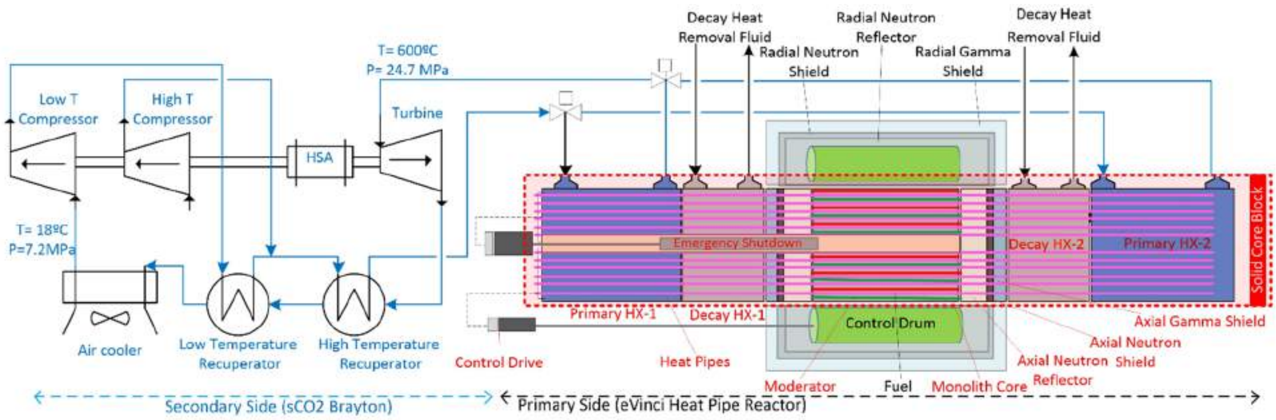
3. Specific Design Features

(a) Design Philosophy

The design of the Westinghouse eVinci micro reactor utilizes proven heat pipe technology developed by the Los Alamos National Laboratory (LANL) for space application. This uranium-fuelled reactor doesn't use a bulk primary coolant. Instead, heat is removed from its solid monolithic core using passive heat pipes, limiting the number of its moving parts and providing overall plant simplicity.

(b) Design Overview

The major components of the eVinci micro reactor are shown in the figure below. The reactor system consists of uranium nitride or oxide fuel, and metal hydride moderator housed in a compact monolith, constructed of creep-resistant, high-temperature materials with embedded heat pipe channels arranged in a hexagonal pattern with the fuel channels. The monolith core by itself is subcritical; to achieve criticality the core must be surrounded by large radial and axial reflectors. Control drums in the reflector are used to control reactor temperatures and power levels. A central emergency safety shutdown or the hydrogen release in the metal hydride moderator is used to perform the safe shut down the reactor.



System schematic of eVinci energy generator

High-temperature, double-ended sodium heat pipes are used to move heat from the core region to various heat exchangers. Alkali-metal heat pipes are extremely effective at moving heat over long-distances with minimal temperature drop, eliminating the need for reactor coolant pumps and other auxiliary systems related to primary reactor cooling. The eVinci micro reactor allows for operation with a variety of power conversion systems without having to redesign the core each time. For example, although the reference design uses a 600ºC supercritical carbon dioxide (sCO_2) Brayton cycle, the reactor can be easily adapted to work with Stirling engines.

A high reliability is obtained through the selection of a heat pipe solid core block with nearly no moving parts. The only mechanically moving part in eVinci (excluding power conversion) is the reactor control drum. The limited moving parts and autonomous operation reduce the need for personnel and periodic maintenance, further enhancing the economic case. The solid monolith core block enables proliferation resistance by encapsulating fuel in the monolith. Heat pipes eliminate the need for reactor coolant pump and all its auxiliary fluid systems, thereby leading to plant simplification. The inherent load following capability of heat pipes, self-adjusting solid core and inherent decay heat removal via solid state conduction and air rejection, enables the autonomous operation and superior safety of the eVinci Micro Reactor.

4. Resilient Features

The resiliency of eVinci Micro Reactor is enabled through its packaging in a secure canister, installed in a strong secure vault on site and connected to a micro-grid coordinating both heat and power. A SMART device incorporated within its power conversion and heat delivery systems, along with its autonomous load following capability make it ideal to operate together with other energy resources. The integration capability of the eVinci Micro Reactor to connect to a micro-grid can make users resilient by utilizing the diversity, high-reliability and safety of nuclear energy.

5. Operational Performances

The eVinci Micro Reactor's energy efficiency performance is expected to be 86% (27-35% electrical efficiency), and its load-follow capabilities are expected to be around 30% through inherent core physics, 100% with controls. Inherent load following capability is enabled through the physics based characteristics of the heat pipe and reactor core design.

6. Instrumentation and Control Systems

The eVinci™ Micro Reactor's inherent control characteristics leads to an inherently safe reactor that does not require a safety-related instrumentation and control (I&C) system. The instrument and control will be used for normal operational activities and allows for coupling to micro-grids and remote monitoring systems.

7. Plant Arrangement

The eVinci Micro Reactor is designed to be fully self-contained, factory built and transportable via truck, rail or ship. The eVinci Micro Reactor addresses the large economic uncertainty of deploying a typical nuclear system that primarily uses on-site construction. In contrast to several years of construction for current nuclear plants, the eVinci design is targeting for less than a 30 day on-site installation. Once the transportable reactor canister reaches the customer site, it can be installed below or above grade. The only connection needed to be made is the power conversion, grid connection and/or process heat distribution. After a useful life of 10 years, the eVinci Micro Reactor can be easily disconnected and transported back to the factory for long term storage or reuse. Due to the lack of large bulk coolant, eVinci does not involve the separation of fuel from coolant, eliminating any decontamination, decommissioning and remediation (DD&R) activities and will leave the site with greenfield-like conditions. The design of the reactor allows for the reactor to be repacked for re-use or prepared for long term storage at the factory.

8. Design and Licensing Status

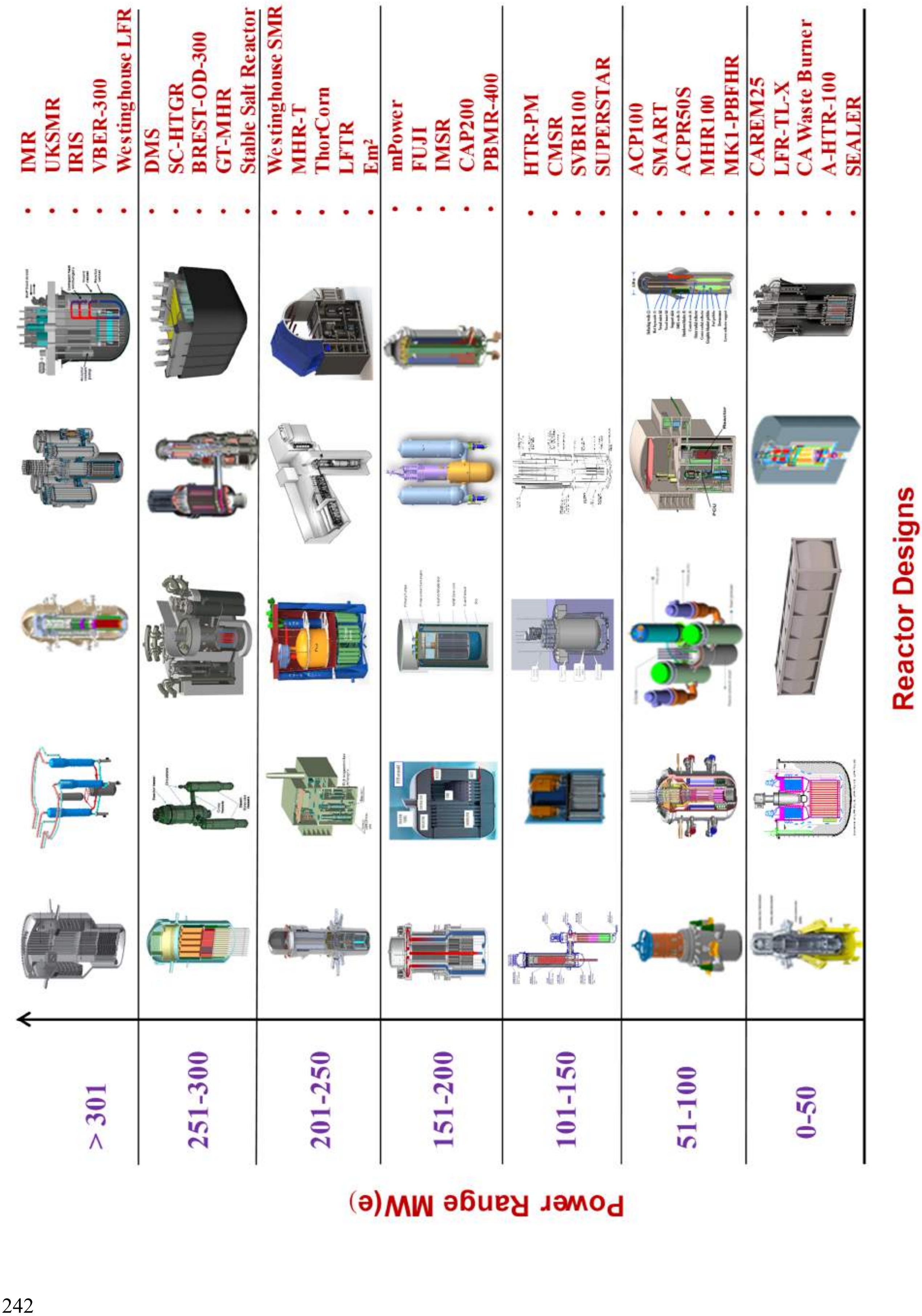
The eVinci micro reactor is currently at technology readiness level (TRL) 5. Westinghouse is currently working with both the U.S. Nuclear Regulatory Commission (NRC) and the Canadian Nuclear Safety Commission (CNSC) to license the technology.

9. Development Milestones

2019	Electric demonstration
2023	First nuclear demonstration

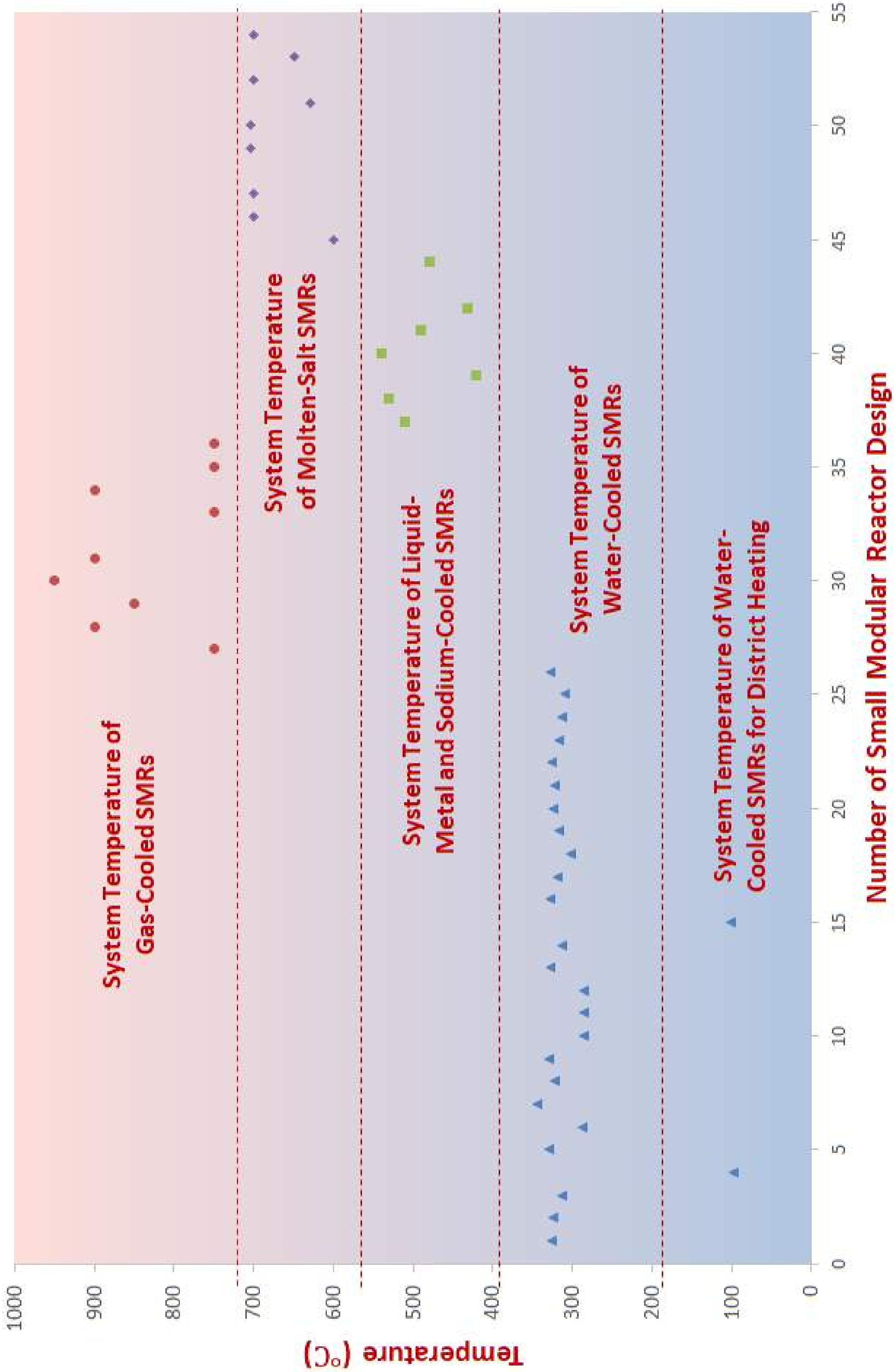
ANNEX I

Summary of SMR Designs Based on Power Range



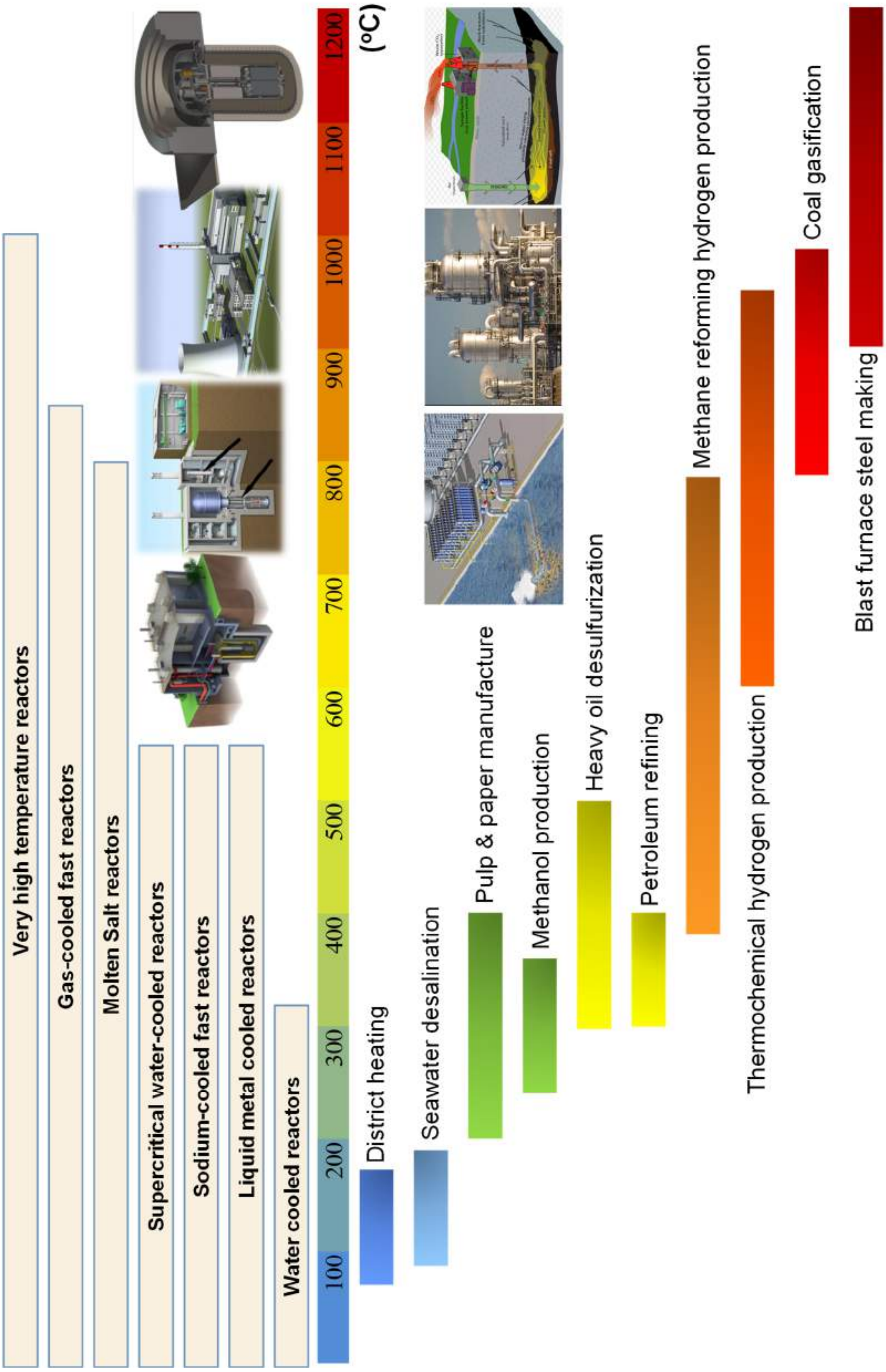
ANNEX II

Summary of SMR Designs Based on Core Exit Temperature

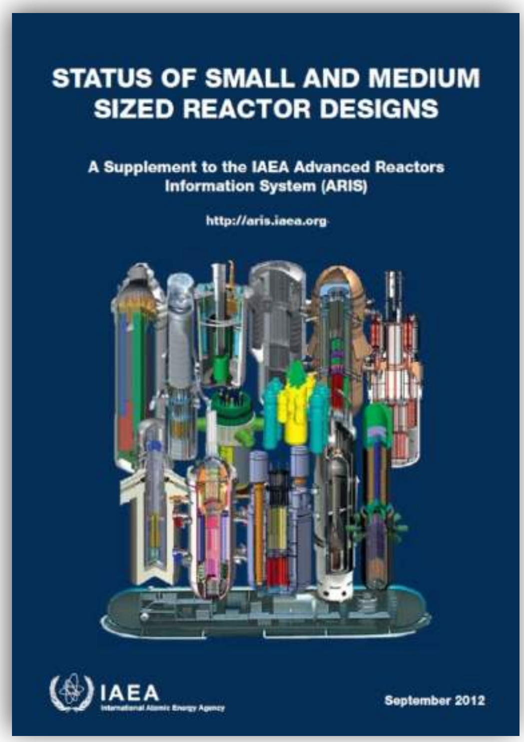


ANNEX III

Summary of SMR Designs for Non-Electric Applications

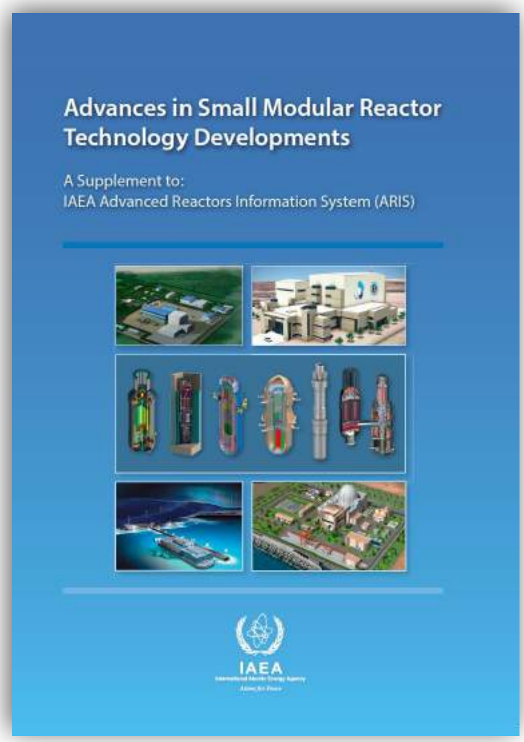


ANNEX IV
Bibliography

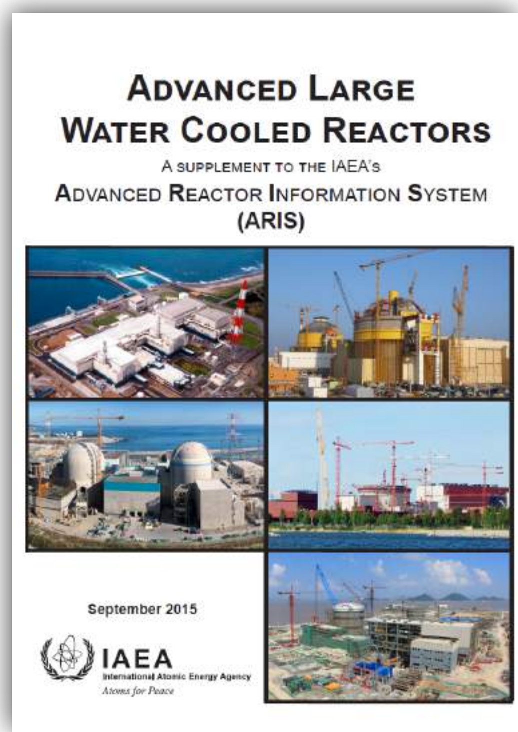


- Contained status, design description and main features of 32 selected SMR designs;
- Sorted by type/coolant: iPWR, PHWR, GCR, and LMFR;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), FBNR (Brazil), CNP-300 (China), Flexblue (France), IMR (Japan), SMART (Republic of Korea), ABV-6M (Russian 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

- Contained status, design description and main features of 48 selected SMR designs;
- Sorted by Land based and Marine based LWRs, HTGR, Fast spectrum SMRs and Molten Salt SMRs;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), ACP100, CAP150, CAP200 (China), AHWR-300 (India), IRIS (International Consortium), DMS and IMR (Japan), SMART (Republic of Korea), UNITHERM, KARAT-45, KARAT-100, ELENA, RUTA-70 (Russian Federation), NuScale, mPower, Westinghouse SMR, SMR-160 (United States of America)
  ACPR50S (China), Flexblue (France), KLT-40S, RITM-200, VBER-300, ABV-6E, SHELF (Russian Federation), HTR-PM (China), GTHTR300 (Japan), GT-MHR, MHR-T, MHR-100 (Russian Federation), PBMR-400, HTMR-100 SMR (South Africa), SC-HTGR, Xe-100 (United States of America)
  LEADIR-PS (Canada), 4S (Japan), BREST-OD-300, SVBR-100 (Russian Federation), G4M, EM2 (United States of America)
  Integral Molten Salt Reactor (Canada), MSTW (Denmark), ThorCon (International Consortium), FUJI (Japan), Stable Salt Reactor (United Kingdom), SmAHTR, Liquid Fluoride Thorium Reactor, Mk1 PB-FHR (United States of America)
- Published August 2016

ANNEX V
Acronyms

AC	Alternating Current
ADS	Automatic Depressurization System
ALARA	As Low As Reasonably Achievable
ARIS	Advanced Reactor Information System
ASEC	Air Heat Sink for Emergency Cooldown
ASME	American Society of Mechanical Engineers
ATWS	Anticipate Transient Without SCRAM
BCR	Back-up Control Room
BDBA	Beyond Design Basis Accident
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCS	Containment Cooling System
CCWS	Component Cooling Water System
CEDM	Control Element Drive Mechanism
CES	Containment Enclosure Structure
CHR	Containment Heat Removal System
CIS	Containment Isolation System
CMT	Core Make-up Tank
CNPP	Cogeneration Nuclear Power Plant
CNSC	Canadian Nuclear Safety Commission
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
DCS	Distributed Control System
DID	Defence in Depth
DLOFC	Depressurized Loss of Forced Cooling
DRACS	Direct Reactor Auxiliary Cooling System
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
FBR	Fast Breeder Reactor
FE	Fuel Element
FPU	Floating Power Unit
FSAR	Final Safety Analysis Report
FSS	Free Surface Separation
GCB	Generator Circuit Breaker
GDA	Generic Design Assessment
GDCS	Gravity Driven Cooling System
GDWP	Gravity Driven Water Pool
GFR	Gas-cooled Fast Reator
HEU	High Enriched Uranium

HFE	Human Factors Engineering
HLMC	Heavy Liquid Metal-Cooled
HHTS	Hybrid Heat Transport System
HPCF	High Pressure Core Flooder
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
HX	Heat Exchanger
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
IHX	Intermediate Heat Exchanger
IPIT	Intermediate Pressure Injection Tanks
I&C	Instrumentation and Control
LEU	Low Enriched Uranium
LFR	Lead-cooled Fast Reactor
LFTR	Liquid-Fluoride Thorium Reactor
LLSF	Low Level Safety Functions
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPFL	Low Pressure Core Flooder
LWR	Light Water Reactor
MA	Minor Actinides
MCR	Main Control Room
MCSFR	Molten Chloride Salt Fast Reactor
MHT	Main Heat Transport
MOX	Mixed Oxide
MSA	Moisture Separator Reheater
MSFR	Molten Salt Fast Reactor
MSR	Molten Salt Reactor
MW(e)	Mega Watt electric
MW(t)	Mega Watt thermal
NDHP	Nuclear District Heating Plant
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTEP	Nuclear Thermoelectric Plant
OBE	Operating Basis Earthquake
OCP	Outside Containment Pool
ORNL	Oak Ridge National Laboratory
OTSG	Once-Through Steam Generators
OTTO	Once Through Then Out
PAR	Passive Autocatalytic Re-Combiners
PC	Primary Containment
PCS	Primary Containment System
PCT	Peak Cladding Temperature
PCU	Power Conversion Unit
PCV	Primary Containment Vessel
PCCS	Passive Containment Cooling System
PDHR	Passive Decay Heat Removal
PHTS	Primary Heat Transport 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
RCCS	Reactor Cavity Cooling System
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RCSS	Reactivity Control and Shutdown System
RDP	Reactor automatic Depressurization System
RFA	Robust Fuel Assembly
RHRS	Residual Heat Removal System
RP	Reactor Plant
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RV	Reactor Vessel
SBO	Station Black-Out
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator
SIT	Safety Injection Tanks
SMR	Small Modular Reactor
SNF	Spent Nuclear Fuel
SSC	Systems, Structures and Components
SSE	Safe Shutdown Earthquake
TC	Turbo Compressor
T/G	Turbine/Generator
TEG	Thermoelectric Generator
TEU	Thermoelectric Unit
TM	Turbo Machine
TRISO	Triple Coated Isotropic
TRL	Technology Readiness Level
UCO	Uranium Oxy Carbide
UHS	Ultimate Heat Sink
WATSS	Waste to Stable Salt
WDS	Waste Disposal System
WPu	Weapon-Grade Plutonium
WWER	Water Moderated Power Reactor

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