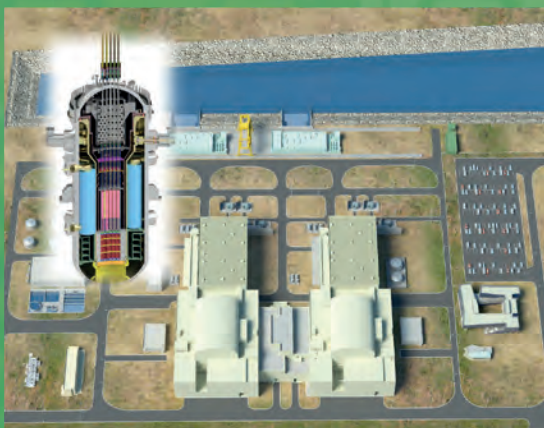
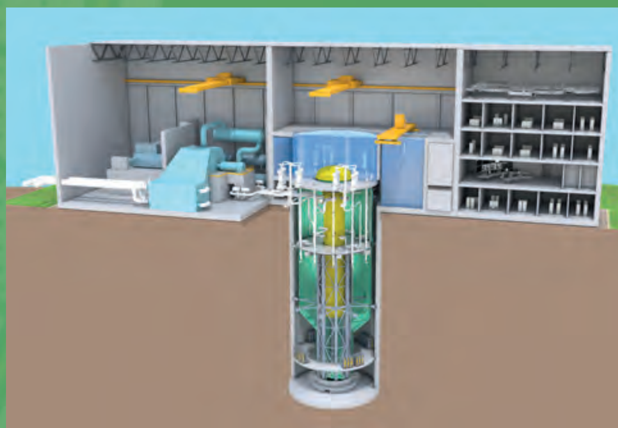
Advances in Small Modular Reactor Technology Developments

A Supplement to:
IAEA Advanced Reactors Information System (ARIS)
2020 Edition





IAEA
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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 advances in SMR design and technology developments, as well as global trend of their deployment. The IAEA Division of Nuclear Power, which has been facilitating Member States in addressing common technologies and issues for SMRs, 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 water-cooled, fast neutron spectrum, and high temperature gas cooled reactor technology development, as well as the non-electric applications.

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 by deploying microreactors to replace diesel generators in small islands or remote regions.

Booklets on the status of SMR technology developments have been published in 2011, 2012, 2014, 2016 and 2018. 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, molten salt reactors, and the recent development of a sub-category called micro modular reactors with electrical power typically up to 10 MW(e). For the first time also that the booklet provides some insights on associated fuel cycles and radioactive waste management of the SMR designs reported herein. 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. Previous booklets published in support of ARIS are listed in Annex VIII.

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 F. Reitsma, M.H. Subki, J.C. Luque-Gutierrez and S. Bouchet 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.

The world will need to harness all low-carbon sources of energy in order to meet the Paris Agreement goal of limiting the rise in global temperatures to well below 2ºC above pre-industrial levels. Use of renewables such as wind and solar power will continue to grow. However, nuclear power provides the steady and reliable stream of electricity needed to run and grow an advanced economy, and to enable developing countries to boost economic output and raise living standards. Together with hydropower, nuclear is the only low-carbon source of energy that can replace fossil fuels for 24/7 baseload power.

There is increasing interest in small modular reactors (SMRs) and their applications. During the International Conference on Climate Change and the Role of Nuclear Power held in September 2019, it was revealed that SMRs are being considered by many Member States as a potential viable nuclear option to contribute mitigating the climate change. SMRs are newer generation reactors designed to generate electric power typically 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, molten salt reactors, and most recently microreactors. 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 economic competitiveness is still to be proven as these units are deployed in future.

A major milestone has been reached in SMR technology deployment. The Akademik Lomonosov floating power unit in the Russian Federation with two-module KLT40S has been connected to the grid and started commercial operation in May 2020. There are more than seventy (70) SMR designs under development for different application. Two 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). They are scheduled to start operation between 2021 and 2023.

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 2020 edition comprises of six (6) parts arranged in the order of the different types of coolants, the neutron spectrum adopted, and the sixth part is dedicated for microreactors, as a new sub-category within SMR and to cover other SMRs that do not use the traditional coolants and/or fuel design, such as heat pipes.

Table 1: Design and Status of SMRs included in 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	Detailed Design
CANDU SMR	300	PHWR	Candu Energy Inc (SNC-Lavalin Group)	Canada	Conceptual Design
CAP200	200	PWR	SNERDI/SPIC	China	Conceptual Design
DHR400	400 MW(t)	LWR (pool type)	CNNC	China	Basic Design
HAPPY200	200 MW(t)	PWR	SPIC	China	Detailed Design
TEPLATOR™	50 MW(t)	HWR	UWB Pilsen & CIIRC CTU	Czech Republic	Conceptual Design
NUWARD	2 × 170	PWR	EDF, CEA, TA, Naval Group	France	Conceptual Design
IRIS	335	PWR	IRIS Consortium	Multiple Countries	Basic Design
DMS	300	BWR	Hitachi-GE Nuclear Energy	Japan	Basic Design
IMR	350	PWR	MHI	Japan	Conceptual Design
SMART	107	PWR	KAERI and K.A.CARE	Republic of Korea, and Saudi Arabia	Certified Design
RITM-200	2 × 53	PWR	JSC "Afrikantov OKBM"	Russian Federation	Under Development
UNITHERM	6.6	PWR	NIKIET	Russian Federation	Conceptual Design
VK-300	250	BWR	NIKIET	Russian Federation	Detailed Design
KARAT-45	45 - 50	BWR	NIKIET	Russian Federation	Conceptual Design
KARAT-100	100	BWR	NIKIET	Russian Federation	Conceptual Design
RUTA-70	70 MW(t)	PWR	NIKIET	Russian Federation	Conceptual Design
ELENA	68 kW(e)	PWR	National Research Centre "Kurchatov Institute"	Russian Federation	Conceptual Design
UK SMR	443	PWR	Rolls-Royce and Partners	United Kingdom	Conceptual Design
NuScale	12 × 60	PWR	NuScale Power Inc.	United States of America	Under Regulatory Review
BWRX-300	270 - 290	BWR	GE-Hitachi Nuclear Energy and Hitachi GE Nuclear Energy	United States of America, Japan	Pre-licensing
SMR-160	160	PWR	Holtec International	United States of America	Preliminary Design
W-SMR	225	PWR	Westinghouse Electric Company, LLC	United States of America	Conceptual Design
mPower	2 × 195	PWR	BWX Technologies, Inc	United States of America	Conceptual Design
PART 2: WATER COOLED SMALL MODULAR REACTORS (MARINE BASED)					
KLT-40S	2 × 35	PWR in Floating NPP	JSC Afrikantov OKBM	Russian Federation	In Operation
RITM-200M	2 × 50	PWR in FNPP	JSC Afrikantov OKBM	Russian Federation	Under Development
ACPR50S	50	PWR in FNPP	CGNPC	China	Conceptual Design
ABV-6E	6-9	PWR in FNPP	JSC Afrikantov OKBM	Russian Federation	Final design
VBER-300	325	PWR in FNPP	JSC Afrikantov OKBM	Russian Federation	Licensing Stage

SHELF	6.6	PWR in Immersed NPP	NIKIET	Russian Federation	Detailed Design
PART 3: HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS					
HTR-PM	210	HTGR	INET, Tsinghua University	China	Under Construction
StarCore	14/20/60	HTGR	StarCore Nuclear	Canada/UK/US	Pre-Conceptual Design
GTHTR300	100 - 300	HTGR	JAEA	Japan	Pre-licensing
GT-MHR	288	HTGR	JSC Afrikantov OKBM	Russian Federation	Preliminary Design
MHR-T	4 × 205.5	HTGR	JSC Afrikantov OKBM	Russian Federation	Conceptual Design
MHR-100	25 – 87	HTGR	JSC Afrikantov OKBM	Russian Federation	Conceptual Design
PBMR-400	165	HTGR	PBMR SOC Ltd	South Africa	Preliminary 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
Xe-100	82.5	HTGR	X-Energy LLC	United States of America	Basic Design
SC-HTGR	272	HTGR	Framatome, Inc.	United States of America	Conceptual Design
HTR-10	2.5	HTGR	INET, Tsinghua University	China	Operational
HTTR-30	30 (t)	HTGR	JAEA	Japan	Operational
RDE	3	HTGR	BATAN	Indonesia	Conceptual Design
PART 4: FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS					
BREST-OD-300	300	LMFR	NIKIET	Russian Federation	Detailed Design
ARC-100	100	Liquid Sodium	ARC Nuclear Canada, Inc.	Canada	Conceptual Design
4S	10	LMFR	Toshiba Corporation	Japan	Detailed Design
microURANUS	20	LBR	UNIST	Korea, Republic of	Pre-Conceptual Design
LFR-AS-200	200	LMFR	Hydromine Nuclear Energy	Luxembourg	Preliminary Design
LFR-TL-X	5~20	LMFR	Hydromine Nuclear Energy	Luxembourg	Conceptual Design
SVBR	100	LMFR	JSC AKME Engineering	Russian Federation	Detailed Design
SEALER	3	LMFR	LeadCold	Sweden	Conceptual Design
EM^2	265	GMFR	General Atomics	United States of America	Conceptual Design
Westinghouse LFR	450	LMFR	Westinghouse Electric Company, LLC.	United States of America	Conceptual Design
SUPERSTAR	120	LMFR	Argonne National Laboratory	United States of America	Conceptual Design
PART 5: MOLTEN SALT SMALL MODULAR REACTORS					
Integral MSR	195	MSR	Terrestrial Energy Inc.	Canada	Conceptual Design
smTMSR-400	168	MSR	SINAP, CAS	China	Pre-Conceptual Design
CA Waste Burner 0.2.5	20 MW(t)	MSR	Copenhagen Atomics	Denmark	Conceptual Design
ThorCon	250	MSR	ThorCon International	International Consortium	Basic Design
FUJI	200	MSR	International Thorium Molten-Salt Forum: ITMSF	Japan	Experimental Phase
Stable Salt Reactor - Wasteburner	300	MSR	Moltex Energy	United Kingdom / Canada	Conceptual Design
LFTR	250	MSR	Flibe Energy, Inc.	United States of America	Conceptual Design
KP-FHR	140	Pebble-bed salt cooled Reactor	KAIROS Power, LLC.	United States of America	Conceptual Design
Mk1 PB-FHR	100	FHR	University of California at Berkeley	United States of America	Pre-Conceptual Design
MCSFR	50 - 1200	MSR	Elysium Industries	USA and Canada	Conceptual Design

PART 6: MICRO MODULAR REACTORS					
Energy Well	8	FHTR	Centrum výzkumu Řež	Czech Republic	Pre-Conceptual Design
MoveluX	3~4	Heat Pipe	Toshiba Corporation	Japan	Conceptual Design
U-Battery	4	HTGR	Urenco	United Kingdom	Conceptual Design
Aurora	1.5	FR	OKLO, Inc.	United States of America	Conceptual Design
Westinghouse eVinci	2 -3.5	Heat Pipe	Westinghouse Electric Company, LLC.	United States of America	Under Development
MMR	5-10	HTGR	Ultra Safe Nuclear Corporation	United States of America	Preliminary Design

Part One: Land-based water-cooled SMRs. This part presents notable water-cooled SMR designs from various configurations of light water reactor (LWR) and heavy water reactor (HWR) technologies for on-land on-the-grid applications. These designs represent the mature technology considering most of the large power plants in operation today are of water-cooled reactors. There are twenty-five (25) water-cooled SMR designs from 12 Member States described in this booklet that comprises integral-PWRs, compact-PWRs, loop-PWRs, BWRs, CANDU-type designs, and pool-type reactors for district heating. An integral-PWR with natural circulation, designated as CAREM is finalizing construction for operation by 2023. Dozens of designs are being prepared for near-term deployment, including the ACP-100 in China and NuScale in the United States.

Part Two: Marine-based water-cooled SMRs. This part presents concepts that can be deployed in a marine environment, either as barge-mounted floating power unit or immersible underwater power unit. This unique application provides many flexible deployment options. This booklet presents six (6) marine-based water-cooled SMRs, some of them have been deployed as nuclear icebreaker ships. The first SMR connected to the grid is from this category, with the deployment KLT-40S for the Akademik Lomonosov floating nuclear power plant in Pevek, Russian Federation that started commercial operation in May 2020.

Part Three: High Temperature Gas Cooled SMRs: This part 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. Eleven (11) HTGR-type SMRs are described in this booklet, including HTR-PM, which is the next SMR to start operation in 2021 in China and three (3) HTGR test-reactors, two that have been in operation for technology testing purposes in Japan and China for over twenty years.

Part Four: Fast Neutron Spectrum SMRs. This part presents eleven (11) SMR designs that adopt fast neutron spectrum with all different coolant options, including sodium, heavy liquid metal (e.g. lead or lead-bismuth) and helium-gas. Tangible advances in technology development and deployment on SMRs in this category have been made. The BREST-OD-300, a lead-cooled fast neutron reactor is in the process of construction at a site in Seversk, Russian Federation with a scheduled operation by end of 2026. This is a demo-prototype project for future design with large power to enable a closed nuclear fuel cycle.

Part Five: Molten Salt SMRs. This part highlights ten (10) SMR designs from molten salt fuelled and cooled advanced reactor technology (MSRs), which is also one of the six Generation IV reactor designs. MSRs promise many advantages including enhanced safety due to salt's inherent property, low-pressure single-phase coolant system that eliminates the need of large containment, a high temperature system that results in high efficiency, and flexible fuel cycle. Several MSR designs are conducting preliminary licensing activities in Canada, United Kingdom and the United States.

Part Six: Micro-sized SMRs. This booklet now contains a dedicated part to present advances on microreactors. An unprecedented development trend emerged on very small SMRs designed to generate electrical power of typically up to 10 MW(e). They are from different types of coolant, including HTGRs and designs that use heat pipes for heat transport. Several designs are undertaking licensing activities in Canada and the United States for planned near-term deployment. In 2019 a site application was submitted by Global First Power for a single small modular reactor using USNC's Micro Modular Reactor (MMR) technology at the Chalk River Laboratories site. Microreactors serve future niche electricity and district heat markets in remote regions, mining, industries and fisheries that for decades have been served by diesel power plants. Six (6) microreactor designs are included and discussed in this booklet.

For 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. Some SMR designers did not make reportable new milestones or advances, however, their inclusion in the booklet are to the benefit of Member State readers by providing lessons learned in design development. Not all small reactor designs presented can strictly be categorized as SMRs. 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.

This booklet is enriched with a set of annexes that provide Member States' readers with various charts and tables to understand the essential technical facts of SMR designs. The description of each annex is given below. For the first time this edition of booklet many designs also contain information on the fuel cycle, waste management and disposal plan and these are also summarized in new annexes. The particular concepts that have been mentioned in these annexes are just examples useful to elaborate on possible different options. The summaries provided have not undergone an official review by the IAEA. The views expressed do not necessarily reflect those of the International Atomic Energy Agency or its Member States and remain the responsibility of the contributors.

Annex I provides charts that summarize global SMR technology development and deployment. It includes a SMR developers world map, general timeline of deployment, government and private sectors on SMR technology development, and an interesting chart that presents stage of design or deployment of SMRs in terms of their output capacity.

Annex II shows power ranges of SMR designs of different category of coolant and neutron spectrum.

Annex III shows comparison of main technical characteristics among several SMR Designs.

Annex IV recaps SMR design for different non-electric applications. A chart is provided that shows SMR according to the core exit coolant temperature.

Annex V shows the dimensions of reactor vessels in water-cooled SMRs and HTGR-SMRs.

Annex VI provides a summary on the fuel cycle approach adopted in by SMR designs, furnished with a table that categorize different SMR types according to the fuel cycle types (open or closed cycle), refuelling cycle interval, enrichment level, spent fuel processing and conditioning, the use of Thorium-cycle and/or Plutonium disposition and the use of spent fuel as fuel.

Annex VII provides a summary on the waste management approach, technology and disposal plan adopted by SMR designs. It contains a table on the adopted categories, i.e. volume reduction and conditioning, waste processing, storage approach, spent fuel pool cooling mechanism, spent fuel take back option, and market potential.

Other booklets previously published in support of ARIS are listed in **Annex VIII**.

Annex IX contains a list of commonly used acronyms.

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).

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.

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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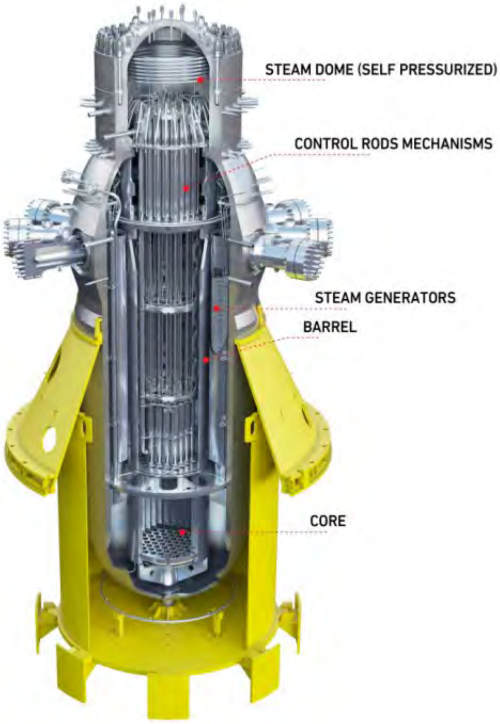


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNEA, Argentina
Reactor type	Integral PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	100 / ~30
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	12.25 / 4.7
Core Inlet/Outlet Coolant Temperature (ºC)	284 / 326
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	61
Fuel enrichment (%)	3.1% (prototype)
Core Discharge Burnup (GWd/ton)	24 (prototype)
Refuelling Cycle (months)	14 (prototype)
Reactivity control mechanism	Control rod driving mechanism (CRDM) only
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	Not available
RPV height/diameter (m)	11 / 3.2
RPV weight (metric ton)	267
Seismic Design (SSE)	0.25g
Fuel Cycle Requirements or Approach	390 full-power days and 50% of core replacement (prototype)
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. Main 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 390 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 and 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 in 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 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 in 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, provisions are 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 withstand the pressure of 0.5 MPa. Ultimate heat sink inside the containment during the grace period provides protection for extreme external events.

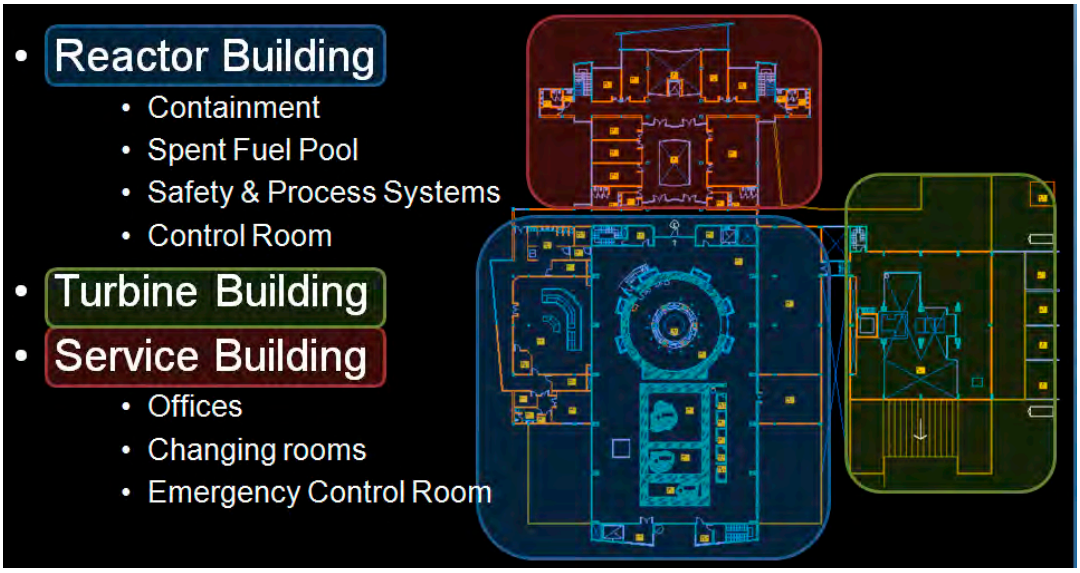
5. Plant Safety and Operational Performances

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.

7. Plant Layout Arrangement



CAREM 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. The nuclear building is under construction.

9. Fuel Cycle Approach

The fuel cycle can be tailored to customer requirements, with a reference open fuel cycle design for the prototype of 390 full-power days and 50% of core replacement.

10. Waste Management and Disposal Plan

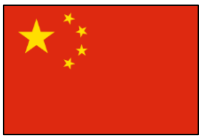
Waste management facilities for waste treatment and storage are provided. Provisions for prolonged temporary storage of solid wastes at the site are considered.

11. Plant Economics

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

12. Development Milestones

1984	CAREM concept was presented in Lima, Peru, during the IAEA Conference on SMRs and was one of the first of the new generation reactor designs. CNEA officially launched the CAREM project
2001-02	The design was evaluated on generation IV international forum and was selected in the near term development group
2006	Argentina Nuclear Reactivation Plan listed the CAREM-25 project among priorities of national nuclear development
2009	CNEA submitted its preliminary safety analysis report (PSAR) for CAREM-25 to the ARN. Announcement was made that Formosa province was selected to host the CAREM
2011	Start-up of a high pressure and high temperature loop for testing the innovative hydraulic control rod drive mechanism (CAPEM)
2011	Site excavation work began and contracts and agreements between stakeholders are under discussion
2012	Civil engineering works
2014	8 February, formal start of construction
2023	First criticality



ACP100 (CNNC, China)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNNC(NPIC/CNPE) China
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	385 / 125
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15 / 4.6
Core Inlet/Outlet Coolant Temperature (ºC)	286.5 / 319.5
Fuel type/assembly array	UO_2/17x17 square pitch arrangement
Number of fuel assemblies in the core	57
Fuel enrichment (%)	<4.95
Core Discharge Burnup (GWd/ton)	<52 000
Refuelling Cycle (months)	24
Reactivity control mechanism	Control rod drive mechanism (CRDM), Gd_2O_3 solid burnable poison and soluble boron acid
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	200 000
RPV height/diameter (m)	10 / 3.35
RPV weight (metric ton)	300
Seismic Design (SSE)	0.3g
Fuel Cycle Requirements or Approach	Temporarily stored in spent fuel pools
Distinguishing features	Integrated reactor with tube-in-tube once through steam generator, nuclear island underground
Design status	Detailed design

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. Main 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 U_{235} 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 20 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

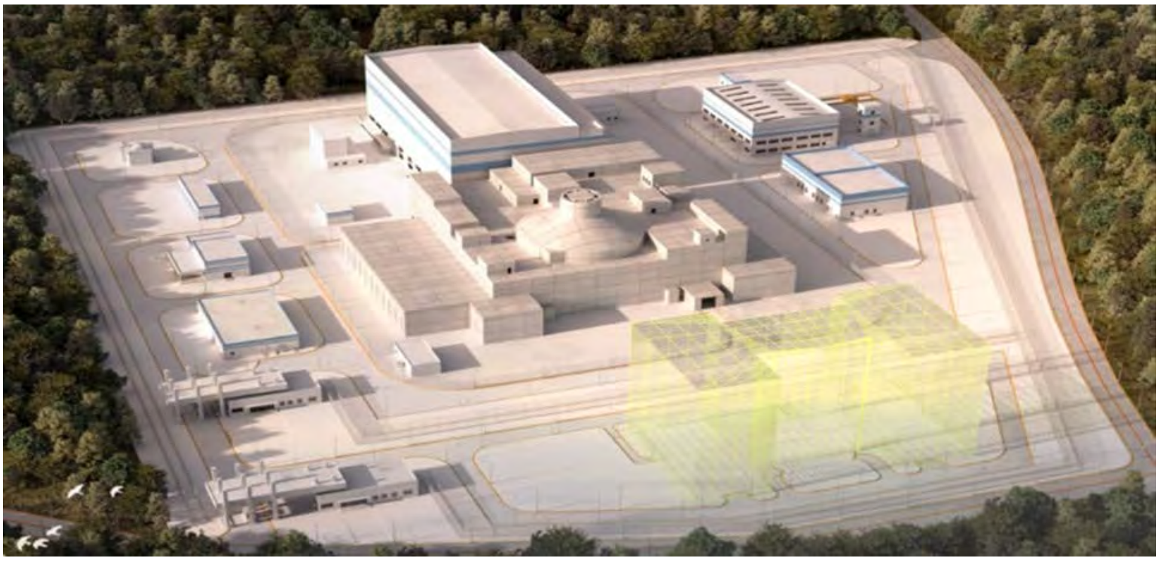
The ACP100 primary cooling mechanism under normal operating condition and shutdown condition is done by forced circulation. The RCS has been designed to ensure adequate cooling of reactor core under all operational states, during and following all postulated off normal conditions. The integral design 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 in-containment 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-containment 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 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 Layout 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 to work 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 385MW(t) unit is planned in Hainan Province, China.

9. Fuel Cycle Approach

Spent fuel processing is similar to other nuclear power plants. It is temporarily stored in spent fuel pools.

10. Waste Management and Disposal Plan

Waste management approach and disposal plan is similar to other nuclear power plants.

11. 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.
2019	PSAR submitted to National Nuclear Safety Authority
2019.07	Site Preparation started
2020.08	Target FCD

CANDU SMR™ (Candu Energy Inc, Canada)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Candu Energy Inc. Member of the SNC-Lavalin Group, Canada
Reactor type	Pressurized Heavy Water Reactor
Coolant/moderator	Heavy Water (D_2O)
Thermal/electrical capacity, MW(t)/MW(e)	960 / 300
Primary circulation	Forced
NSSS Operating Pressure (primary/secondary), MPa	9.9 / 4.6
Core Outlet Coolant Temperature (^oC)	310
Fuel type/assembly array	37 elements
Number of fuel assemblies in the core	2064
Fuel enrichment (%)	Natural Uranium; not enriched
Core Discharge Burnup (GWd/ton)	5.8
Refuelling Cycle (months)	On-line
Reactivity control mechanism	Zone controllers, mechanical adjusters
Approach to safety systems	Combined active and passive
Design life (years)	70 years
Plant footprint (m^2)	21 000
RPV height/diameter (m)	NA; Calandria
RPV weight (metric ton)	300
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Natural Uranium, once-through
Distinguishing features	Natural Uranium fuel (no enrichment); high degree of localization
Design status	Conceptual

1. Introduction

The CANDU® Small Modular Reactor (SMR) is being developed as a proven and mature design to help countries reach their goal of Net Zero. Built on proven CANDU® technology to be quickly deployable, this 300 MW(e) reactor features simplified systems, fewer components and a modular design. The design objectives are a low-cost, low-carbon power with a high capacity factor in a compact layout.

2. Target Application

A CANDU SMR is designed to enable a fast deployment using proven technology, maintaining energy independence by using natural uranium fuel from fuel manufacturers and avoiding the need to import enriched uranium fuel. This maximizes identification and utilization of a high performing supply chain, minimizing project delivery risk and creating high technology jobs in countries. The CANDU SMR (CSMR) offers a smaller and more flexible source of carbon-free electricity. The CSMR is a Generation III+ reactor with a design life of 70 years (extendable to 100 years) and a 90% capacity factor. It accommodates to a broad range of potential reactor sites, with its 0.3g seismic design, compact shape and grid-stability features.

3. Main Design Features

(a) Design Philosophy

The CSMR is based on a decades-long proven design that is licensed in many countries operating CANDU reactors. It meets modern regulatory requirements, with dedicated post-Fukushima features.
The CSMR does not require enriched fuel or a fuel-qualification program because it relies on natural uranium

in the same fuel bundle employed in multiple CANDU6® reactors. To achieve this it uses heavy water for moderation and cooling.

(b) Nuclear Steam Supply System

The CSMR is a horizontal pressure tube, pressurized heavy water reactor developed from the long lineage of successful CANDU reactors. The reactor core consists of a horizontal calandria housing a set of pressure tubes. The reactor coolant system consists of these pressure tubes, two steam generators, four primary circulation pumps plus interconnecting piping and headers. The calandria itself is filled with heavy water that surrounds the pressure tubes and provides neutron moderation plus an element of safety, with the calandria operating as a core catcher for severe accidents.

(c) Reactor Core

The CSMR core is 176 channels in a lattice of 14 rows and 14 columns. The channel array is 12’ 2” across. The calandria main shell radius is 285 cm. Reactor is fueled by standard CANDU natural uranium fuel bundle.

(d) Reactivity Control

Reactor control uses eight zone control units to provide the primary means of reactivity regulation during normal operation. The core design allows control to be achieved through very small changes in these zone control units. The adjuster rods, absorber rods and ability to use soluble poisons found in traditional CANDU designs are retained, as are two fully independent, diverse safety-shutdown systems: SDS1 and SDS2. Like its predecessors, the CSMR provides safety shutdown reliability.

(e) Reactor Pressure Vessel and Internals

The reactor consists of a cylindrical calandria and end shield assembly that is enclosed and supported by the cylindrical shield tank and its end walls. The cylindrical shield tank extension assembly closes the top of both vessels. The calandria contains the heavy water moderator and reflector; the shield tank contains light water. The seismically-qualified moderator system is independent from the pressurized heavy water heat transport system (HTS) in the fuel channel assemblies and has the inlet and outlet nozzles connected high in the calandria to enhance performance in the event of a severe accident.

(f) Reactor Coolant System

The HTS is a single figure-of-eight loop with two steam generators and four heat transport pumps. The HTS is seismically qualified such that a seismic-induced LOCA is outside of the design basis.

(g) Steam Generator

Two steam generators are used, both with inverted U-tubes and integral steam drum and preheaters, and located inside the containment structure.

(h) Pressurizer

The pressure and inventory control system include a pressurizer, bleed condenser, feed pumps (one operating; one on standby), a storage tank and control valves.

4. Safety Features

The CSMR safety features establish defence-in-depth against radiological hazards. CSMR leverages the inherent safety characteristics of the basic CANDU reactor design and supplements them with a judicious application of passive and active safety features that emphasizes provenness and results in an improvement in safety. The irradiated fuel bay is a robust, seismically qualified structure with a large volume of water relative to the decay heat load of discharged fuel, providing many days of passive cooling in the event of a loss of active heat removal.

The CSMR provides large volumes of water that are available to provide cooling to the core in the event of accidents, including by passive means. In addition, it has a large containment volume, contributing to minimizing hydrogen concentrations in severe accidents. The CSMR HTS is seismically qualified to the design basis earthquake. The CSMR also makes use of loop subdivision and feeder interlacing to reduce the rate of coolant voiding that is possible in the event of a large HTS pipe break.

The CSMR places all reactivity devices in the separate, low-temperature, low-pressure moderator, eliminating pressure-driven ejection of reactivity devices from the design. The separation of moderator from coolant also provides two separate heat removal means in the event of accidents and ensures that moderator temperature feedback to the core physics is negligible in normal operation. The characteristic CANDU pressure-tube design means that direct containment heating type of severe accident does not occur in this design.

(a) Engineered Safety System Approach and Configuration

All plant systems are assigned to one of two groups (group 1 or group 2) according to the CSMR grouping and separation approach. Group 1 includes the power production systems and delivers safety functions for group 2 support. Group 2 includes the safety related systems required for mitigation of accidents and group 2 systems are qualified or protected to provide safety functions in the event of severe external events. There is additional separation within the groups.

The CSMR has two separate shutdown systems. These are two fully-capable fast-acting means of shutdown for use at the third level of defence in depth, fully independent of each other and of the reactor regulating system which acts at the second level of defence in depth.

(b) Decay Heat Removal System

Decay heat removal is accomplished in CSMR by application of several provisions in the design, including three group 2 systems, as follows:

The large inventory in the HTS, including the pressurizer inventory, provides heat removal for normal operating transients. The HTS layout enhances natural circulation to the steam generators, which have a large inventory. The steam generators have normal make-up capability and back-up feedwater.

When the steam generators are not available, or are not effective for heat removal, heat can be removed from the HTS using the shutdown cooling system. Under emergency conditions this system can be valved in at full HTS pressure and temperature.

(c) Emergency Core Cooling System

The emergency core cooling system supplies emergency coolant to the reactor headers in the event of a loss-of-coolant accident (LOCA). The system operation is divided into two parts, short-term injection and long-term recirculation. Short-term injection consists of two stages: high pressure and low pressure injection. During the high pressure injection stage, water from the accumulator tanks is injected into the HTS by pressurized gas. After this water is depleted, low pressure injection automatically takes over, injecting water from a grade level tank via the emergency core cooling pumps. A connection is provided to this tank for demineralized water makeup, and for initial filling. For small LOCAs provision of steam generator crash cooldown and the maintenance of continued feedwater flow is used instead of the ECC heat exchangers to provide cooling. The crash cooldown is performed via the MSSVs.

(d) Containment System

The basic function of the containment system is to form a continuous, pressure-confining envelope about the reactor core and primary cooling system in order to limit the release of radioactive material to the external environment resulting from an accident. This accident could be either a failure of fuel cooling, or an accident which releases radioactive material into the containment without a rise in containment internal pressure.

To achieve this overall function, the containment system includes the following related safety functions:

i. Isolation: to ensure closure of all openings in containment when an accident occurs.

ii. Pressure/activity reduction: to control and assist in reducing the internal pressure and free radioactive material released into containment by an accident.

iii. Hydrogen control: to limit concentrations of hydrogen/deuterium within containment after an accident to prevent detonation.

iv. Monitoring: to monitor conditions within containment and the status of containment equipment, before, during and after an accident.

In addition to its safety role, the containment structure also serves the following functions:

i. To limit the release of radioactive materials from the reactor to the environment during normal operations.

ii. To provide external shielding against radiation sources within containment during normal operations and after an accident.

iii. To protect reactor systems against external events such as tornados, floods, etc.

The containment system includes a reinforced concrete containment structure (the reactor building) with a reinforced concrete dome and an internal steel liner, access airlocks, equipment hatch, building air coolers for pressure reduction, and a containment isolation system.

5. Plant Safety and Operational Performances

The CSMR design aligns with the concept of defence in depth, leveraging inherent safety features through the application of proven engineered systems with an emphasis on providing high reliability through a prudent mix of active and passive features. The overall CSMR design philosophy is to reduce total unit energy cost by reducing specific capital cost, shortening the construction schedule, reducing operating, maintenance and administration costs and providing for plant life extension. In addition, CSMR enhances or improves the traditional CANDU advantages including real safety, low man-rem exposure, high capacity factor and ease of maintenance. Proven systems, system parameters, components, and concepts are used, including proven technologies from other industries.

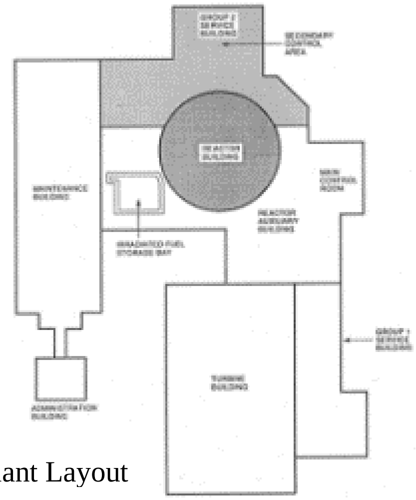
6. Instrumentation and Control Systems

The overall I&C architecture and design is aligned with the five levels of defense-in-depth, with suitable independence between the different levels. Most automated plant control functions are implemented in a modern distributed control system (DCS) using a network of modular, programmable controllers that communicate with one another using reliable, high-security data transmission methods. The plant is automated to the extent that requires a minimum of operator actions for all phases of operation.

The control systems are backed up by the safety systems, which include the two independent shutdown systems, the emergency core cooling system, emergency heat removal system, and the containment system. Each of these safety systems operates completely independent of the other and independent of the reactor and process control systems.

7. Plant Layout Arrangement

The CSMR layout is arranged to minimize and achieve a short practical construction schedule. This is achieved by simplifying the layout, optimizing interfaces, reducing construction congestion, providing access to all areas, providing flexible equipment installation sequences and reducing material handling requirements. The principal structures of the CSMR include the reactor building, the reactor auxiliary building, the turbine building, the group 1 service building and maintenance building. The auxiliary structures include the Group 1 pump house (main pump house), Group 2 pump house and administration building. In general, Group 1 systems sustain normal plant operation and power production, while Group 2 systems have a safety or safety support functions.



Plant Layout

8. Design and Licensing Status

Candu Energy Inc. has requested the CNSC to initiate the formal Vendor Design Review (VDR) process for the CSMR design, which will be conducted based on the regulatory requirements applicable to the new build projects in Canada. The design of CSMR is at the Conceptual Design stage. The current scope is to confirm the safety and OPEX driven design changes affecting NSP in support of licensing review, and to finalize the testing schedule to support Preliminary Safety Analysis Report (PSAR) development and detailed design.

9. Fuel Cycle Approach

The standard CSMR uses a once-through fuel cycle based on natural uranium, producing a very low residual-uranium spent fuel with low heat generation. It is also capable of burning recovered uranium from light-water reactors, making it a valuable addition to any light-water fleet. The use of MOX and Thorium fuels are also possible with suitable customization.

10. Waste Management and Disposal Plan

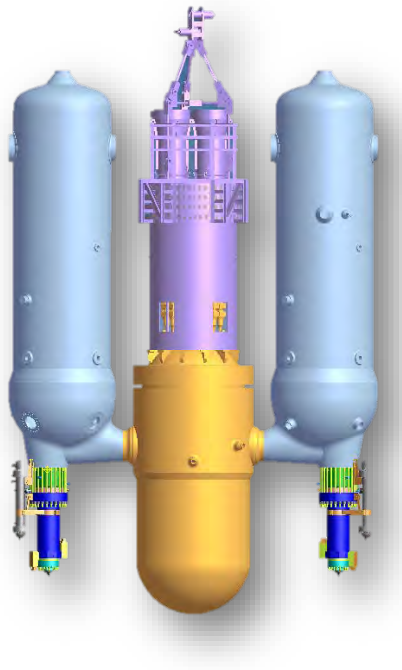
The CSMR implements systems and equipment for handling radioactive wastes consistent with the state-of-the-art in Canada. The radioactive waste management systems are designed with the objective of limiting routine releases in accord with the ALARA principle. Facilities are provided for interim storage, or controlled release, of all radioactive gaseous, liquid and solid wastes.

It is anticipated that most of the of low-level and intermediate-level solid waste produced over the reactor’s 70-plus year lifetime will be stored at the waste management area, with high-level waste being stored on-site in a dry storage facility. Candu Energy Inc. is working on establishing the plan to safely store, handle and dispose of all the spent fuel including the on-site storage, long term storage and potential design for new long-term storage containers that meet the requirements for the deep geological repository design of the Nuclear Waste Management Organization (NWMO).

11. Development Milestones

2020	Conceptual Design complete
2022	Detailed Design complete
2025	First Concrete for first unit
2028	First Unit in service

CAP200 (SNERDI/SPIC, China)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	SNERDI/SPIC, China
Reactor type	PWR
Coolant/moderator	Light water/ Light water
Thermal/electrical capacity, MW(t)/MW(e)	660/ >200
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15.5 / -
Core Inlet/Outlet Coolant Temperature (°C)	289 / 313
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies in the core	89
Fuel enrichment (%)	4.2 (Average)
Core Discharge Burnup (GWd/ton)	37 (Average)
Refuelling Cycle (months)	24
Reactivity control mechanism	Control rod drive mechanism and soluble boron
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	--
RPV Height/Diameter (m)	8.845 / 3.280
Seismic Design (SSE)	0.30 (g)
Distinguishing features	Compact layout; Passive safety; Underground containment
Design status	Conceptual design finished

1. Introduction

The China Advanced Passive pressurized water reactor 200MW(e) (CAP200) is one of the serial research and development products of PWRs adopting passive engineered safety features initiated by Shanghai Nuclear Engineering Research and Design Institute (SNERDI). The design of CAP200 is based on more than 45 years of experience in PWR R&D, and more than 20 years of experience in PWR construction and safe operation 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 the Fukushima event. Figure 1 shows reactor system configuration of CAP200.

2. Target Application

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

3. Main Design Features

(a) Design Philosophy

CAP200 is a small PWR which is designed with improved safety, flexibility and environmental friendliness, and is also comparable with other SMRs on economy. Compared with large PWRs, CAP200 has a number of advantages such as higher inherent safety, lower frequency of large radioactivity release, longer time of post-accident no 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 primary system: Steam generators (SGs) are connected to reactor pressure vessel (RPV) directly

and main pipes are eliminated. Compact layout of system and components results in lower risk and possibility of loss of coolant accidents and smaller primary system footprint.
- Modular design and fabrication: main modules can be fabricated in factory and transported to the site for installation. Construction period can be shortened because of high modularization.
- Redundant and diverse safety features: redundant and diverse active and passive safety features are deployed, which ensures the reactor core safety and extremely low risk of large radioactivity release.

(b) Nuclear Steam Supply System

The NSSS of CAP200 consists of reactor pressure vessel (include IHP), steam generators, reactor coolant pumps, pressurizer and auxiliary systems. Due to the cancellation of RCS pipe, 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 for PWR required 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 case of CAP200, to connect RPV and SGs replacing main pipes. The vessel is cylindrical with a removable flanged hemispherical upper head and hemispherical bottom head. The area with highest neutron fluence of active core region of the vessel is completely forged and free of welding which enhances the confidence of 60 years’ design life and decreases the task of in service inspection. As a safety enhancement, neutron and temperature detectors enter the reactor through upper head of reactor vessel which eliminates penetrations in the lower head of RPV. This reduces the possibility of a loss of coolant accident (LOCA) and uncovering of the core The core is positioned as low as possible in the vessel to decrease re-flood time in the event of an accident. Furthermore, this arrangement is very helpful for successful execution of IVR. 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 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 with electrical heater at the bottom and spray at the top. The pressure control is steady and reliable. The pressurizer is a vertically mounted cylindrical pressure vessel with hemispherical top and bottom heads which adopts the traditional design that is based on proven technology. The pressurizer volume is designed to be large which increases margins for transient operation and minimizes unplanned reactor trips and provides higher reliability. In addition, fast-acting power-operated relief valve, which is one of the reactor coolant system leakage sources and the component requiring potential maintenance, won’t be needed because of the improved transient response by using large volume pressurizer.

(h) Reactor Coolant Pump

Both leak-tight canned motor pump or wet coil pump are possible choices for the RCP of CAP200. There is rich experience of use of canned motor pump and wet coil pump in nuclear power plants, and the current advanced large nuclear reactors are also employing canned motor pump or wet coil pump. The RCP is designed to produce a head of 65m at design flow rate of 12000 m^3/h with a cold leg temperature of 289℃. The reactor coolant pump has no shaft seals, eliminating the potential 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.

4. Safety Features

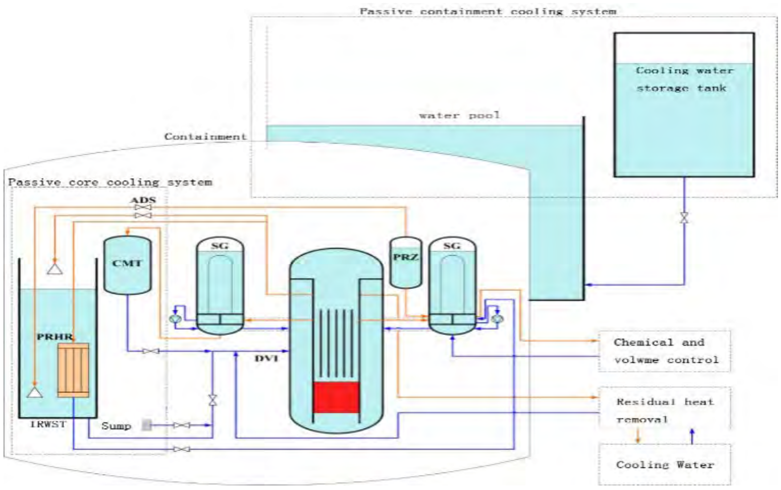
CAP200 adopts passive safety systems which take advantage of natural forces such as natural circulation, gravity and compressed air to make the systems work, offering improvements for plant in simplicity, safety, O&M, availability and investment protection. No active components such as pumps, fans and 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.

(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 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, as shown in the above figure. 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 centers is crucial to the plant's capability to remove heat by natural circulation following a loss of forced circulation. For CAP200, these two components are vertically separating 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. Electric Power Systems

Electric power system has three levels to satisfy the defense-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.

8. 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 of CAP200 is designed to be capable of different types of extraction, and which enables CAP200 to meet different levels of heating and industrial steam supply requirements.

9. Plant Arrangement

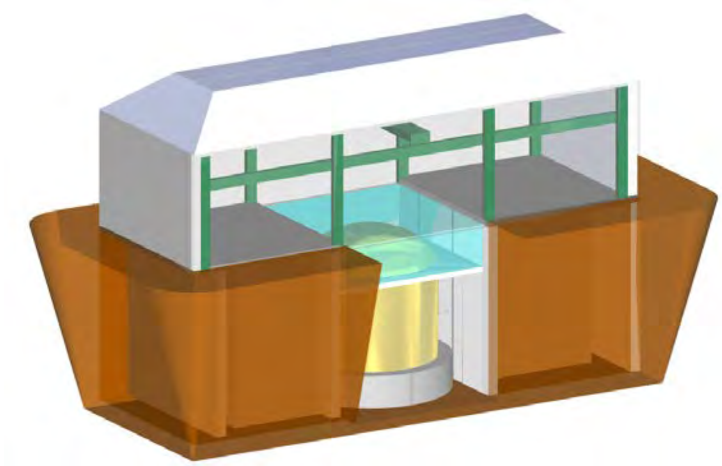
(a) Reactor building

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 and auxiliary building, etc.



10. Design and Licensing Status

Conceptual design finished

11. Development Milestones

2014	Conceptual design started
2015	Conceptual design finished
2018	Further deepen conceptual design

DHR400 (CNNC, China)

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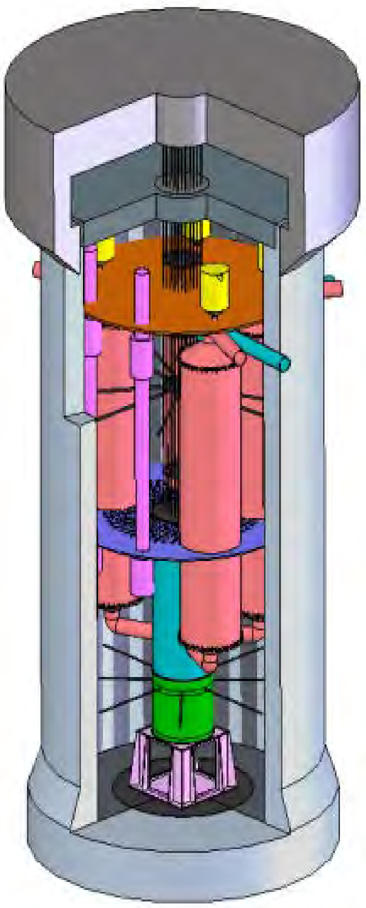


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	China National Nuclear Corporation (CNNC), China
Reactor type	Pool type reactor
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	400 / *does not produce electricity*
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.3 (Core inlet pressure)
Core Inlet/Outlet Coolant Temperature (°C)	68 / 98
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	69
Fuel enrichment (%)	< 5.0
Core Discharge Burnup (GWd/ton)	30
Refuelling Cycle (months)	10
Reactivity control mechanism	Control rod drive mechanisms
Approach to safety systems	Inherent safety features with large water volume in the reactor pool
Design life (years)	60
Plant footprint (m^2)	40 000
Pool depth/diameter (m)	26 / 10
Seismic Design (SSE)	0.3g
Distinguishing features	Coupling with desalination and radioisotope production
Design status	Basic design

1. Introduction

The DHR400 is a pool type District Heating Reactor with a thermal power of 400 MW. The DHR400 operates under low temperature in atmospheric pressure. Established light water reactor technologies are used in the design of DHR400. Inherent safety features are incorporated to enhance its safety and reliability. Significant design features include: Large water inventory in the pool that provides large thermal inertia and a long response time, thus enhances the resistance to system transients and postulated accidents. It has low probability of core meltdown that eliminates the large radioactivity release and simplifies the off-site emergency response. Simplification in designs and convenience in maintenance lead to the improvement in economics. With high reliability and inherent safety features, DHR400 can be located in the 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. Main Design Features

(a) Design Philosophy

The DHR400 is designed based on the pool type research reactor. It operates under low temperature in atmospheric pressure above the water surface. With the deep 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) Nuclear Steam Supply System

Not Applicable (N/A).

(c) Reactor Core

The core of DHR400 consists of 69 fuel assemblies. Each fuel assembly is 2.1 m long and its design is modified from the 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 U_{235} enrichment is below 5%. 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.

(d) 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.

(e) 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 2.0 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 2m 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.

(f) Primary Coolant System

During normal operations, a 68°C inlet water enters the core from the bottom, the water is then heated to a 98°C. 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°C 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.

(g) 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.

(h) Primary Heat Exchanger

The 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 advantages to radioactivity isolation.

(i) Primary Heat Cooling System

The residual heat cooling system of DHR400 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°C 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) Engineered Safety System Approach and Configuration

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

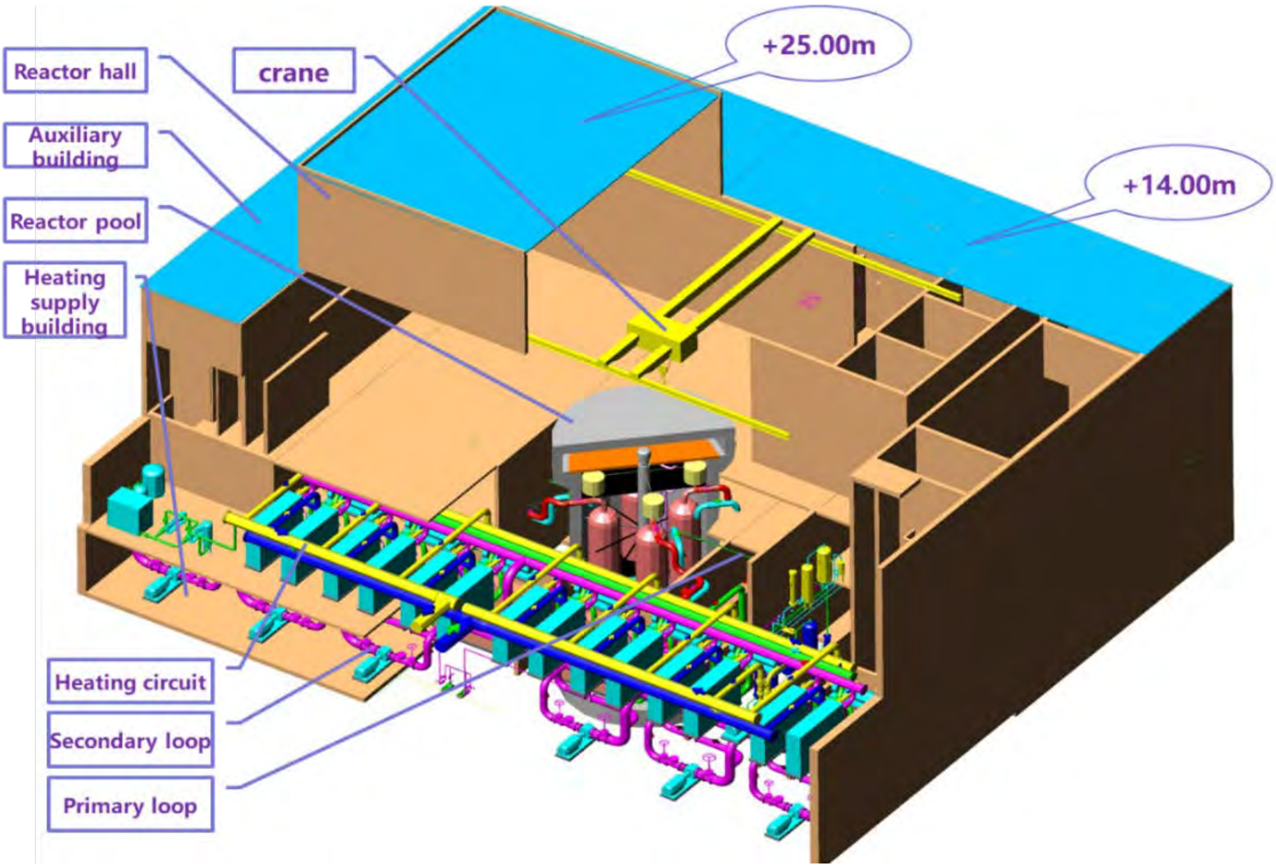
A waste gas treatment system is introduced to store and decay short life gaseous fission products and to absorb krypton and xenon isotopes.

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 Layout 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

HAPPY200 (SPIC, China)

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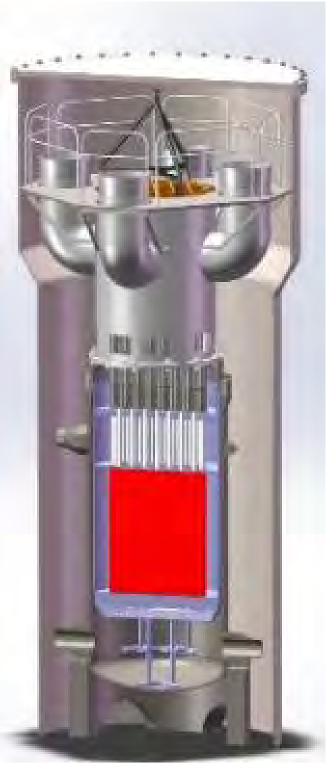
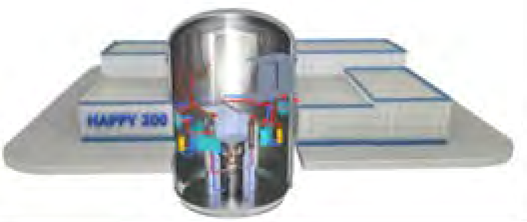




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	State Power Investment Corporation, Ltd. (SPIC), People's Republic of China
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	200 / 0 (thermal power only)
Primary circulation	Forced (2 pumps)
NSSS Operating Pressure (primary/heating), MPa	0.6 / 0.8
Core Inlet/Outlet Coolant Temperature (ºC)	80 / 120
Fuel type/assembly array	UO_2 / Square 17x17
Number of fuel assemblies in the core	37
Fuel enrichment (%)	2.76 avg / 4.45 max
Core Discharge Burnup (GWd/ton)	40
Refuelling Cycle (months)	18
Reactivity control mechanism	Rods
Approach to safety systems	Active / Passive
Design life (years)	60
Plant footprint (m^2)	1150
RPV height/diameter (m)	6.0 / 2.25
RPV weight (metric ton)	26
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	LEU / once through
Distinguishing features	Pool-loop combined type reactor, heat generator.
Design status	Detailed Design

1. Introduction

The Heating-reactor of Advanced low-Pressurized and Passive SafetY system – 200 MW(t) (HAPPY200) is a so called pool-loop combined type reactor, which has both features of swimming pool reactor and PWR to some extent. The HAPPY200 operates under low temperature and at low pressure in the closed primary circuit boundary with forced circulation mode. All engineered safety systems operate in a passive mode, and can cope at least for 1 month after accident without any active operator intervention, taking advantage of external cold air as the ultimate heat sink. A large pool of water inventory is incorporated to enhance its safety and reliability under postulated accident conditions. The key design objectives of HAPPY200 include: inherent safety; good economy; proven technology; and easy decommissioning. Aiming for high reliability with inherent safety features, HAPPY200 can be deployed in the vicinity of the targeted heating supply district or community with high population density.

2. Target Application

The HAPPY200 is designed for heat generating power source, dedicated to provide northern cities in China with a clean heating solution, and can be operated for 4-8 months each year during the winter. Its non-electric applications include sea water desalination, house cooling in summer, energy storage etc. without any need of substantial design change.

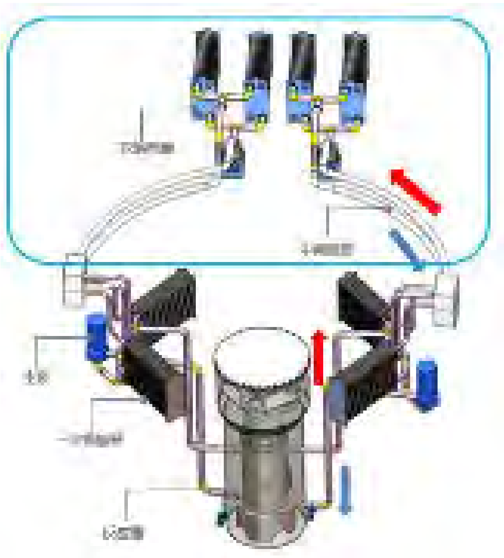
3. Main Design Features

(a) Design Philosophy

The HAPPY200 draws on the mature operating experience of light water-cooled reactors, pool reactors and passive nuclear safety technology. HAPPY200 adopts proven equipment, such as shortened fuel assembly, plate-type heat exchanger, etc. This equipment has full operational record, high reliability and high maintainability. The primary safety objective of HAPPY200 design is practical elimination of core melting and technical cancellation of off-site emergency, so that the reactor can meet basic evaluation principle for heating reactor formulated by the Chinese National Nuclear Safety Administration.

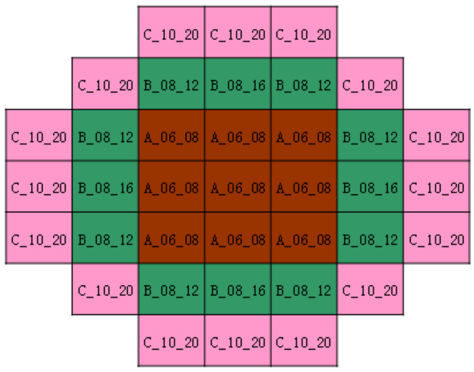
(b) Primary Circuit

During normal operation, the 80°C inlet water enters into the core from the bottom, and the water is then heated to a temperature of 120°C. The outlet water enters the 2 hot legs and then enters four separate primary heat exchangers. Inside the primary heat exchanger, the primary water is cooled to 80°C by the secondary side water. The water flowing through the primary exchangers enters 2 cold legs separately. The water then flows down to the bottom and enters the core again. No boiling will occur during normal operations.



(c) Reactor Core and Fuel

The core of the HAPPY200 consists of 37 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. The fuel is UO_2 with Gd as a burnable absorber. The U_{235} enrichment is below 5%. The reactor operates 180 days per year (typical 6-month winter period in northern China) and a three-batch refuelling is conducted off power on an 18-month refuelling cycle.



(d) Secondary Side

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 could contaminate the third loop.

(e) Reactivity Control

HAPPY200 uses 21 control rod clusters to provide enough reactivity compensating capacity, with magnetic force type and hydraulic force type control rod driving mechanism (CRDM). No chemical shim (e.g. Boron) is used for reactivity control. HAPPY200 blends a lot of gadolinium oxide in fuel rods. Because there is non-soluble boron in the core, the reactor is operated by shifting control rods to maintain criticality.

(f) Reactor Pressure Vessel and Internals

The reactor pressure vessel is submerged inside the large water pool, isolated from the pool during normal operation and other conditions, RCS is connected to pool only if the RCS is depressurized and need injection of cooling water from the pool.

4. Safety Features

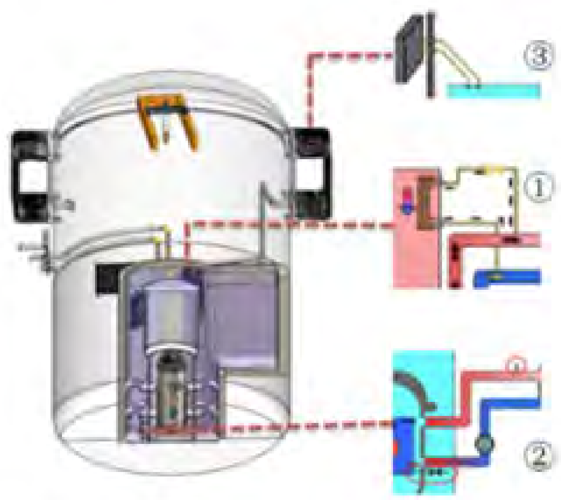
The primary safety objective of HAPPY200 design is practical elimination of core melting and technical cancellation of off-site emergency. To achieve this safety objective, the safety concept of HAPPY200 is based on inherent safety features, the defence in depth principle, the use of passive systems to prevent accidents and mitigate their consequences, and multi-barriers to the release of radioactive materials into the environment.

(a) Engineered Safety System Approach and Configuration

HAPPY200 has many characteristics including low temperature, low pressure parameters, high thermal safety margin, negative power reactivity, simplified engineered safety features (ESF), etc. And the system uses passive cooling system and anti-seismic system. The safety systems of HAPPY200 consist of: redundant shutdown system, passive feed-bleed system (PFB), passive residual heat removal system (PHR), passive pool air cooling system (PAC), etc. These systems could maintain core integrity through the plant lifetime.

(b) Decay Heat Removal System

The passive residual heat removal system is designed to remove the residual heat from the reactor core during unexpected operational occurrences and events, especially for the non-LOCA events. It consists of two series vertical tube heat exchangers connected with the hot legs and cold legs of each loop. The passive residual heat removal mode is provided by the natural circulation of the coolant in primary circuit through the heat exchangers, which use the large capacity pool water as the heat sink. The heat sink capacity is adequate to passively cool down the reactor and prevent hazardous superheating of the core.



Engineered safety features:
-1 Passive Residual Heat removal System
-2 Passive Safety Injection System
-3 Passive Air Cooling System

(c) Emergency Core Cooling System

Pool water heat sink is used for emergency core coolant injection and residual heat removal.

(d) Containment System

The containment of HAPPY200 is a steel shell type, acting as the last barrier of fission product. It also partially functioned as heat removal sink to environment during normal operation and accident condition.it is partially deployed underground. A total of 5 safety barriers: fuel pellet, fuel cladding, primary circuit system, large volume shielding pool and steel containment.

5. Plant Safety and Operational Performances

HAPPY200 unit itself does not consider generating electricity. In normal operation, external power supply is required to ensure the operation of each device and system. In accident conditions, the unit can be returned to the shutdown state and take away the decay heat without relying on the external power supply by the inherent safety features and passive safety system.

6. Instrumentation and Control Systems

The instrumentation and control (I&C) system provides the capability to monitor, control and operate plant systems. The I&C system is implemented using mature technology based on high economy.

7. Plant Layout Arrangement

HAPPY200 is designed to be sited in inland areas near the targeted heating supply area centre. No special requirements to adapt with the local air temperature, humidity and other conditions. The principle structures are the reactor building, electrical building, fuel building and auxiliary building. HAPPY200 unit itself does not generate electricity. In normal operation, external power supply is required to ensure the operation of each device and system. In accident conditions, the unit can be returned to the shutdown state and take away the decay heat without relying on the external power supply by the inherent safety features and passive safety system. HAPPY200 does not discharge waste heat or require a large amount of cooling water, because the plant design adopts closed cycle. By using closed cycle, the amount of water replenished in normal operation is small and all of the heat is supplied to the heat consumer. HAPPY200 with smaller or zero emergency planning zones (EPZ) up to only 3 km in radius is expected to be approved by the regulators.

(a) Engineered Safety System Approach and Configuration

The reactor building is used to prevent the radioactive materials escaping to the environment at the condition of LOCA accident. At normal conditions and accident conditions, it provides radiation protection and protects the internal systems from external disasters. The reactor building is mainly used to arrange reactors and other primary loop equipment, such as main pump, primary/secondary heat exchanger, pressurizer, shielding pool, timeless air cooling system, chemical and volume control system, equipment cooling water system, NI ventilation and air conditioning system, etc.

(b) Electrical building

The electrical building is mainly used to arrange power distribution equipment, instrumentation and control equipment, main control room, battery, ventilation system, fire protection system, and so forth.

(c) Fuel building

The fuel building is mainly used for equipment layout and operation of fuel handling, transportation and storage systems. It is also used to arrange pool cooling and purification system, ventilation and air conditioning system, chemical and volume control system, etc.

(d) Auxiliary building

The liquid waste treatment system is performed at the auxiliary building. The system includes the radioactive liquid waste recovery system, nuclear sampling system, exhaust gas treatment system, and nuclear auxiliary building ventilation system.

8. Design and Licensing Status

HAPPY200 has completed conceptual design. Preliminary design is underway. The commercial demonstration project is carrying out preliminary work of the project, and the site selection and preliminary feasibility analysis have been completed. HAPPY200 meets regulatory requirements for design and licensing in most countries. The first project has completed the site selection and preliminary feasibility analysis report review. The next step will submit site safety assessment report and site stage environmental impact assessment report to the Chinese regulatory authorities

9. Fuel Cycle Approach

Because in the north of China, heating system is required to work about six months in a year, the HAPPY200 reactor should be operated about 18 months in the 3-year cycle, the reactor should be shutdown for six months every year. However, there is no need for refuelling in one cycle for 3 years. The enrichment of uranium used by HAPPY200 is less than 5%, and the average discharge burnup of fuel assemblies is about 40 GWd/tU.

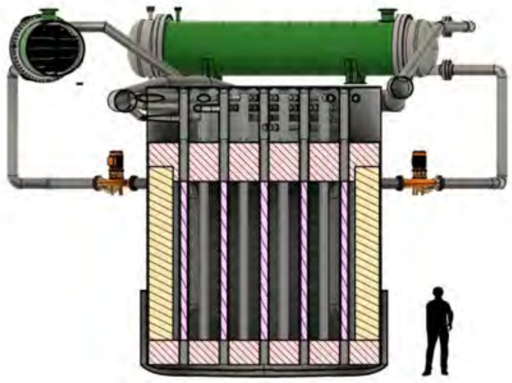
10. Waste Management and Disposal Plan

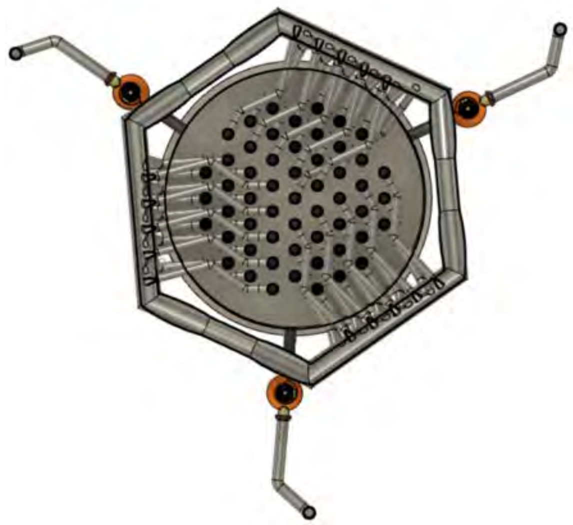
After spent fuel discharge from the reactor, it is stored on plant site and then cooled and decrease in radioactivity for some years. Once-through fuel cycle option is considered after plant site decommissioning. Spent fuel leave at safety level and transfer to specific disposal site.

11. Development Milestones

2015	Start market investigation and concept design (changes)
2016	Concept design completed
2019	Start pre-project work in north of China
2022	To start construction of a proto-type NPP in China
2024	Target commissioning and commercial operation dates

TEPLATOR™ (UWB Pilsen & CIIRC CTU, Czech Republic)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	UWB Pilsen & CIIRC CTU Prague, Czech Republic
Reactor type	Channels in Reactor Vessel
Coolant/moderator	Heavy Water (D_2O)/ Heavy Water (D_2O)
Thermal/electrical capacity, MW(t)/MW(e)	50 / does not produce electricity
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Ambient/Ambient
Core Inlet/Outlet Coolant Temperature (oC)	45 / 98
Fuel type/assembly array	VVER-440 / hexagonal with 126 fuel pins
Number of fuel assemblies	55
Fuel enrichment (%)	Spent fuel (< 1.2 wt% U_{235} equivalent)
Core Discharge Burnup (GWd/ton)	2.3
Refuelling Cycle (months)	10 months with online option
Reactivity control mechanism	Moderator height, Control blades
Approach to safety systems	Inherent and passive safety with built/in decay heat sink
Design life (years)	60
Plant footprint (m2)	≤ 2000
RPV height/diameter (m)	6.5 / 3.7
RPV weight (metric ton)	N/A
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	LEU - reuse of LWR spent FAs, possibility to run on fresh SEU (≤1.2% U_{235})
Distinguishing features	District heating zero CO_2 source with zero fuel cost, low pressure.
Design status	Conceptual design

1. Introduction

The TEPLATOR is the innovative concept for future district and process heat production. The TEPLATOR facility will use already irradiated fuel from conventional light waters reactor (which are not burnt up to its regulatory and design limits). In order to harvest additional energy from already used FAs, the TEPLATOR is a critical assembly derived by the state-of-the-art computational tools using better moderation, more optimal fuel lattice pitch, lower fuel temperature and lower coolant pressure. Different TEPLATOR variants are proposed; using either used BWR, PWR or VVER irradiated fuel assemblies with output power range between 50 and 200 MW(t). The initial TEPLATOR operates at 50 MW(t) with VVER440 fuel.

2. Target Application

The TEPLATOR is designed for clean district heating energy production for cities with 100 000 or more inhabitants. It will replace the out-dated conventional heating plants based on fossil fuels. The TEPLATOR will produce heat without any emissions and with negligible fuel costs. TEPLATOR solutions are especially suitable for countries that have thousands LWR FAs stored either in interim storage casks or spent fuel pools. These FAs are now financial liability which, once used for heat production, can turn into a sizeable financial asset. The calculated investments cost for the first TEPLATOR DEMO 50 MW(t) facility is 30 M EUR. Then the final price of produced heat is 4 EUR/GJ (using prices of 2019).

3. Main Design Features

(a) Design Philosophy

The design philosophy is to use only proven, known, verified, and tested high TRL components. This ensures low investments costs and low risks. The design itself includes 3 circuits. The primary circuit includes a so-called calandria, a core with the spent LWR fuel FAs, three heat exchangers and three pumps. The core is made from Zr channels in which the fuel is inserted. The space between the channels is filled by the moderator, heavy water. The coolant flows in all channels, through a system of pipes to the collector. Three pipes are led out of this collector, each of which is led into a separate heat exchanger. The coolant passes through the primary side of the heat exchanger and returns to the fuel channels through the pump and the lower distribution chamber. A secondary or intermediate circuit transfers the heat from the primary circuit to the district heating circuit. The secondary circuit heat transfer fluid (HTF) could either be water or another fluid (based on the operating parameters). The intermediate circuit includes two storage tanks connected to the circuit serving as an energy storage system for shaving off demand peaks. These storage tanks are also able to simultaneously dissipate and store heat from the residual power of the fuel, i.e. the intermediate tanks are designed to be able to absorb decay heat of the core in DBAs. The tertiary or district heating circuit, which distribute the heat to the end customer, is therefore separated from the core by two sets of heat exchangers.

(b) Nuclear Steam Supply System

The TEPLATOR operates only with liquid phase, no steam generation or electricity production is expected.

(c) Reactor Core

The TEPLATOR core consists of equally spaced channels filled with spent nuclear fuel from LWR reactors. More customization options are possible, the initial one is the reuse of VVER-440 spent nuclear fuel. In that case, totally 55 fuel assemblies are placed in large-pitched hexagonal array. Typical VVER-440 spent nuclear fuel had 3.6 wt% U_{235} initial enrichment, 35 GWd/ton average burnup and 30 years of cooling. Alternative use of slightly enriched fresh fuel (< 1.2 wt% U_{235}) is possible. Each fuel assembly is placed in a coolant tube filled with heavy water or alternatives for temperatures up to 98ºC. Atmospheric pressure of heavy water moderator eliminates the need for a thick and expensive pressure vessel. The TEPLATOR is a heat generator with typical operation up to 10 months each year (typical heating season) with an option to be refuelled online.

(d) Reactivity Control

Two independent systems are deployed. Reactivity control under normal operation is achieved by changes in moderator level in the reactor pool. Safety-shutdown system is based on three borated steel blades that can be dropped in the core. Due to the low temperatures, relatively short cycle and use of spent fuel the excess reactivity is quite small.

(e) Calandria and Internals

The TEPLATOR internals consist of the fuel channels, channel outlets, absorber blades, absorber blades drive mechanism, I&C systems, reflector and bottom collector. Through the bottom collector the coolant (heavy water) is distributed back to individual channels. The calandria is a stainless-steel vessel, since the TEPLATOR works on low pressure, it does not need to be very thick. The space between fuel channels and calandria is filled with heavy water which serves as a moderator, the total volume of heavy water in calandria is around 30 m^3. The core is surrounded by a graphite reflector.

(f) Reactor Coolant System

The primary coolant (D_2O) enters the core of TEPLATOR with the temperature of 45°C. It flows through the fuel channel and then it leaves the individual channels at 98 °C at the channel outlet. This outlet is attached to the collector where the primary coolant is collected. From the collector the coolant is distributed to the three heat exchangers where it heats the secondary heat transfer fluid (HTF). The primary coolant flows through the pump, then through the pipe on the inside of the calandria to the bottom collector where it is distributed again to the individual channels. Roughly 20 m^3 of D_2O is required in the primary circuit.

(g) Primary Heat Exchanger

The TEPLATOR is a three-loop design, thus it has three primary heat exchangers (HE) to transfer the heat from the primary to the secondary circuit. The heat exchanger is a horizontal type with U-shaped tubes and water-water heat exchange. Each of the HE has a heat transfer surface about 520 m^2 and is capable, under forced circulation, of cooling the TEPLATOR core on its own: decay heat under emergency conditions can be safely removed by HE to the energy storage tanks using natural circulation.

(h) Pressurizer

As the TEPLATOR operates under ambient pressure, the function of a PWR pressurizer is replaced by a volume compensator attached to the primary circuit. The compensator is linked to the heavy water management systems.

(i) Secondary side

The secondary circuit is an intermediate loop that separates the primary circuit and the tertiary circuit while transferring heat from the primary to the third circuit. The secondary circuit consists of the secondary side of primary heat exchangers (HE I) and the primary side of secondary heat exchangers (HE II). As part of this circuit the energy storage system, consisting of two tanks, can be connected having identical heat transfer fluid (HTF) as the secondary HTF. This energy storage system is based on thermal energy storage (TES) heat mechanism which serves several purposes: 1) TEPLATOR power fluctuations, 2) compensation and smoothing of the demand curve and 3) emergency and safety heat sink.

4. Safety Features

General safety features

The TEPLATOR operating conditions (e.g., fuel/coolant temperature, pressure, linear heat rate) are much lower than those for which the used FAs were certified and used in LWRs. The safety features establish defence-in-depth against radiological hazards. The TEPLATOR leverages the inherent safety characteristics of the basic LWR reactor design and supplements them with passive and active safety features that emphasizes provenness and results in an improvement in safety.

The TEPLATOR secondary circuit provides large volumes of fluid that are available to provide cooling to the core in the event of accidents, including by passive means.

The TEPLATOR places all reactivity devices in low-temperature, low-pressure moderator, eliminating pressure-driven ejection of reactivity devices from the design. The separation of moderator from coolant also provides two separate heat removal means in the event of accidents and ensures that moderator temperature feedback to the core physics is negligible in normal operation.

(a) Engineered Safety System Approach and Configuration

The TEPLATOR has two separate shutdown systems. These are two fully-capable fast-acting means of shutdown for use at the third level of defence in depth, fully independent of each other.

(b) Decay Heat Removal System and Emergency Core Cooling System

Decay heat removal system is integrated as the energy storage system interconnected to the secondary circuit. During TEPLATOR shut down, heat generated in the core is transported by natural circulation inside the cooling loops. This heat is removed in the primary heat exchanger using thermal energy storage (TES). The TES system consists of two tanks, a ‘cold' and a 'hot’ one. In order to remove decay heat from the TEPLATOR, heat transfer fluid (HTF) from the cold tank flows via natural convection through the secondary side of the primary heat exchanger (HE I) to the hot tank. The volume of both tanks is designed to be sufficient for removing decay heat for long enough that the auxiliary cooler dissipates the heat.

(c) Containment System

The TEPLATOR containment system includes a reinforced concrete containment structure (the reactor building) with a reinforced concrete dome and an internal steel liner, access airlocks, equipment hatch, building air coolers for pressure reduction, and a containment isolation system.

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 large negative temperature reactivity coefficient.

6. Instrumentation and Control Systems

The instrumentation and control (I&C) system provides the capability to monitor, control and operate plant systems. The I&C system is implemented using mature technology; i.e., known and tested LWR detectors and systems.

7. Plant Layout Arrangement

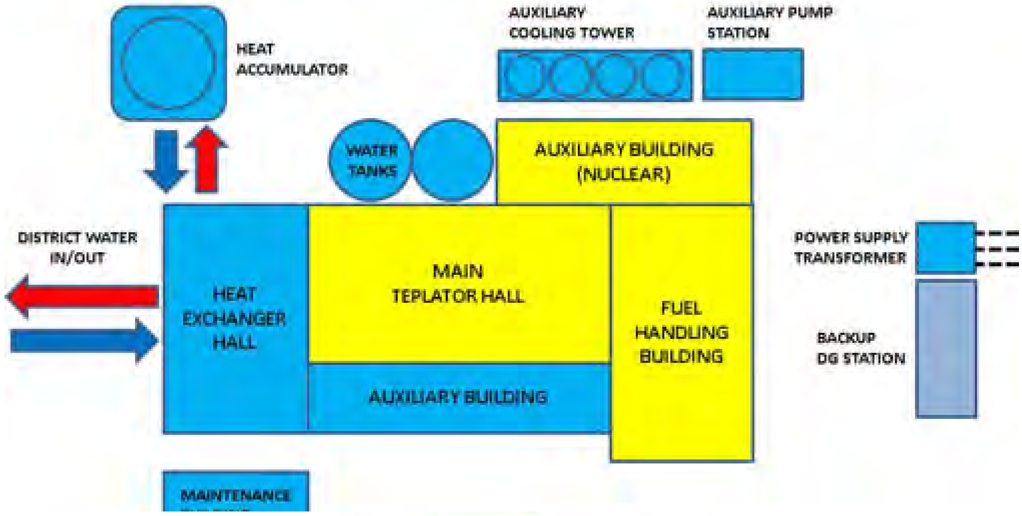
The plan layout of the TEPLATOR is illustrated below.

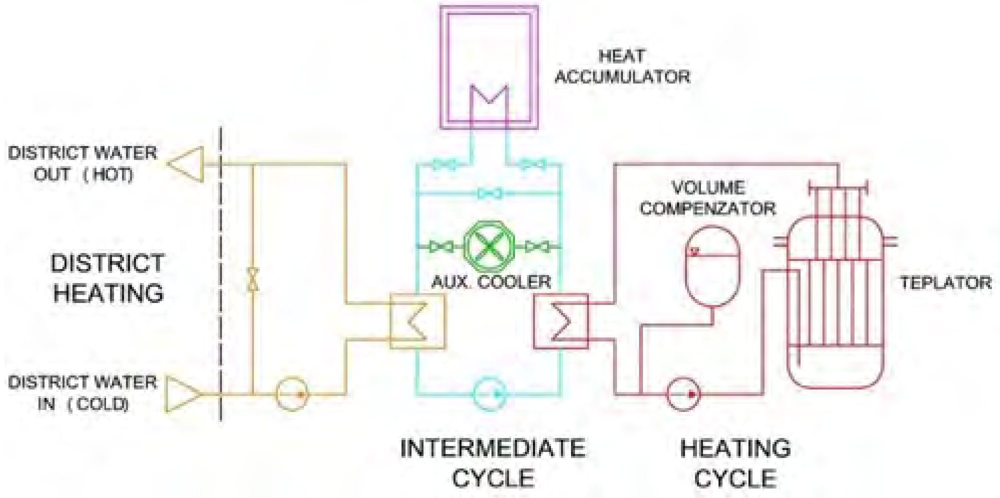
(a) Reactor Building

The TEPLATOR facility consists of one main structure which further contains nuclear and non-nuclear sectors/buildings. Nuclear sectors are the main TEPLATOR hall, the fuel handling building, and the auxiliary nuclear building. Non-nuclear sectors are the heat exchanger hall and the auxiliary building. Other buildings and structures within the facility layout are a heat accumulator, water storage tanks, auxiliary cooling towers with a pumping station, transformers of power supply and backup diesel generators.

(b) Balance of Plant

Heating / Chilling Supply system is located in the heating exchanger hall next to the main TEPLATOR hall. An option to use the TEPLATOR heat source for district chilling solutions is also investigated.





8. Design and Licensing Status

The TEPLATOR project completed its preconceptual design and the works on preliminary/basic design will start in the Q4 of 2020. The commercial demonstration unit with thermal power of 50 MW is in the preliminary phase. The preliminary phase includes the feasibility study, the site location selection and obtaining the license for construction. Once the feasibility study is done and the site location is approved, the environmental impact assessment report will be carried out and will be submitted to the regulatory authorities.

9. Fuel Cycle Approach

The unique feature of TEPLATOR is the reuse of spent nuclear fuel from commercial LWRs which is normally considered a waste. This can be achieved due to significantly lower operation parameters (far from limits) when compared to the conditions in large LWRs. Based on heating or cooling demand, the core can be operated up to 10 months each year with subsequent refuelling of fuel assemblies. Online refuelling is optional as well as usage of fresh SEU assemblies.

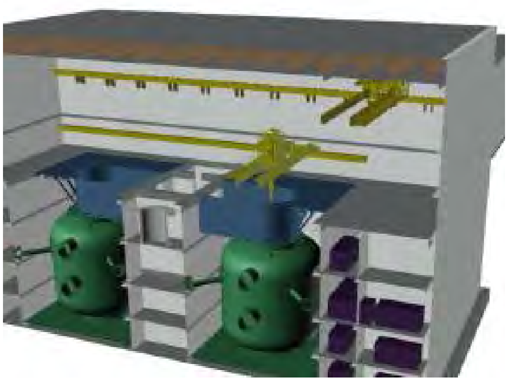
10. Waste Management and Disposal Plan

When removed from the core, reused fuel will be stored and cooled in the fuel handling building and thereafter transported back to the original spent fuel storage.

11. Development Milestones

2019- 2020	Preliminary studies and pre-conceptual design
2020-2021	Conceptual design phase and technology validation
2021-2023	Design phase - Basic design Detail design / Licensing
2024-	Start of DEMO unit construction, first unit in service at 2027

NUWARD™ (EDF Consortium, France)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	EDF-led consortium with CEA, Naval Group, and TechnicAtome, France
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	2x540 / 2x170
Primary circulation	Forced circulation
Operating Pressure (primary/secondary), MPa	15 / 4.5
Core Inlet/Outlet Coolant Temperature (°C)	280 / 307
Fuel type/assembly array	UO_2 / 17x17 square pitch arrangement
Number of fuel assemblies in the core	76
Fuel enrichment (%)	<5
Core Discharge Burnup (GWd/ton)	-
Refuelling Cycle (months)	24
Reactivity control mechanism	Control rod drive mechanism (CRDM), solid burnable poisons
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	3500, nuclear island including fuel storage pool
RPV height/diameter (m)	13 / 4
RPV weight (metric tonnes)	310
Seismic Design (SSE)	0.25g
Distinguishing features	Highly compact NSSS and containment, boron-free design, load follow
Design status	Conceptual Design

1. Introduction

NUWARD™ is being developed as a Generation III+ SMR NPP to generate a total net electrical power of 340 MW(e) from two independent reactor modules to offer flexible operation. It is an integral-PWR that fully integrates the main components of the Nuclear Steam Supply System (NSSS) including control rod drive mechanism, steam generators and pressurizer within the Reactor Pressure Vessel (RPV). Adopting shortened RPV, the NSSS is installed in a steel containment submerged in the underground water wall allowing for enhanced in-factory manufacturing. NUWARD™ specific design includes the management of all Design Basis Conditions (DBC) through passive systems without the need for any external classified electrical power supply and is self-reliant on an internal ultimate heat sink (the water wall) for a coping time of more than 3 days.

2. Target Application

The design enables easy integration in any electrical grid, as replacement of fossil-fuel fired power plant or as a complement to renewable capacities. NUWARD™ offers baseload or load-following capacities. To account for seasonal variations, the concept of a set of reactors inside the same plant provides the operator with a range of solutions to adapt the maintenance schedule with priority given to the grid supply needs. There is always at least one reactor of the plant in-operation and supplying the grid, whereas the other one may be in outage.

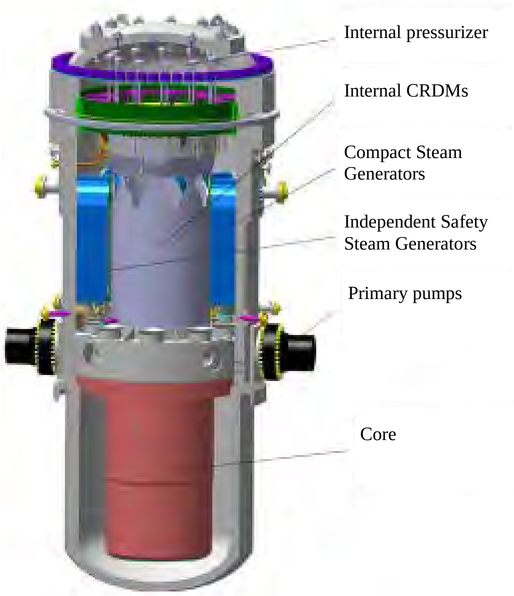
3. Main Design Features

(a) Design Philosophy

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

(b) Nuclear Steam Supply System

The NUWARD™ reactor is a fully integrated PWR reactor, housing in one unique vessel all the main reactor coolant system components, including the Steam Generators, the Pressurizer and the Control Rod Drive Mechanisms (CRDM).



(c) Reactor Core

The reference core is based on proven 17x17 fuel assemblies used in the operating PWR with a shortened core height and UO2 rods (enrichment < 5wt% U_{235}). Due to the boron-free design, various U_{235} enrichments and burnable poisons are used. The reference refuelling interval is 24 months.

(d) Reactivity Control

The reactivity is controlled by means of control rods and solid burnable poison. The reactor boron-free reactivity control allows for simplification of auxiliary systems design and operations in both normal and accident situations (Design Basis Conditions) as well as a drastic reduction in effluents production from operation.

(e) Reactor Pressure Vessel and Internals

The RPV and equipment layout are designed to facilitate enhanced in-factory manufacturing. A specific design effort has been made to reduce the number of pipes connected to the RPV with the objective to limit the maximum LOCA size to a 30 mm in diameter.

(f) Reactor Coolant System and Steam Generator

NUWARD™ reactor coolant system is based upon the use of an innovative once-through steam generator technology derived from the plate heat exchanger concept for which specific developments made in terms of design and manufacturing process allow for a nuclear application. This technology offers a Compact Steam Generator (CSG) with a direct connection to the reactor. The overall size, and particularly the height of the reactor coolant system is therefore significantly reduced given the reactor thermal power.
There are six (6) compact steam generators (CSG).

(g) Pressuriser

The pressurizer of NUWARD™ is also integrated within the RPV. The large volume of the pressurizer provides margins for the operational transients as well as for the normal operation of the reactor.

(h) Primary Pumps

Six (6) canned pumps are horizontally mounted onto the RPV, positioned under the SGs in the cold leg for efficient hydraulic conditions.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

The NUWARD™ reactor and associated safety systems are designed for:
- Passive management of all DBC scenarios with no need of any operator's action, any external ultimate heat sink source, any boron injection or any external electrical power supply (normal and emergency) for more than 3 days.
- Active management of DEC-A accidents, with simple diagnosis and implementation of diversified systems.
- Passive management of DEC-B accidents with in-vessel retention of the corium (IVR concept).

The safety approach for NUWARD™ benefits from the following inherent features of the design to satisfy and maintain a safe state that requires minimum intervention from the operating team:
- Large reactor coolant inventory in kg/MW(t), providing inertia versus power transients.
- Boron-free operation providing large and constant moderator counter-reaction and preventing boron dilution.
- Integrated RCS architecture reducing the maximum LOCA break size thus providing more time for coping with the design basis LOCA. Internal CRDMs preventing rod-ejection accidents.
- A metallic submerged containment providing passive cooling for several days.
- A small core in a large vessel enabling the success of the in-vessel retention strategy for DEC-B accidents.

(b) Safety Approach and Configuration to Manage DBC

With regard to Decay Heat Removal: NUWARD™ incorporates 2 trains of passive heat removal system transferring by natural circulation the decay heat from the core to the water wall surrounding the containment through two safety compact steam generators (S-CSG) independent from the 6 normal Steam generators (CSG). Each train is actuated by 2 diversified channels (diversified sensors, diversified I&C and diversified actuators). The water-wall surrounding the containment ensures the heat removal function for more than 3 days without the need for an external ultimate heat sink. The passive vessel heat removal system shall be considered as a D-passive system according to IAEA classification. The minimized break size in a LOCA and the efficient passive heat removal system result in a minimized coolant loss during LOCA. A set of 2 redundant low-pressure safety injection accumulators provides the make-up of reactor coolant water inventory in case of LOCA.

With regard to criticality and reactivity control: NUWARD™ includes safety features to prevent criticality risks. The core is sub-critical with clear water at 20°C even if the most efficient absorber is stuck in upper position. This option prevents the occurrence of criticality accidents even in post-accident conditions. Moreover, the use of internal CRDM eliminates the occurrence of a rod ejection accident. The reactivity control management system (passive absorber drop) shall be considered as a D-passive system according to IAEA classification.

Safety systems used to manage DBC are not shared between the different reactors in the same plant.

(c) Safety Approach and Configuration to Manage DEC

DEC-A systems are:

- Low flowrate depressurization system and active primary/secondary water injection. This system provides for the removal of the decay heat in case of a postulated common mode failure of redundant trains of passive DBC safety systems,
- High pressure borated water injection to cope with ATWS accidents.

DEC-B systems are:

- Low flowrate depressurization system to reach a primary pressure less than 2 MPa before corium relocation,
- Passive flooding of vessel pit in order to provide in-vessel retention of corium,
- Nitrogen injection to manage the risk of hydrogen combustion.

(d) Containment System

In order to fulfil the containment function, NUWARD™ adopts a steel containment as the 3rd barrier submerged in a water wall. The minimized LOCA break size and the efficient passive heat removal system result in a limited peak pressure inside the steel containment which is passively cooled by the surrounding water wall. The containment is protected against hydrogen burning risk in DBC by passive recombiners. The containment valves are parts of the D-passive system. The passive containment heat removal system is considered as an A-passive system according to IAEA classification.

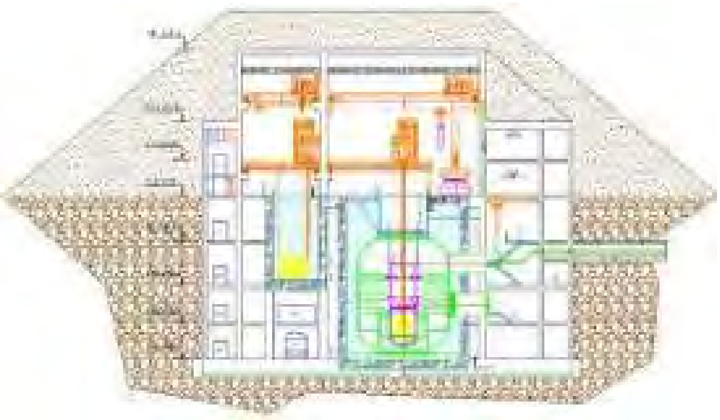
5. Plant Safety and Operational Performances

The design target values for the lifetime capacity factor is 90% with the major planned refuelling outages scheduled for 15 days every 24 months. The reference refuelling strategy is to replace half of a core every 2 years. The plant provides a storage capacity of spent fuel assemblies for 10 years of operation (20 operation years as an option).

6. Instrumentation and Control Systems

The Instrumentation and Control (I&C) system designed for NUWARD™ is based on defence in depth concept, compliance with the single failure criterion and diversity.

7. Plant Layout Arrangement



The nuclear island (NI) houses 2 independent modules and an associated fuel storage pool. No system/resource (including heat sink) outside the NI is required to ensure the safe-state for at least 3 days; the NI is self-reliant for at least this period due to the adoption of water-wall in which each containment vessel is immersed.

As for site requirements during construction, NUWARD™ plants are suitable for sea-onshore and/or river-side sites, with open-loop conventional condenser cooling. Nevertheless, inland site with dry aero condensers is a possible option.

As for site considerations during operation, water is needed for filling-in and making-up of various onsite tanks, water circuits, and the water wall; for the balance of plant, water is required for providing continuous flow of cooling water for conventional condenser

cooling, in case the open-loop design is used;
NUWARD™ is also designed to satisfy grid interface/code requirements: basic grid interface compliant with ENTSO-E and EUR requirements (typically 225 kV/400 kV and 50 Hz). Possible adaptation to specific user requirements such as 60 Hz is possible.

NUWARD™ plant layout.

8. Design and Licensing Status

NUWARD™ is in conceptual design stage and internal evaluation for future licensing activities are taking place.

9. Fuel Cycle Approach

The reference core of NUWARD™ is directly derived from the proven 17x17 UO_2 fuel assembly in use in most operating PWRs worldwide. The reference refueling strategy is by half a core every 2 years. The plant provides a storage capacity of spent fuel assembly for 10 years after operation before the decommissioning. An option of 20 years of storage could also be proposed. Other options regarding the fuel cycle are currently under assessment relying on then best practices observed in the industry

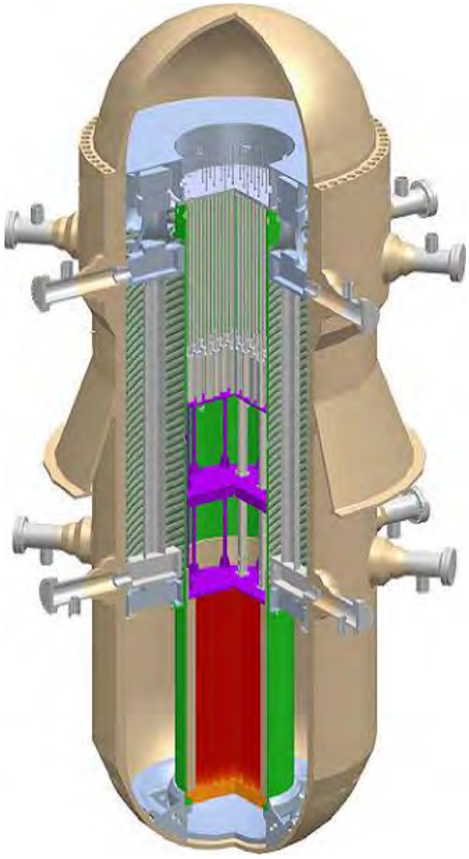
10. Waste Management and Disposal Plan

Options regarding waste disposal are currently under assessment relying on then best practices observed in the industry.

11. Development Milestones

2012-2016	Preliminary studies and technological innovation (using previously developed patents).
2017-2019	Pre-conceptual design phase and technology validation
2019-2022	Conceptual design phase
2030	Projected deployment time

IRIS (IRIS International Consortium)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	IRIS, International Consortium
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	1000 / 335
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 5.8
Core Inlet/Outlet Coolant Temperature (ºC)	292 / 330
Fuel type/assembly array	UO_2/MOX/17x17 square
Number of fuel assemblies in the core	89
Fuel enrichment (%)	4.95
Core Discharge Burnup (GWd/ton)	65 (max)
Refuelling Cycle (months)	48 (max)
Reactivity control mechanism	ICRDM (Internal Control Rod Drive Mechanism)
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	14 000 (four-unit layout)
RPV height/diameter (m)	21.3 / 6.2
RPV weight (metric ton)	1045
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	3-year cycle with half-core reload
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:

- A medium power of 335 MW(e) per module;
- A 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. Main 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 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. 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 houses 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. 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 adopts once-through steam generators (OTSGs) with helical-coil tube bundle design with the primary fluid outside the tubes. Eight (8) OTSG modules are installed in the annular space between the core barrel and the RV. Each IRIS OTSG 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 OTSG has 656 tubes. 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

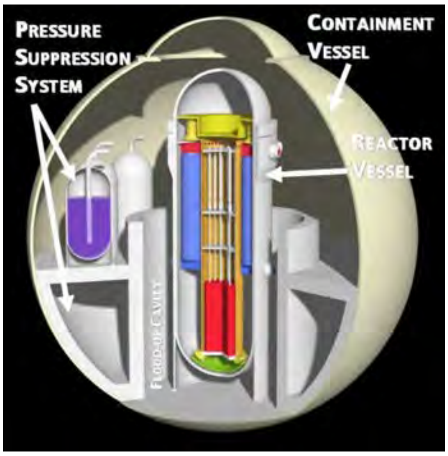
The IRIS pressurizer is integrated into the upper head of the reactor vessel. The pressurizer region is defined by an insulated, inverted top-hat structure that divides the circulating reactor coolant flow path from the saturated pressurizer water. This structure includes a closed cell insulation to 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 m^3, which includes a steam volume of 49 m^3.

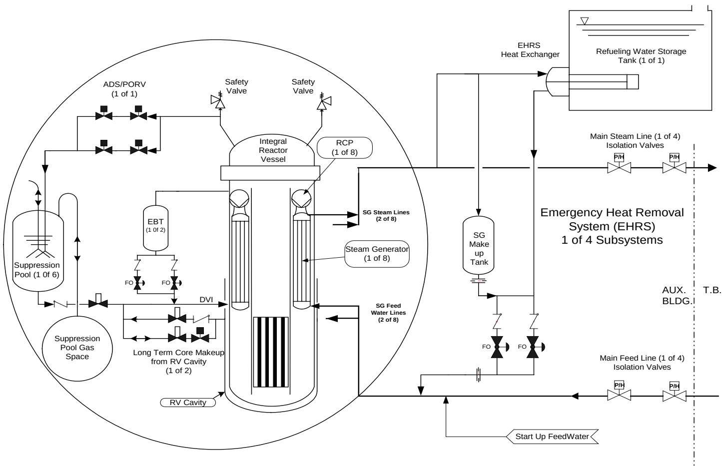
4. Safety Features

IRIS adopts passive safety systems and the safety by design philosophy including the risk informed approach. Due to the integral configuration, by design, with no intervention of safety systems, a variety of accidents either are eliminated or their consequences and/or probability are greatly reduced. In fact, 88% of class IV accidents 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 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 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 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

IRIS integral RV configuration eliminates the loop piping and the externally located steam generators, pumps and pressurizer with their individual vessels. Hence, the footprint of the containment 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. 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 pressure suppression pool 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 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. Even though the reference design features a two-batch, 3-year fuel cycle, IRIS is capable of eventually operating in straight burn with a core lifetime of up to 8 years. IRIS has been designed to extend the need for scheduled maintenance outages to at least 48 months. With the 4-year maintenance cycle, the capacity factor of IRIS is expected to exceed the 95% target, and personnel requirements are expected to be significantly reduced. Both considerations will result in decreased O&M costs.

6. Instrumentation and Control Systems

The instrumentation systems and components for IRIS are in principle similar to those for Generation III PWRs. Innovative solutions have been developed specifically to address: level measurement in the RPV upper head volume (pressurizer), primary flow rate measurement in annular space geometries, primary fluid temperature measurements at the outlet of steam generator modules. For safety purposes IRIS eliminated all the lower head RPV penetrations for instrumentation guide tubes.

The control system envisaged for IRIS is based on an Autonomous and Hierarchical Control Functional Architecture, adopting Model-predictive, Multivariate robust and Fault-tolerant Controllers. Moreover, a Multi-modular IRIS Control and Operational Reconfiguration for a Flow Control Loop has been developed. That approach was used both to control a multi-module IRIS NPP as well as to monitor and control a hybrid system, adopting IRIS units as electricity and heat supply, to feed both the grid and cogeneration units, like desalination plants.

7. Plant Layout Arrangement

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

8. Design and Licensing Status

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

9. Fuel Cycle Approach

IRIS longer-term objective is to further enhance its economic and proliferation resistance characteristics by extending the reloading interval to 4 years and beyond. Therefore, a multiprong approach is adopted including a range of fuel options:

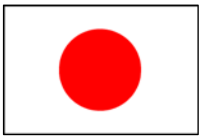
1. rely on proven and licensed fuel technology to enable the near-term deployment objective;
2. perform research on advanced core designs with higher discharge burnup and longer cycle for longer-term deployment;
3. additionally, different needs and preferences of different countries should be addressed, such as emphasis on proliferation resistance, or use of MOX or thorium fuel.

10. Waste Management and Disposal Plan

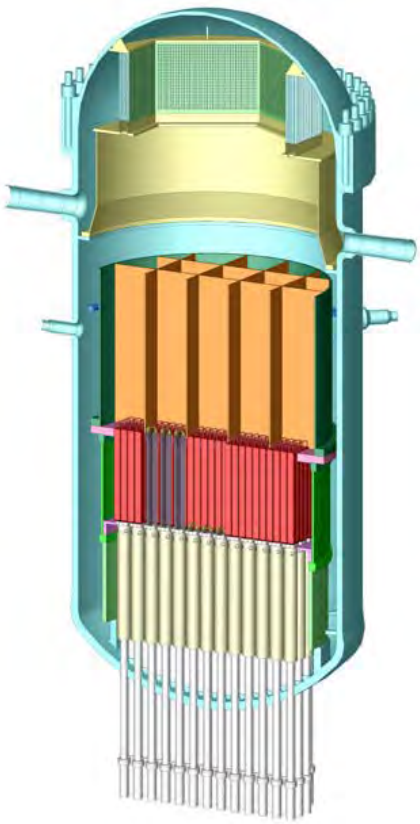
Waste management and disposal plan are similar to those adopted by in Generation III PWR reactors. About decommissioning, systematic dose reduction to personnel in decontamination and decommissioning (D&D) activities was planned. A specific feature of IRIS is the radial water layer of 1.7 m between the edge of the core and the RV. This natural shielding decreases the fast neutron fluence on the RV by a factor of 10^5, essentially eliminating vessel embrittlement and the need for surveillance coupons and for periodic in-service inspection of the RV. Hence, reduced vessel activation, which simplifies D&D, with potential to dispose of significant portion of the vessel as nonradioactive materials.

11. 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 / ~300
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	7.17
Core Inlet/Outlet Coolant Temperature (°C)	186 / 287
Fuel type/assembly array	UO_2 pellet / 10x10 square configuration in channel box
Number of fuel assemblies in the core	400 (short fuel assembly)
Fuel enrichment (%)	< 5
Core Discharge Burnup (GWd/ton)	< 60
Refuelling Cycle (months)	24
Reactivity control mechanism	Control rod drive and soluble boron injection
Approach to safety systems	Hybrid (passive + active)
Design life (years)	60
RPV height/diameter (m)	15 / 4.8
Seismic Design (SSE)	0.45g
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. Main 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 the turbine directly. NSSS includes the reactor core and internals, reactor pressure vessel and the coolant/steam piping within the containment system. As for the reactor coolant flow, natural circulation is adopted. Therefore, DMS does not require any reactor coolant pump. This feature eliminates loss of flow accident (LOFA). Penetrations of large piping connected into the RPV are located in the active fuel so the reactor core will always be covered by coolant even in the most severe LOCA condition. The NSSS is designed to remove core power by natural circulation during normal operation, and the core can be also cooled by natural circulation flow even in anticipated transients or accident cases.

(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 5wt%. 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 reactor 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 Coolant System

The DMS reactor primary cooling mechanism under normal condition and shutdown condition is by natural circulation of coolant. In order to enhance the driving force, divided chimney of about 3m height is installed 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. The increased height and diameter of the RPV ensure the availability of large coolant inventory. 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.

(f) Reactor Pressure Vessel and Internals

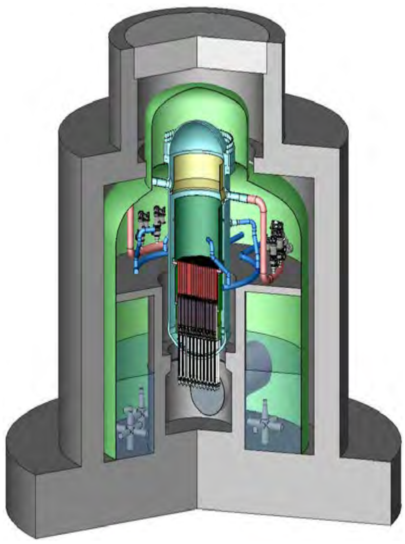
The size of the RPV is a main factor that determine the size of primary containment vessel (PCV) and influences construction cost of the reactor building. The small size RPV is made possible by simplifying core internals through smaller core and natural circulation, and by eliminating steam separator. The flat core concept with short-length and large core diameter is adopted to reduce the power density 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 the FSS system.

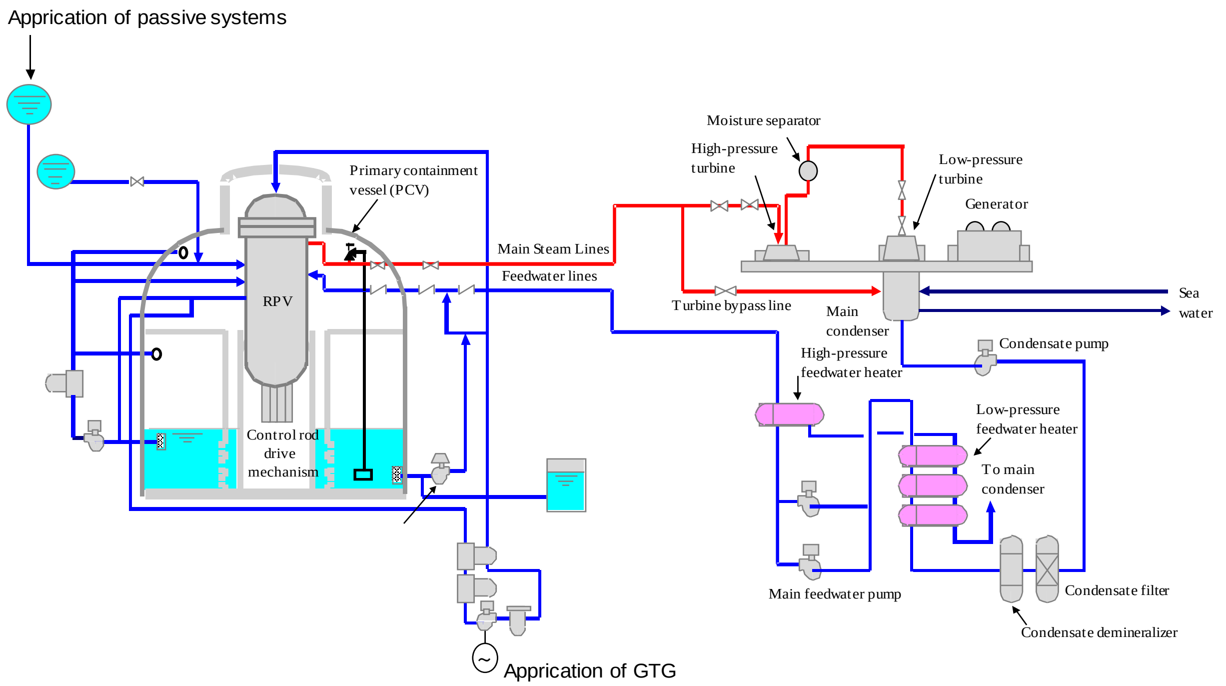
4. Safety Features

DMS has a larger coolant inventory compared to forced circulation type LWRs of the same output. This is because RPV’s height is increased to secure the driving power for natural circulation. The RPV’s diameter is also increased to accommodate more fuel assemblies. These features eliminate the need of high-pressure injection system.

(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. There are four main distinctive features: (1) High pressure core flooder (HPCF) equipped in conventional BWR is eliminated due to the larger coolant inventory in the DMS; (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 analysed and it was confirmed that the core could be cooled for 10 days against SBO and peak cladding temperature (PCT) was kept less than 1200ºC even against the most severe DBA.





Cut-away view of DMS power plant

(b) Decay Heat Removal System

The residual heat removal system (RHR) includes a number of pumps and heat exchangers to cool the reactor or the suppression pool (S/P) in the PCV. The RHR can remove residual heat not only during normal shutdown plant's outage but also during an accident. 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 passive safety systems without using any AC power. Both systems can condense steam generated in the RPV and steam released from the RPV to the PCV via ruptured piping during LOCA. The heat exchangers of the IC and PCCS are cooled by the 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 used to achieve the design pressure of 427 kPag 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. As in BWR and ABWR, the pressure suppression containment is 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. DMS uses only two main steam lines with diameter equivalent to that of conventional BWR thus minimizes the size of the primary containment vessel.

6. Instrumentation and Control Systems

The DMS adopts digital I&C systems that include microprocessors and the field programmable gate arrays (FPGAs) making use of fault detection and fault tolerance. A diversity is important 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 ECCS are included in the SSLC.

7. Plant Layout 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 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 plant is connected to the external grid via a main connection and a standby connection. The main connection is the connection between the generator transformer and the external grid. The standby connection is the connection between the auxiliary standby transformer (AST) and the external grid.

8. 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)

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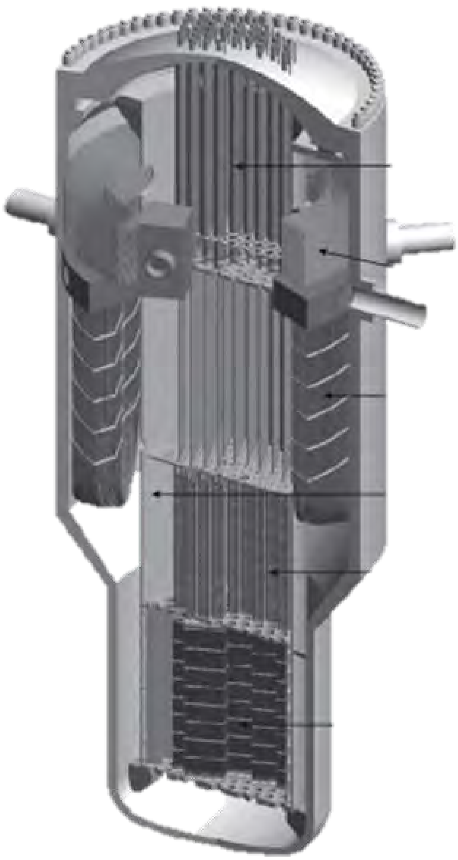


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Mitsubishi Heavy Industries, Ltd. (MHI), Japan
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	1000 / 350
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 5.0
Core Inlet/Outlet Coolant Temperature (^{o}C)	329 / 345
Fuel type/assembly array	UO_2 pellet / 21x21 square
Number of fuel assemblies in the core	97
Fuel enrichment (%)	4.8
Core Discharge Burnup (GWd/ton)	> 40
Refuelling Cycle (months)	26
Reactivity control mechanism	Control rods drive mechanism
Approach to safety systems	Hybrid (Passive + Active) system
Design life (years)	60
Plant footprint (m^2)	4900
RPV height/diameter (m)	17 / 6
RPV weight (metric ton)	--
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Similar to existing PWR plants
Distinguishing features	Integral PWR with natural circulation; employs two types of in-vessel steam generator
Design status	Conceptual design completed

1. Introduction

The Integrated Modular Water Reactor (IMR) is a medium sized power reactor with a reference thermal 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 the primary heat transport. 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 modular power station to generate electricity. Because of its modular characteristics, it is suitable for large-scale power stations consisting of several modules and also for small distributed-power stations, especially for small grids. 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 components that require no large-scale infrastructure. This facilitates regulatory authority's approval for the construction and operation of power plant.

3. Main Design Features

(a) Design Philosophy

The IMR is an integral PWR where the primary system components are installed within the reactor pressure vessel (RPV). Main coolant piping and primary coolant pumps are eliminated by adopting natural circulation system. Pressurizer is eliminated by adopting the self-pressurization system. There are two types of steam generator (SG) inside the RPV; The first type is located in the vapour/upper region of the RPV and the other is located in the liquid/lower region of the RPV. These SGs are also used as decay heat removal heat exchangers during normal startup and shutdown operations and in accidents. Hence to eliminate the need of ECCS, the SGs serve as a passive safety system that do not require any external power. The IMR has a moderation ratio similar to the operating PWRs. Thus, its properties of fresh and spent fuel are similar. This allows for conventional safeguards measures and PWR management practices for new and spent fuel. 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 an output of 1000 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 assure 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 of that conventional PWRs. 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 used 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 for the control bank. 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 applies 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

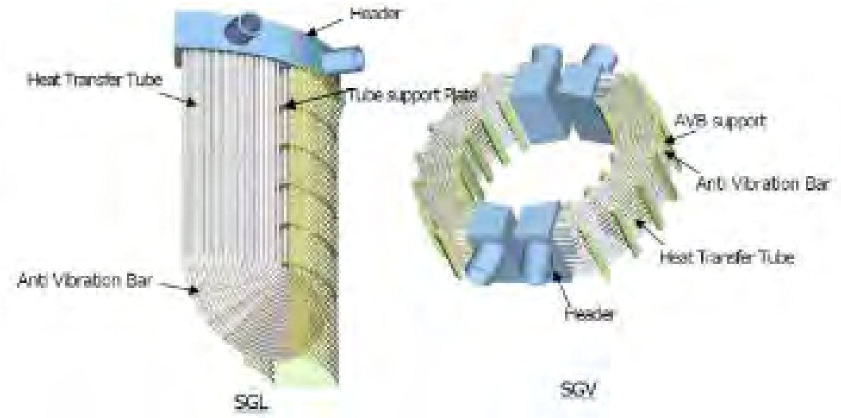
The upper part of the RPV inside diameter is about 6 m in order to accommodate the in-vessel SGs. The inside diameter of the lower part of the RPV is reduced to about 4 m in order to minimize the cold-side water volume. In order to eliminate the necessity for the consideration of LOCA, the largest diameter 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. The HHTS reduces the size of RPV. The coolant starts boiling in the upper part of the core, and two-phase coolant in bubbly flow flows up in the riser and condenses in the SGs. This design approach increases coolant flow rate and thus reduces the height of RPV 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

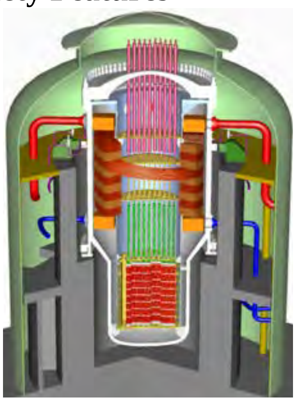
The IMR adopts two types of SG. The first one is the SG in vapour region (SGV) located above the water level in the RPV. The energy transported by vapour formation generates secondary steam through SGV. As the vapour in the RPV is condensed by SGV, controlling the feedwater flow rate to SGV controls the RPV pressure. The other is the steam generator in liquid region (SGL) of the RPV. The energy transported by liquid temperature rise generates secondary steam through SGL. Because the core inlet temperature can be controlled by the amount of heat removal through SGL, the core power can be controlled by 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 to optimize space utilization in the vapour part of the RPV.

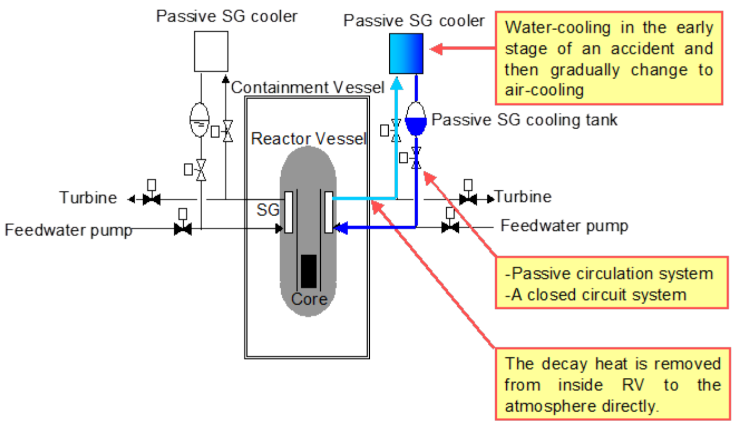


(h) Pressurizer

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

4. Safety Features





(a) Engineered Safety System Approach and Configuration

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 in IMR. Since the diameter of the pipes connected to the RPV 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. If a malfunction such as SG tube leakage occurs, system functions are maintained. The capacity of chemical and volume control system (CVCS) is provided via eight 3-inch pipes connected to the RPV. 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. Hence, containment spray system is also eliminated. The auxiliary feedwater system is used for startup and shutdown procedures during normal operation. The auxiliary feedwater system is not a safety system. When the auxiliary feedwater system becomes unavailable, the SDHS is actuated. 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 powered by a stand-alone diesel generator.

(b) Decay Heat Removal System

The SDHS is activated to remove the decay heat from the RPV to the atmosphere. Even if water leakage occurs 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, water, power and operators are not necessary for maintaining the plant safety.

(c) Containment System

A compact containment vessel (CV) is made possible due to the integrated primary system and simplified auxiliary systems. The IMR uses reinforced concrete containment. A higher design pressure of the containment is to meet the safety requirement that water leakage from RPV shall be terminated automatically. Since this

CV is about one size larger than the RPV, it is expected to resist high pressure. 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.

5. Plant Safety and Operational Performances

The IMR is designed to operate 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. The primary system pressure and reactor power are controlled by feedwater and control rods.

6. Instrumentation and Control Systems

The instrumentation and control (I&C) systems provide the capability to control and regulate the plant systems manually and automatically during normal plant operation and provide reactor protection against unsafe plant operation. Fully digitalized I&C system including computerized control board for plant operator are provided with the required conventional system.

7. Plant Layout Arrangement

The IMR concepts of building layout are reducing the bill of quantity for construction material, shortening the construction period and standardizing the plant design. Utilizing the steel plate reinforced concrete and simplify the shape of building and structures achieve the construction cost and period reduction.

(a) Reactor Building

Ground level is assumed to be flat land above sea level. The bedrock is assumed to be less than 40 m below ground to enable the use of pile foundations. The integrated reactor building can house two units. Exclusion of waste disposal facilities in another building.

(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.

8. Design and Licensing Status

The IMR conceptual design study was initiated in 1999 by MHI. A group led by MHI 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.

9. Fuel Cycle Approach

The IMR fuel cycle approach including spent fuel management is in line with the approach for the existing PWR plants. It leads the minor design modification for existing fuel cycle facilities, and the IMR approach is accepted by public without discomfort.

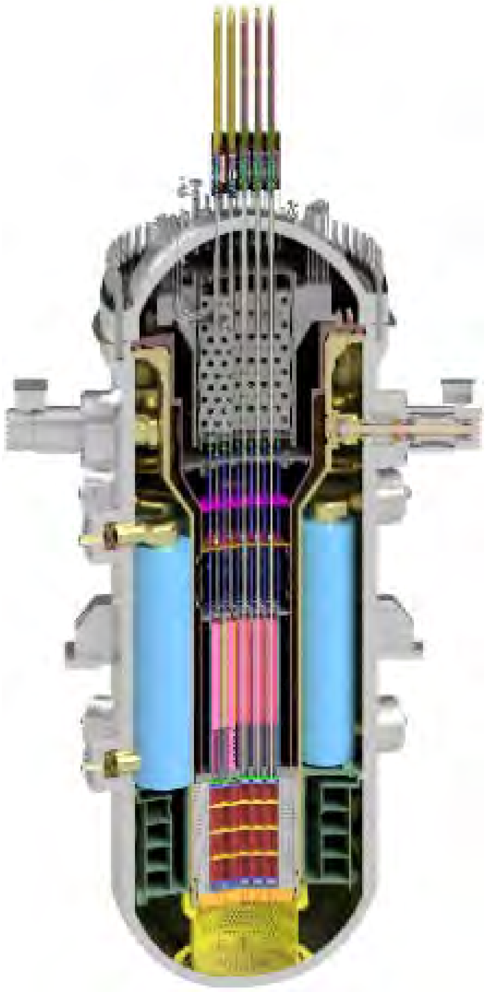
10. Waste Management and Disposal Plan

The IMR waste management and disposal plan is in line with the existing PWR plants concept. The IMR approach is accepted by public without discomfort.

11. Development Milestones

1999	MHI started conceptual design study for IMR.
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.
2009-2011	Startup transient tests to verify the startup flow instability were studied
2019	MHI is developing a new Small Reactor based on the IMR experiences with funding support by the Ministry of Economy, Trade and Industry.

SMART (KAERI, Republic of Korea and K.A.CARE, Saudi Arabia)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin)	KAERI, Republic of Korea and K.A.CARE, Kingdom of Saudi Arabia
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	365 / 107
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15 / 5.8
Core Inlet/Outlet Coolant Temperature (ºC)	296 / 322
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	57
Fuel enrichment (%)	< 5
Core Discharge Burnup (GWd/ton)	< 54
Refuelling Cycle (months)	30
Reactivity control mechanism	Control rod driving mechanisms and soluble boron
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	90 000
RPV height/diameter (m)	18.5 / 6.5
RPV weight (metric ton)	1070 (including coolant)
Seismic Design (SSE)	> 0.3g with 0.18g of automatic shutdown
Fuel cycle requirements / Approach	Conventional LWR requirements applied (spent fuel capacity: 30 years)
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 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 100 000.

3. Main 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. 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 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. In the core, there are 57 fuel assemblies (FA) of 2 m long, standard 17x17 square of UO_2 ceramic fuel with less than 5% enrichment, similar to standard PWR fuel. A two-batch refuelling scheme without reprocessing provides a cycle of 870 effective full power days for operation.

(d) Reactivity Control

Reactivity control during normal operation is achieved by control rods and soluble boron. Burnable poison rods are introduced to give flat radial and axial power profiles, which results in an increased thermal margin of the core. 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

The RPV houses the reactor core, eight (8) steam generators (SGs), four (4) canned motor reactor coolant pumps, 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.

(g) Pressurizer

The in-vessel pressurizer uses the free volume in the upper part of the RPV. The primary system pressure is maintained nearly constant due to the large pressurizer steam volume and the heater control. Due to the large volume of the pressurizer, condensing spray is not required for load manoeuvring operation. The reactor over-pressure at the postulated design basis accidents can be reduced through the actuation of pressurizer safety valve.

4. Safety Features

(a) Engineered Safety System and Configuration

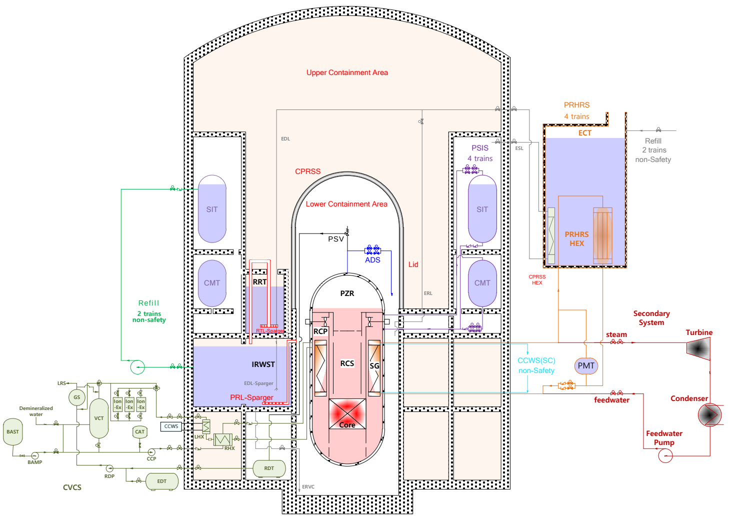
Safety systems of SMART are designed to function automatically. These consist of a reactor shutdown system, a passive safety injection system, a passive residual heat removal system, and containment pressure and radioactivity suppression system. Additional safety systems include an automatic depressurization system (ADS) and pressurizer safety valves, and a severe accident mitigation system.

(b) 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.

(c) 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. 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.



(d) 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 radioactivity 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 CPRSS Heat Removal System (CHRS). 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 CHRS. Fission products are scrubbed in the IRWST water. For combustible gas control, passive autocatalytic hydrogen recombiners are equipped inside the LCA and UCA.

5. Plant Safety and Operational Performances

Load following operation of SMART is simpler than that of large PWR because only a single bank movement and small insertion is required to induce small reactivity change. This feature minimizes coolant temperature change, relatively high lead bank worth due to a small number of fuel assemblies and the short core height leading to rapidly damping the xenon oscillation. The daily load following performance simulation of SMART core shows that radial peaking factor, 3D 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 and 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 in–core instrumentation system consists of 29 detector assemblies which are developed as mini type for SMART with four stacked rhodium self–powered neutron detectors.

7. Plant Layout 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 the reactor containment and auxiliary building (RCAB). The RCAB houses reactor containment, auxiliary and fuel handling areas. For efficient radiation management, the RCAB is sub-divided into two zones; the duty zone and the clean zones.

(a) Reactor Building

The 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 refuelling 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.

(b) Main Control Room

The SMART compact main control room (MCR) is designed for one reactor operator operation with shift supervisor under normal conditions of the plant. The MCR is a key facility to cope with any emergency situations and 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.

(c) Balance of Plant

The balance of plant design (BOP) consists of:

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 provides the preferred power circuit for the auxiliary power system of SMART.

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. This PPE project was successfully completed in February 2019 and First-of-a-Kind (FOAK) plant construction in Saudi Arabia will follow in due course.

9. Fuel Cycle Approach

The fuel cycle of SMART is 30 months. KEPCO-NF can provide SMART fuel with its fuel fabrication facility increment schedule. The SMART spent fuels are stored in a spent fuel pool using storage racks. The current storage capacity of spent fuel storage racks is 30 years which can be variable upon owner’s requirements.

10. Waste Management and Disposal Plan

SMART has several design solutions to minimize radioactive waste generation. All liquid radioactive waste will be processed through demineralizer package which can make the system design to be simple and minimize shipment of solid waste. Gaseous radwaste system provides sufficient holdup decay of radioactive waste gases and release gases in a controlled manner. Solid radwaste system adopts polymer solidification technology which can minimize shipped volume for spent resin.

11. Plant Economics

The target overnight plant construction cost of a FOAK unit is $10 000/kW(e) and an operating and maintenance cost of 2.8 ¢/kWh. For NOAK unit of SMART, the total cost is expected to be 30~40% lesser.

12. 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
November 2015	Pre-Project Engineering started.
February 2019	The Pre-Project Engineering completed.
January 2020	SMART100 Standard Design Approval Applied.

RITM-200 (JSC "Afrikantov OKBM", Russian Federation)

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RITM-200 Reactor

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	165 / 53
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15.7
Core Inlet/Outlet Coolant Temperature (°C)	277 / 313
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	199
Fuel enrichment (%)	<20
Core Discharge Burnup (GWd/ton)	-
Refuelling Cycle (months)	72-84
Reactivity control mechanism	Control rods
Approach to safety systems	Combined active and passive
Design life (years)	60 with possible extension
Plant footprint (m^2)	60 000 – NPP with 2 reactors 89 500 – NPP with 4 reactors
RPV height/diameter (m)	7.5 / 3.4
Seismic Design (SSE)	0.3g
Distinguishing features	Integral reactor, in-vessel corium retention, double containment
Design status	Six RITM-200 reactors are installed on icebreakers Arktika, Sibir and Ural. Land-based NPP version under development

1. Introduction

RITM-200 is the latest development in III+ generation SMR line designed by the JSC 'Afrikantov OKBM'. It has incorporated all the proven features from its predecessors. It is also based on proven PWR technology and 400 reactor-years of Rosatom experience in operation of small reactors in icebreakers. Six RITM-200 reactors are successfully installed on icebreakers Arktika, Sibir and Ural. Two reactors of Arktika icebreaker successfully passed all power up tests during dock-side trials.

2. Target Application

RITM series reactors can be used for multiple applications including electrical power generation, water desalination, and cogeneration.

3. Main 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). Incorporation of the steam generators into the reactor pressure vessel (RPV) has made the reactor system and containment very compact as compared to the KLT-40S. Using RITM, it is possible to increase electric output (40% more) and reduce the dimensions (45% less) and the mass (35% less) in contrast to KLT-

40S. While integral reactor configuration virtually eliminates large loss-of-coolant accident (LOCA), the other inherent features and active and passive safety systems form diversity, redundancy, physical separation, and functional independence to achieve the required safety level and reliability.

(b) Nuclear Steam Supply System

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

(c) Reactor Core

RITM reactor core accommodates low enriched fuel assemblies similar to KLT-40S that ensures long time operation without refueling and meets international non-proliferation requirements. The height of the core is 1650 mm. The core consists of 199 fuel assemblies with uranium-intensive cermet fuel. The core has the service life of 8 TWh. The fuel rods are resistant to power changes with a design rate of 0.1% Prated/s.

(d) 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 safety rods is designed to scram the reactor and to maintain it in subcritical condition. The design of control and safety rods is based on the drives used in KLT-40S.

(e) Reactor Coolant System

The RPV is a thick-walled cylindrical pressure vessel with an integrally welded bottom head and a removable top head. The integral RPV has four main circulation pumps located in separate external hydraulic chambers with side horizontal sockets for SG cassette nozzles. The pumps are single-stage vaned and have a canned asynchronous electric motor. The SGs provide steam of 295°C at 3.82 MPa and capacity of 261 t/h.

(f) Steam Generator

The RITM-200 uses once-through (straight tube) SGs. The SG is divided into four primary circuit loops. Each loop consists of three once-through cassettes (12 in total) with a common feed water and common steam manifolds. Each cassette consists of 7 modules. There is a specially designed system for SG leakage detection. In case a leak is detected, it is possible to find out and isolate the leaking module individually.

(g) Pressurizer

The design adopts pressure compensation gas system well-proven 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 independent parts to reduce the pipe diameter in the compensatory nozzles of the steam generating unit and to decrease a coolant leakage rate in large break LOCA. It is possible to use one of pressurizers as hydraulic accumulator, increasing reactor plant reliability in case of LOCA.

4. Safety Features

RITM-200 applies the defense-in-depth safety principle combined with inherent features and passive systems. Inherent safety features are applied to control power density and reactor scram, limit primary coolant pressure and temperature, heating rate, primary circuit depressurization rate, fuel damage scope, and maintain reactor vessel integrity in severe accidents. RITM-200 optimally combines passive and active safety systems to cope with abnormal operating occurrences and design basis accidents. Pressurizer is divided into two independent ones to minimize size of potential coolant leakage rate.

(a) Engineered Safety System Approach and Configuration

The high safety level of RITM series reactors is achieved by inherent safety features and a combination of passive and active safety systems. Redundancy of safety system equipment and channels and their functional and/or physical separation are provided to ensure high reliability. 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. Safety rods drop into the core by gravity with spring assist when power is removed from electromagnetic couplings consequently ensure reactor scram even in case of total station blackout.

(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 the heat exchanger of primary-third circuits of primary circuit coolant purification loop.
- Two passive safety loops with natural circulation from water tanks through SGs. All safety train are connected to different SGs and provide residual heat removal in compliance with single failure criterion.

(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

The containment consists of three levels:
- Primary containment designed for internal pressure of 0.5 MPa (abs) with dimensions of 6 m × 6 m × 15.5 m around the reactor vessel to localize possible radioactive releases (~300 m^3 of free space).
- Secondary containment is a solid building core made of thick reinforced concrete walls (800 mm thick) to protect primary containment from external impact.
- Third level of containment is a collapsible building structure of thin reinforced concrete walls to dissipate most of the energy of external impact and minimize influence on secondary containment.

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

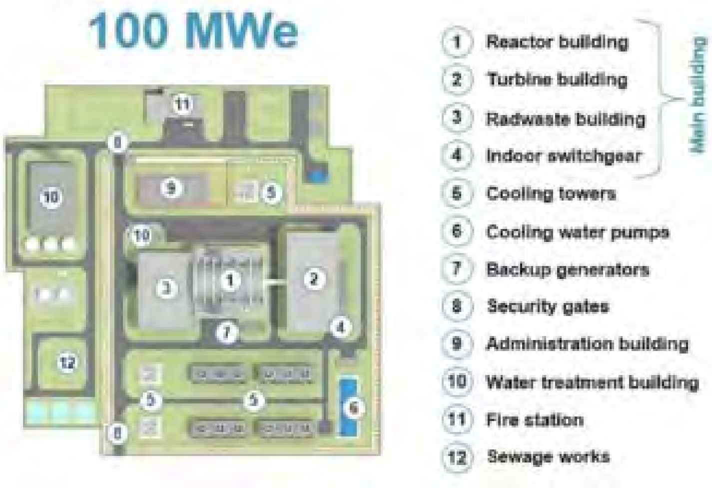
5. Instrumentation and Control Systems

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 Layout Arrangement

The land-based small NPP consists of two RITM-200 reactors with a specified electrical capacity of 100 MW. Buildings are arranged to optimize interconnections and interfaces between buildings and to minimize unused areas. An aerial view of small NPP and the plant layout are presented below.

RITM-200 plant layout



Small NPP layout is sectioned in two areas:

- Secured area related to power generation systems itself, including reactor and turbine buildings and cooling towers. It also includes the building for nuclear material management (radwaste building).

Auxiliary system area consists of water treatment buildings, fire station, administration building, etc.

Small NPP secured area.

Small NPP auxiliary area.

Small NPP site segmentation allows utilizing modular approach for simplification of possible future plant electrical capacity growth. Construction of additional areas with reactor and turbine buildings allows increasing power generation incrementally by step of 100 MW. While the auxiliary building systems stay in shared use for all nuclear units. Small NPP site area for 100 MW is 14.8 acres (59 894 m^2), 200 MW - 22.2 acres (89 840 m^2), and 300 MW - 29.7 acres (120 192 m^2).

Scalability of small NPP based on RITM-200 rectors.

(a) Main building

The optimized design of small NPP consists of one reactor building with two RITM-200 reactors. It is assumed that the two reactors will be commissioned simultaneously. The main building actually consists of three buildings:

- Reactor building footprint is 45 m x 44 m and 35 m of height.
- Turbine building footprint is 30 m x 63 m and 31 m of height.
- Nuclear material management building (radwaste building) footprint is 36 m x 48 m and 19 m of height.

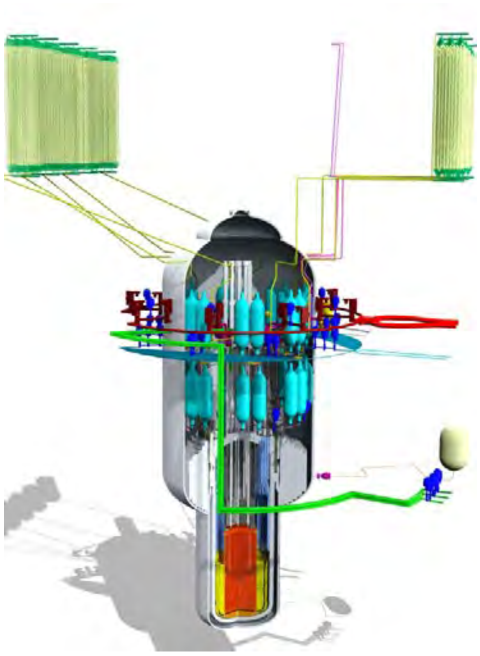
7. Design and Licensing Status

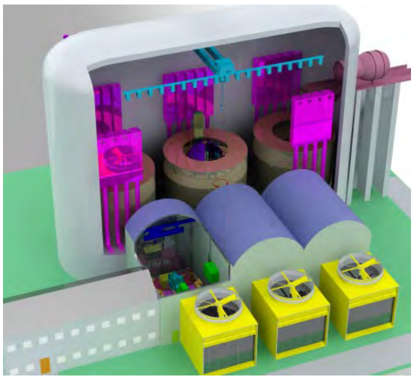
The RITM-200 design is developed in conformity with Russian law, codes and standards for nuclear power plants and safety principles developed by the world community and IAEA requirements.

8. Development Milestones

2012	Detailed design of RITM-200
2018	Land-based NPP conceptual design
2023	Site license
2024	License for construction, first concrete
2027	License for operation, NPP commissioning

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
NSSS Operating Pressure (primary/secondary), MPa	16.5
Core Inlet/Outlet Coolant Temperature (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 in the core	265
Fuel enrichment (%)	19.75
Core Discharge Burnup (GWd/ton)	1.15
Refuelling Cycle (months)	200
Reactivity control mechanism	Soluble boron and control rod insertion
Approach to safety systems	Hybrid (passive + active) system
Design life (years)	30
Plant footprint (m²)	~10 000
RPV height/diameter (m)	9.8 / 2.9
RPV weight (metric ton)	32
Seismic Design (SSE)	VIII-IX-MSK 64
Fuel cycle requirements / Approach	Traditional
Distinguishing features	Autonomous passive reactor decay removal system; guard vessel; iron-water biological shielding; and the 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 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 as both 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. Main 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

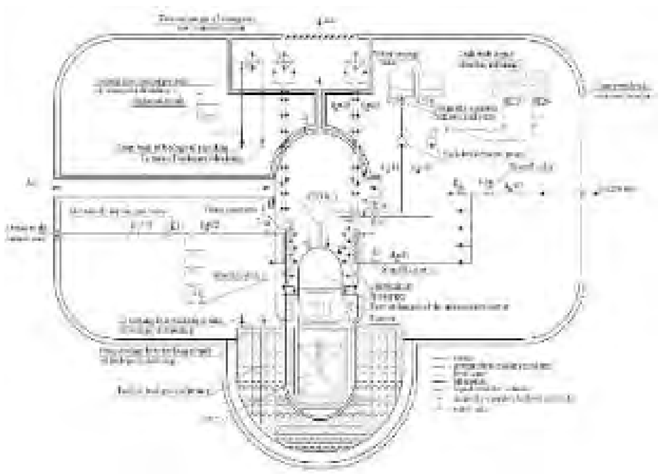
Primary circuit system is intended for heat removal from the reactor core and heat transfer to the intermediate circuit fluid inside the intermediate heat exchanger. The system consists of a main circulation train and a pressurizing system. The natural circulation of primary coolant takes place in the 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 turbo generator 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 potentially 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

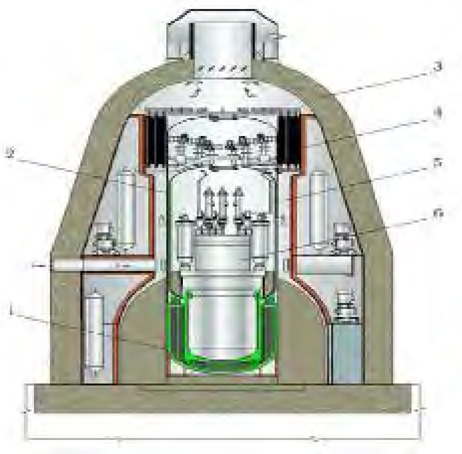
The UNITHERM safety systems are based upon redundancy, diversity and the maximum use of the fail-safe systems. The UNITHERM employs passive safety systems and devices which do not require actuation (such as containment, independent heat removal system, etc.) or can be passively actuated (such as primary 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.



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

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.

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 ± 1 Hz. 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 trialed in 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 Layout 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. Fuel Cycle Approach

The duration of the campaign reactor core is 15 year.

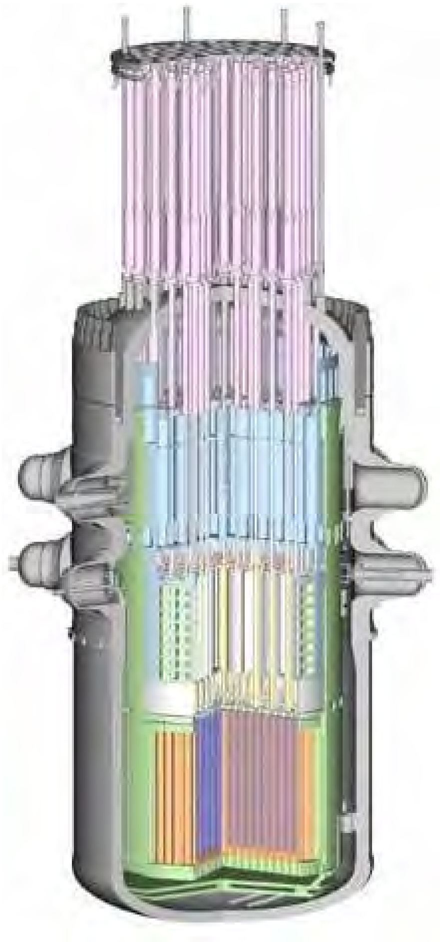
10. Waste Management and Disposal Plan

Fuel handling is based on the traditional scheme implemented for the marine-based prototype reactor. Fuel processing and disposal will be performed at a specialized enterprise.

11. 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
NSSS Operating Pressure (primary/secondary), MPa	6.9
Core Inlet/Outlet Coolant Temperature (°C)	190 / 285
Fuel type/assembly array	UO_2 pellet/hexahedron
Number of fuel assemblies in the core	313
Fuel enrichment (%)	4
Core Discharge Burnup (GWd/ton)	41.4
Refuelling Cycle (months)	72
Reactivity control mechanism	Rod insertions
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	40 000
RPV height/diameter (m)	13.1 / 4.535
RPV weight (metric ton)	325
Seismic Design (SSE)	Max 8 point of MSK-64
Fuel cycle requirements / Approach	Once through fuel cycle with UO_2
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 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 of electricity and heat for district heating and for sea water desalination, having excellent characteristics of safety and economics.

3. Main 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 has successfully operated in Russian Federation over the last 50 years. Therefore, the enhanced reliability and economics are achieved by the use of some proven modified structures and components in the design.

(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 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 Pressure Vessel and Internals

The VK-300 reactor vessel is a WWER-1000 reactor vessel in terms of external dimensions and material.
The VK-300 reactor includes the following internals:
- a shell with the basket of the core;
- a traction pipe unit;
- a separator unit.

The traction pipe unit is an assembly of 90 vertical traction pipes of triangular-oval section and 25 circular pipes. The separator unit consists of 133 axial centrifugal separators.

(f) 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.

(g) Steam Generator

The VK-300 reactor employs in-vessel cyclone separators that are designed and experimentally optimized to be used in the 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 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) Engineered Safety System Approach and Configuration

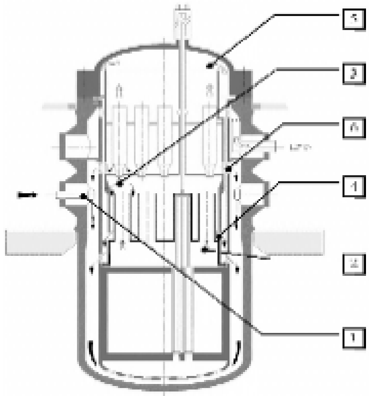
The main technological solutions of VK-300:
- single-loop reactor with natural coolant circulation;
- power self-limitation due to negative reactivity and thermal coefficients;
- passive removal of residual heat;
- placement of reactor, turbine, emergency cooling tanks, spent fuel storage pool, reloading machine and central hall crane under a single secondary containment;
- two independent power control and reactor shutdown systems (CPS using absorbing rods and CPS using rods and boron fluid);
- fully integrated reactor layout.

(b) Decay Heat Removal System

The primary goal 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 due 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 pumps.

(c) 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 period of time; (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

(d) Containment System

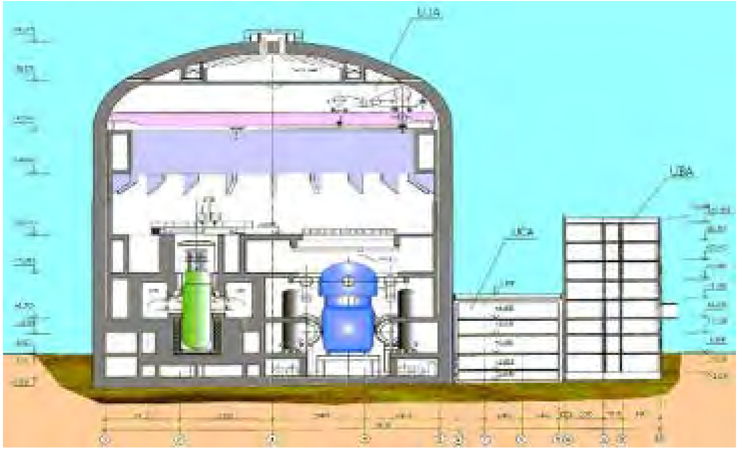
The VK-300 reactor adopts a metal-lined primary containment system (PCS) of reinforced concrete. The PCS helps to solve the safety assurance problem economically and reliably using structurally simple passive safety systems. The PCS is rather small, with volume about 2000 m^3. The PCS of the VK-300 performs 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 defense-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 Layout 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

Given 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(e) 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. Fuel Cycle Approach

The standard fuel cycle option for the VK-300 is a once-through fuel cycle with uranium dioxide fuel. According to the design of the VK-300, spent fuel assemblies should be stored in the cooling pond for 3 years after discharge from the reactor core and then transported to the fuel reprocessing plant without further long-term on-site storage. The standard fuel reprocessing method as used for WWER-1000 type reactors.

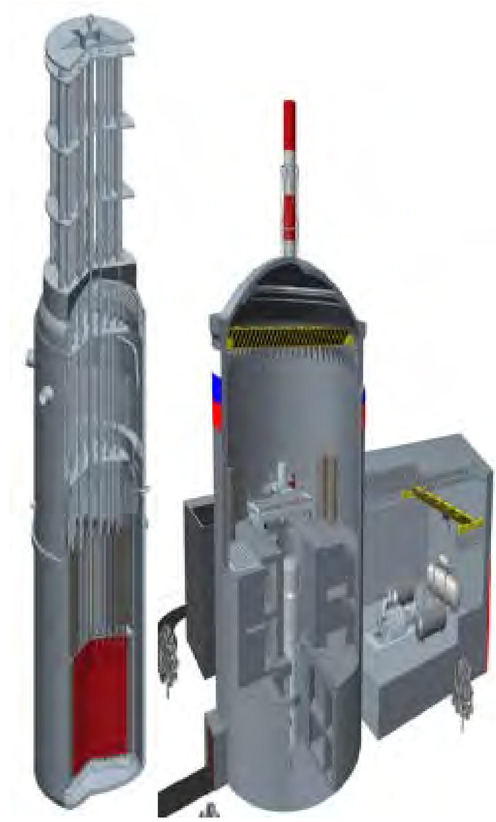
10. Waste Management and Disposal Plan

Radioactive waste is to be transferred to the National Radioactive Waste Management Operator for subsequent disposal.

11. 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

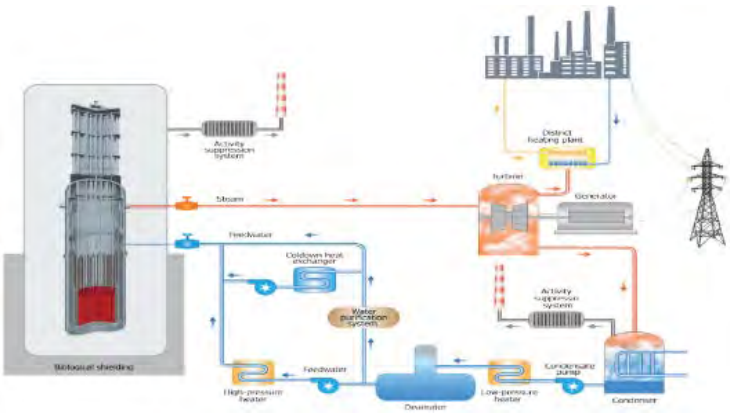
KARAT-45 (NIKIET, Russian Federation)



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
NSSS Operating Pressure (primary/secondary), MPa	7.0 / –
Core Inlet/Outlet Coolant Temperature (^{o}C)	180 / 286
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	109
Fuel enrichment (%)	4.5
Core Discharge Burnup (GWd/ton)	45.9
Refuelling Cycle (months)	84
Reactivity control mechanism	Control rods drive
Approach to safety systems	Passive
Design life (years)	80
Plant footprint (m^2)	9000
RPV height/diameter (m)	11.15 / 3.10
RPV weight (metric ton)	176
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Refueling (fuel shuffling) interval is up to 800 EFPD; Fuel assembly life cycle is about 6.6 years
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. Main Design Features

(a) Design Philosophy

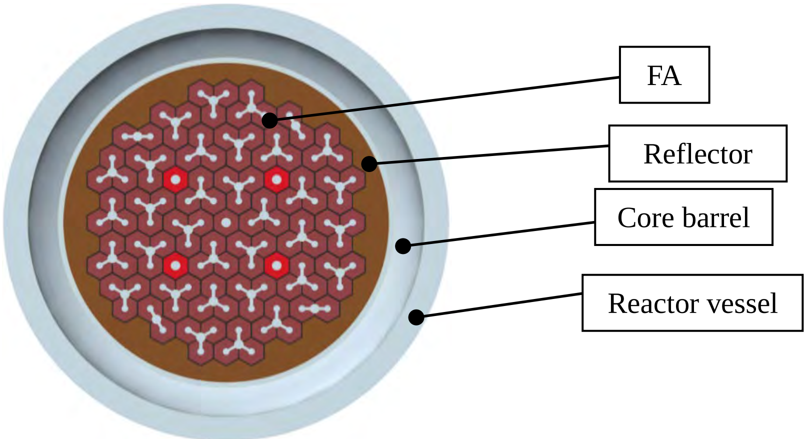
The BWR technology was selected as a basis for the design and technology development of KARAT-45 due to the following rationales: BWR employs single circuit removal of heat so capital cost for construction can be minimized; lower system pressure poses fewer challenges to the reactor vessel; BWR has inherent 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 11 150 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) 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.

(d) 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 Layout 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.

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. Fuel Cycle Approach

Concept of the fuel cycle foresees long term reload period. Within this concept the interval between reloads equals to 800 effective days while the share of reloaded fuel assemblies is one third/. The full campaign at this condition is 6.6 years.

10. Waste Management and Disposal Plan

Strategy of the decommissioning and RAW management is being developed during the conclusion of the contract and is prepared in accordance with the IAEA recommendations.

11. Development Milestones

T_0+2	Development of a preliminary design
T_0+4	Technical requirements, R&D program, basic design
T_0+6	R&D, development of PSAR, expert review and licensing, architecture engineering
T_0+7	Detailed design, fabrication of equipment, construction
T_0+11	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
NSSS Operating Pressure (primary/secondary), MPa	7.0
Core Inlet/Outlet Coolant Temperature (ºC)	104 / 286
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	199
Fuel enrichment (%)	4
Core Discharge Burnup (GWd/ton)	45.9
Refuelling Cycle (months)	90
Reactivity control mechanism	Control rods drive mechanism
Approach to safety systems	Passive
Design life (years)	80
Plant footprint (m^2)	22 500
RPV height/diameter (m)	13.25 / 4.00
RPV weight (metric ton)	348
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Refueling (fuel shuffling) interval is up to 900 EFPD; Fuel assembly life cycle is about 7.5 years
Distinguishing features	Multi-purpose reactor: cogeneration of electricity and heat
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. Main 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 Layout 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 components and equipment, and a bridge crane for moving operations. The dimensions of the turbine block are 42 m x 28 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. Fuel Cycle Approach

Concept of the fuel cycle foresees long term reload period. Within this concept the interval between reloads equals to 900 effective days while the share of reloaded fuel assemblies is one third/. The longevity of the full campaign at this condition is 7.5 years.

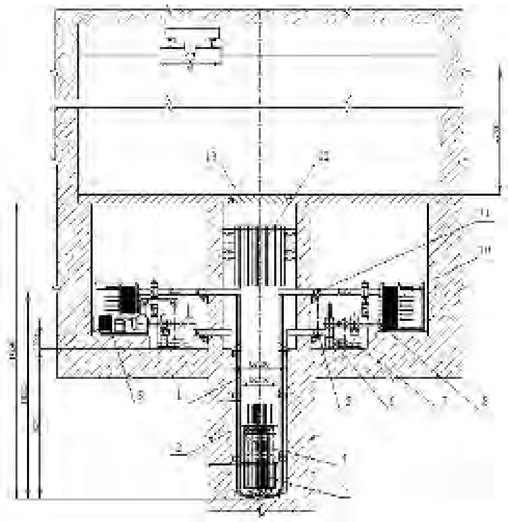
10. Waste Management and Disposal Plan

Strategy of the decommissioning and RAW management is being developed during the conclusion of the contract and is prepared in accordance with the IAEA recommendations.

11. Development Milestones

T_0+1	Development of technical proposal
T_0+2	Development of a preliminary design
T_0+4	Technical requirements, R&D program, basic design
T_0+6	R&D, development of PSAR, expert review and licensing, architecture engineering
T_0+7	Detailed design, fabrication of equipment, construction
T_0+11	Operation license, first criticality, first start, commissioning

RUTA-70 (NIKIET, Russian Federation)



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 rated power)/forced (for 30-100% of rated power)
NSSS Operating Pressure (primary/secondary), MPa	Atmospheric pressure at reactor poll water surface
Core Inlet/Outlet Coolant Temperature (^oC)	75 / 102
Fuel type/assembly array	Cermet (0.6 UO_2+ 0.4 Al alloy) / hexagonal
Number of fuel assemblies in the core	91
Fuel enrichment (%)	3.0
Core Discharge Burnup (GWd/ton)	25-30
Refuelling Cycle (months)	36
Reactivity control mechanism	Control rods
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	100 000
RPV height/diameter (m)	17.25 / 3.20
Seismic Design (SSE)	> 0.8g (automatic shutdown)
Fuel cycle requirements / Approach	Once-through fuel cycle with UO_2 fuel
Distinguishing features	Design capable for radioisotopes production for medical, neutron beams for neutron therapy and industrial purposes
Design status	Conceptual design

1. Introduction

RUTA-70 is a multi-purpose water-cooled water-moderated integral pool-type reactor serving as a Nuclear Heating Plant (NHP) of 70 MW(t) thermal capacity for district heating, desalination and radioisotopes production for medical and industrial purposes. It has no power 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. 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 raises the down-comer temperature. RUTA-70 can perform continuous operation without any maintenance for about one year. RUTA-70 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, RUTA can also perform seawater and brackish water desalinations based on distillation process.

3. Main Design Features

(a) Design Philosophy

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

(b) Nuclear Steam Supply System

RUTA-70 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. 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 using 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) Steam Generator

The turbine and associated systems (including steam generator) are not used in the NHP RUTA-70.

(g) 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. 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. 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 in both 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 Systems

RCP actuators based on two diverse principles of action have been chosen for the RUTA70:

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

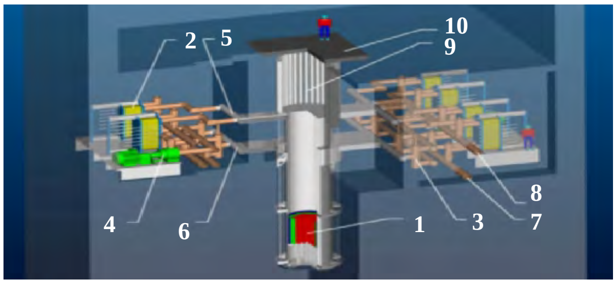
In the core there are 42 reactor control and protection system (RCP) rods composing two shutdown systems with diverse actuators. One of these systems intended specifically for core emergency protection (EP) includes 12 rods. The second shutdown system performing the concurrent functions of shutdown and 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 Layout 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



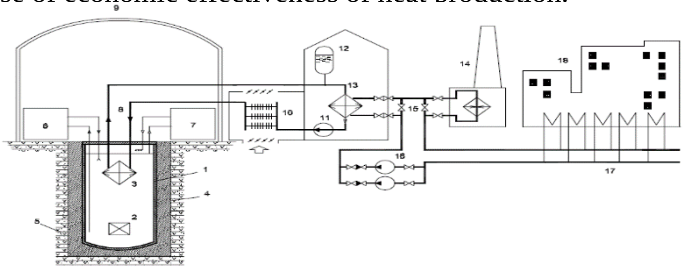
(b) Control Building

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. 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 in spite of some reduction of autonomy of district heating system (in comparison to high-temperature reactors) possesses several major advantages: Increase in reliability of heat supplying due to diversification of heat sources, provide redundancy required by relatively cheap heat sources and increase of economic effectiveness of heat production.

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. Fuel Cycle Approach

The standard fuel cycle option for the RUTA70 NHP is a once-through fuel cycle with uranium dioxide fuel. The alternative fuel cycle option is a once-through cycle with cermet fuel (microparticles of fuel in a metallic matrix). Standard fuel reprocessing method as used for VVER type reactors could be applied. In this, fuel reprocessing can be made centralized. According to the design of the NHP RUTA70, spent fuel assemblies should be stored in the cooling pond for 3 years after discharge from the reactor core and then transported to the fuel reprocessing plant without further long-term on-site storage.

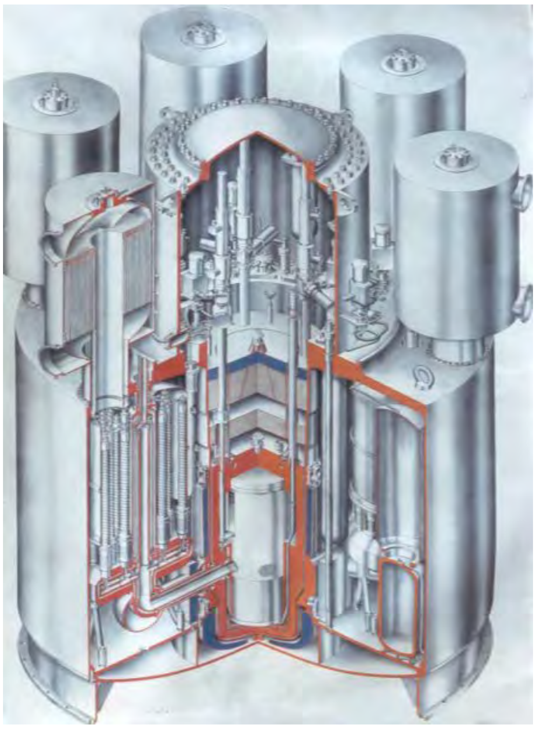
10. Waste Management and Disposal Plan

Radioactive waste is to be transferred to the 'National Radioactive Waste Management Operator' for subsequent disposal.

11. 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

ELENA (NRC “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
NSSS Operating Pressure (primary/secondary), MPa	19.6 / 0.36
Core Inlet/Outlet Coolant Temperature (^{o}C)	311 / 328
Fuel type/assembly array	UO_2 pellet; MOX is an option
Number of fuel assemblies in the core	109
Fuel enrichment (%)	15.2
Core Discharge Burnup (GWd/ton)	57 600 / 27 390
Refuelling Cycle (months)	300
Reactivity control mechanism	Control rods and absorber rods
Approach to safety systems	Passive
RPV height/diameter (m)	3.7/1.25
Seismic Design (SSE)	VIII (MSK-64)
Fuel cycle requirements / Approach	Initial factory load for the entire lifetime
Distinguishing features	25 years without refuelling, passive reactivity control and 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 National Research Centre “Kurchatov Institute” (NRC 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. Main 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 U_{235} enrichment of 15.2%; the neutron moderator and coolant is water specially treated according to specified water chemistry. Cylindrical fuel elements with stainless steel cladding are installed in 109 fuel assemblies of 55 fuel elements each; 216 absorber rods with boron carbide based neutron absorber are divided into 6 groups. Fuel assemblies also include burnable absorbers made of Gd-Nb-Zr alloy. The U_{235} load is 147 kg.

(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 about 100°C. The power level is primarily dependent upon the temperature of the third loop. The internal space for heat transport to consumers is connected to an air-cooled heat exchanger enclosed in the draft tube for excess heat discharge to the atmosphere.

(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: (1) Fuel elements; (2) Leak-tight primary circuit; (3) Caisson; (4)

Reactor vessel and the guard vessel designed to withstand the pressure arising within each of them at their consecutive failure; and (5) 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 start-up. 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

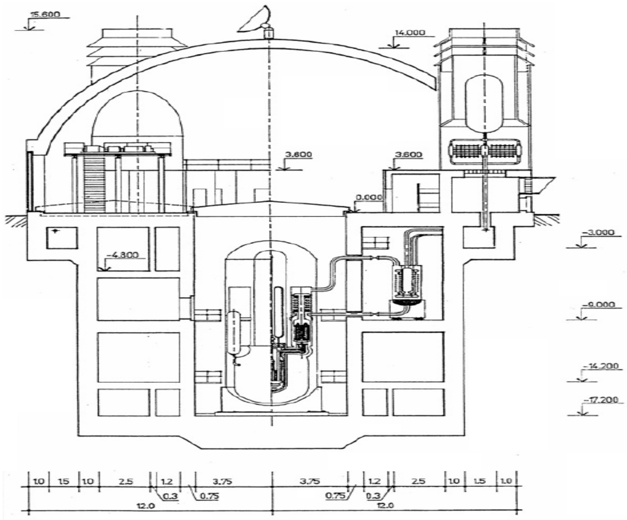
The instrumentation and control (I&C) system of the ELENA reactor is specially arranged to register parameter deviations at early stages of the accidental conditions to predict their further progression.

7. Plant Layout 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. Fuel Cycle Approach

The factory-fabricated reactor vessel is delivered to the site loaded with fresh fuel. This initial load is designed to provide the whole NPP lifetime without refuelling.

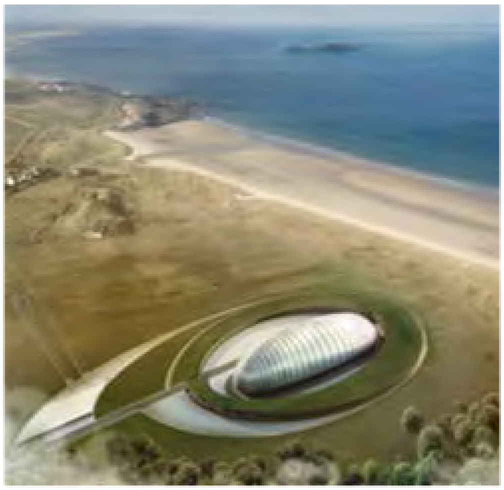
10. Waste Management and Disposal Plan

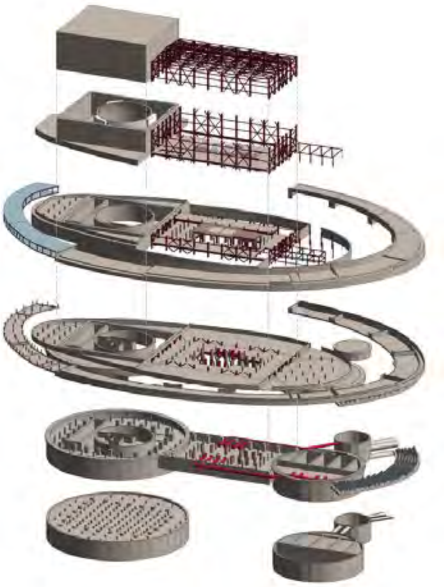
The waste management is not required during the ELENA-NTEP lifetime due to the safety barriers and no need for maintenance. At its lifetime end, the reactor vessel is removed with the spent fuel in a shipping cask. Liquid and solid radioactive waste is also disposed using special equipment. The site is either provided with a new ELENA-NTEP or proceeds to "greenfield" status.

11. Development Milestones

Not determined.

UK SMR (Rolls-Royce and Partners, United Kingdom)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Rolls-Royce and Partners, United Kingdom
Reactor type	3-loop PWR
Coolant/moderator	Light-water / Light-water
Thermal/electrical capacity, MW(t)/MW(e)	1276 / 443
Primary circulation	Forced (3 pumps)
Operating Pressure (primary/secondary), MPa	15.5 / 7.6
Core Inlet/Outlet Coolant Temperature (^oC)	296 / 327
Fuel type/assembly array	UO_2 / 17x17 Square
Number of fuel assemblies in the core	121
Fuel enrichment (%)	4.95 (max)
Core Discharge Burnup (GWd/ton)	55 – 60
Refuelling Cycle (months)	18 – 24
Reactivity control mechanism	Rods and Gd_2O_3 solid burnable absorber
Approach to safety systems	Active and passive
Design life (years)	60
Plant footprint (m^2)	10 000
Site footprint (m^2)	40 000
RPV height/diameter (m)	11.3 / 4.5
RPV weight (metric tonnes)	220
Seismic Design (DBE)	> 0.3g
Fuel cycle requirements / Approach	Open cycle; Spent fuel transferred to a pool for storage prior to transfer to long term dry cask storage.
Distinguishing features	Modular approach facilitating rapid and cost-effective build.
Design status	Conceptual design

1. Introduction

The UK SMR has been developed 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 a class leading safety outlook and attractive market offering with minimum regulatory risk.
A three loop, close-coupled, Pressurised Water Reactor (PWR) provides a power output of 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 build is enhanced through site layout optimisation and maximising modular build, standardisation and commoditisation.

2. Target Application

The UK SMR is primarily intended to supply baseload electricity for both coast and inland siting. The design can be configured to support other heat-requiring or cogeneration applications, as well as provide a primary, carbon free, power source for the production of e-fuels.

3. Main Design Features

(a) Design Philosophy

The design philosophy for the UK SMR is to optimise levelised cost of electricity against low capital cost. The power output is maximised whilst delivering robust economics for nuclear power plant investment and a plant size that enables modularisation and standardisation throughout.

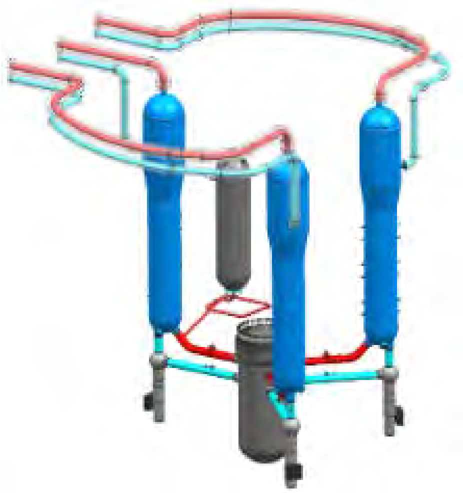
(b) Reactor Core

The nuclear fuel is industry standard UO_2 enriched up to 4.95%, clad with a zirconium alloy and arranged in a 17x17 assembly. The core contains 121 fuel assemblies and has an active fuelled length of 2.8 m, delivering a thermal power of 1276 MW(t). Each fuel assembly contains 40 poisoned fuel pins, with the remaining 224 fuel pins being unpoisoned. The poison used is distributed Gd_2O_3 (containing natural gadolinium) at 8 wt%.

(c) Reactivity Control

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

(d) Reactor Coolant System



The RCS is a three loop, close-coupled configuration; Steam Generators are located around the circumference of the Reactor Pressure Vessel (RPV), with short close-coupled pipework connections between them. The pressuriser is connected to the RCS pipework hot leg. A centrifugal Reactor Coolant Pump is mounted via a close-coupled nozzle, from the bottom of each SG outlet header.

(e) Reactor Pressure Vessel

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

(f) Steam Generator

A vertical u-tube SG design has been selected as a mature and readily deployable technology; other designs were considered but deemed insufficiently mature for commercial deployment in 2030.

(g) Pressuriser

The primary circuit pressure is controlled by electrical heaters located at the base of the pressuriser and spray from a nozzle located at the top. Steam and water are maintained in equilibrium to provide the necessary overpressure. The pressuriser is a vertical, cylindrical vessel constructed from low alloy steel, sized to provide passive fault response for bounding faults, with accidents causing either rapid and significant cooldown or heat-up accommodated.

4. Safety Features

(a) Safety System Approach

The UK SMR design has been developed through a combined system engineering and safety assessment approach. The safety informed design supports the process by which risks are demonstrated to be acceptable and As Low as Reasonably Practicable (ALARP).

Defence in depth is provided through the provision multiple layers of fault prevention and protection in the form of independent and diverse active and passive systems, with multiple trains per system. Passive safety systems are designed to deliver their safety functionality autonomously for 72 hours, minimising the demand on human actions and AC electrics.

(b) Engineered Safety Measures

In addition to duty 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. Full bore RCS large 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.

The primary circuit and other key systems are located within a steel containment vessel to confine release of radiation sources during both normal and faulted operation. The UK SMR also adopts in-vessel retention (IVR) to confine the postulated melt in severe accidents.

5. Plant Safety and Operational Performances

The behaviour of the plant during normal and faulted conditions has been analysed and assessed using industry validated codes to demonstrate significant safety margin across the levels of defence in depth.

The Probabilistic Safety Assessment (PSA) calculates an overall core damage frequency from plant faults of $<10^{-7}$ per year of power operation, and that no single fault or class of faults makes a disproportionate contribution to risk, i.e. a balanced risk profile is achieved. Internal and external hazard assessments have defined the design basis and informed the plant layout from a perspective of segregation and separation of safety related equipment. Key equipment is protected by the hazard barrier which is resilient against external hazards including aircraft impact and tsunami.

6. Instrumentation and Control Systems

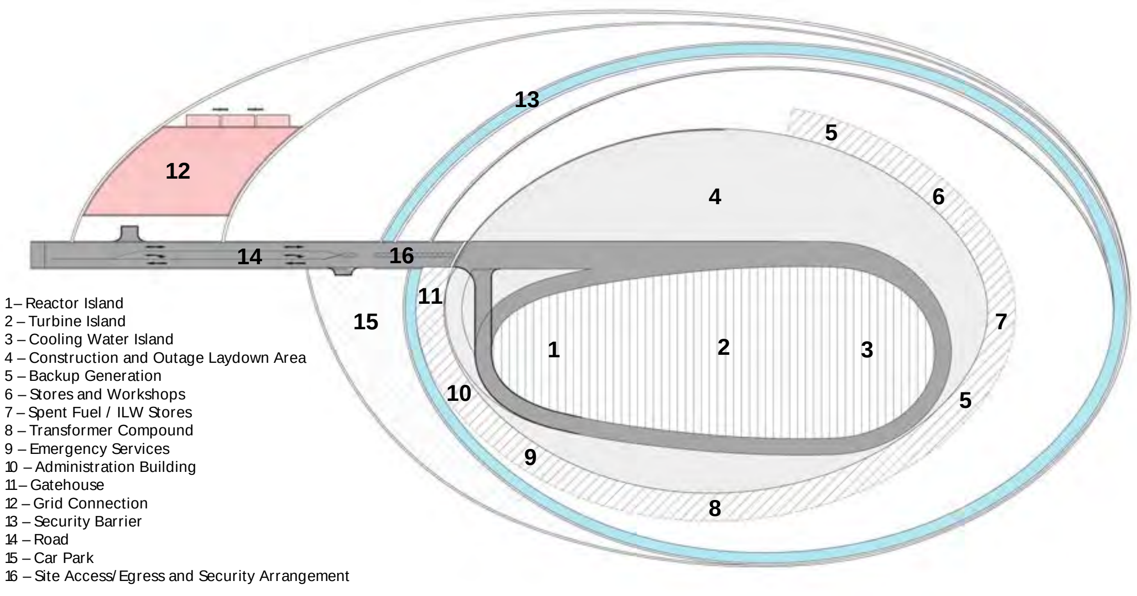
The UK SMR plant is controlled and protected by a number of control and instrumentation (C&I) systems. The reactor plant control system manages duty operations and uses an available in industry programmable logic controller (PLC) or distributed control system (DCS). It uses mixed analogue and nonprogrammable 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 shutdown in response to a fault. The RPS contains priority logic, which from the range of input signals received determines whether to initiate reactor shutdown. The hardwired diverse protection system (HDPS) uses non-programmable electronics and as such provides a diverse means to shut down the plant in response to fault conditions.

Post-Accident and Severe Accident Management Systems within the Nuclear C&I System provide clear plant status displays, over the days and months following a postulated accident.

7. Plant Layout

The power station is designed for installation on an extensive range of inland and coastal sites, across a wide range of soil and earth conditions, whilst maintaining a compact site footprint of approximately 40 000 m^2. This flexibility is enabled through design features such as seismic isolation for safety related areas. The three-loop PWR is located in the Reactor Island, adjacent to Turbine Island with the Cooling Water Island following. Support buildings and auxiliary services are situated within a berm that sweeps around the site and provides a layer of protection from external hazards such as tsunami or aircraft impact.



8. Design and Licensing Status

The project targets completion of the UK Office for Nuclear Regulation Generic Design Assessment process in time for construction of the first of a kind power station to commence in 2025. This timescale is considered achievable through the optimised use of proven technologies to minimise development time and regulatory risk. A consortium has been formed to deliver the UK SMR, with a wide range of additional UK based academic and industrial partners engaged to further develop capability. 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

9. Fuel Cycle Approach

The UK SMR operates on an 18- to 24-month fuel cycle, with a three-batch equilibrium core. The duration of refuelling outage is currently estimated at 18 days, with significant scope for further optimisation as the design progresses. Refuelling is managed through the provision of an in-containment refuelling pool which temporarily stores both new and spent fuel during a refuelling outage. Spent fuel is subsequently transferred to an external spent fuel pool for storage prior to transfer to long term dry cask storage.

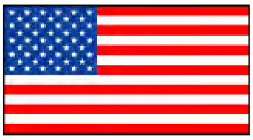
10. Waste Management and Disposal Plan

The UK SMR waste treatment systems are based on use of proven technologies and best available techniques. Industry lessons learned and good practices have been used in the development of systems to minimise active and non-active wastes and discharges, through both design and operational practices adopted. Standardised waste treatment system components and modules are used to achieve the flexibility required for the waste informed design.

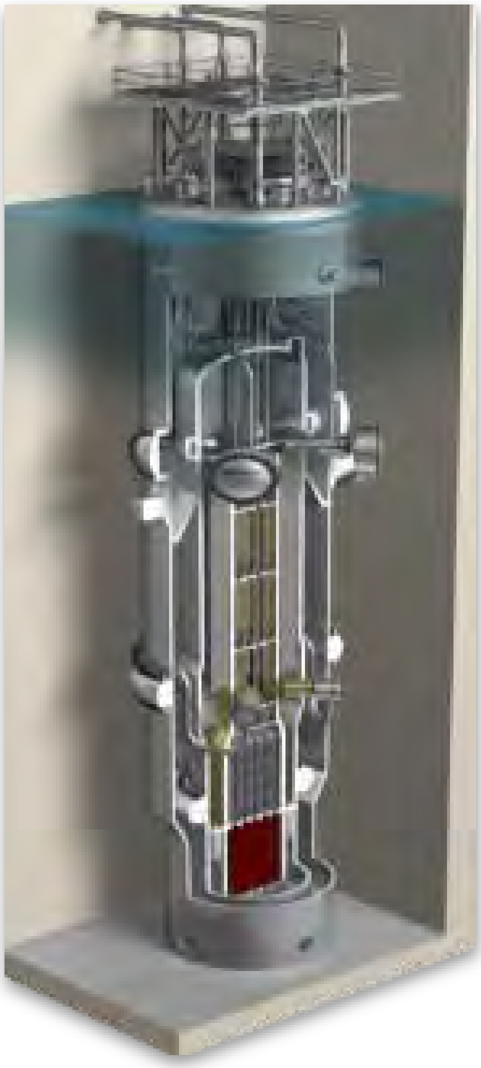
Operation without soluble boron in the primary coolant allows significant reduction in environmental discharges and concomitant simplification of the waste treatment systems. It is a design goal to achieve a zero-discharge plant.

11. 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
2025	Projected start of first of a kind construction
2030	Planned first of a kind commercial operation



NuScale (NuScale Power, LLC, United States of America)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NuScale Power, LLC, United States of America
Reactor type	Integral PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	200 / 60 (gross)
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	13.8 / 4.3
Core Inlet/Outlet Coolant Temperature (°C)	265 / 321
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	37
Fuel enrichment (%)	< 4.95
Core Discharge Burnup (GWd/ton)	> 30
Refuelling Cycle (months)	24
Reactivity control mechanism	Control rod drive, boron
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	140 000
RPV height/diameter (m)	17.7 / 2.7
Seismic Design (SSE)	0.5g horizontal and 0.4g vertical peak ground accelerations
Fuel cycle requirements / Approach	Three stage in-out refuelling scheme
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 customer's energy demands. The 60 MW(e) NPM provides power in increments that can be scaled to 720 MW(e) gross in a single facility with twelve modules. A twelve-module configuration is the 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 for 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. Main Design Features

(a) Design Philosophy

The NuScale plant design adopts design simplification, 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 mean 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 U_{235} enrichment is below the current U.S. manufacturer limit of 4.95% enrichment.

(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 Coolant System

The reactor coolant system (RCS) is a subsystem of the NPM that provides for the circulation of the primary coolant relying on natural circulation. Hence, the RCS does not require reactor coolant pumps or an external piping system. The RCS includes the reactor pressure vessel (RPV) and integral pressurizer, the reactor vessel internals, the reactor safety valves, RCS piping inside the containment vessel and others.

(f) Reactor Pressure Vessel and Internals

The RPV consists of a cylindrical steel vessel with an inside diameter of 2.7 m, an overall height of approximately 17.7 m, and is designed for an operating pressure of 13.8 MPa. The upper and lower heads are torispherical 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.

(g) 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.

(h) Pressurizer

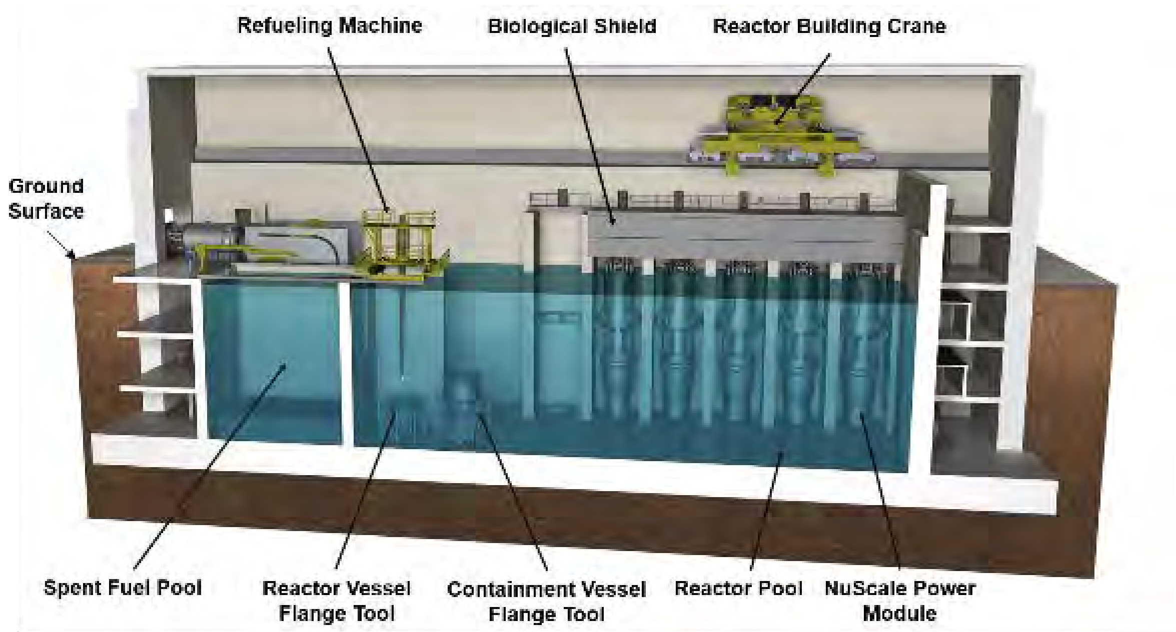
The internal pressurizer provides the primary means for controlling reactor coolant system pressure. It is designed to maintain a constant reactor coolant pressure during operation. Reactor coolant pressure is increased by applying power to a bank of heaters installed above the pressurizer baffle plate.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

4. Safety Features

The NuScale plant adopts a set of engineered safety features designed to provide reliable long-term core cooling under all conditions, including severe accident mitigation. They include integral primary system configuration, a containment vessel, passive heat removal systems, and severe accident mitigation features.

(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 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 functions of the Containment Vessel (CNV) are to contain the release of radioactivity following postulated accidents, protect the RPV 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 4.5 m and an overall height of 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, approved by the US Nuclear Regulatory Commission, is based on the fundamental I&C design principles of independence, redundancy, predictability, repeatability, and diversity. 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. The FPGA technology is not vulnerable to cyber-attacks. The NuScale design effectively integrates human factors engineering (HFE) into the development, design, and operation of the plant.

7. Plant Layout Arrangement

(a) Reactor Building

The NuScale plant consists primarily of a reactor building, a control room building, two turbine-generator buildings, a radwaste 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 below-grade reactor pool and the reactor building are 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 use graphical representations of plant systems and components.

(c) Balance of Plant

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.

8. Design and Licensing Status

In December 2016, NuScale Power submitted the Design Certification Application (DCA) to the NRC. The design certification review is on track with a target completion date of September 2020, with final design approval expected in 2021. The first plant owner, the Utah Associated Municipal Power Systems, has a target commercial operation date of 2027 for the first plant, to be built in Idaho.

9. Fuel Cycle Approach

The three-batch refuelling is conducted on a 24-month refuelling cycle, in an ‘in-out’ shuffle scheme. During the refuelling process, one-third of the fuel assemblies are removed from the NPM and placed in the spent fuel pool. The pool is connected to the ultimate heat sink, and hence, protected by the reactor building. The spent fuel storage racks include sufficient storage for approximately 18 years of operation, including five defective fuel assemblies and for non-fuel core components such as a control rod assembly.

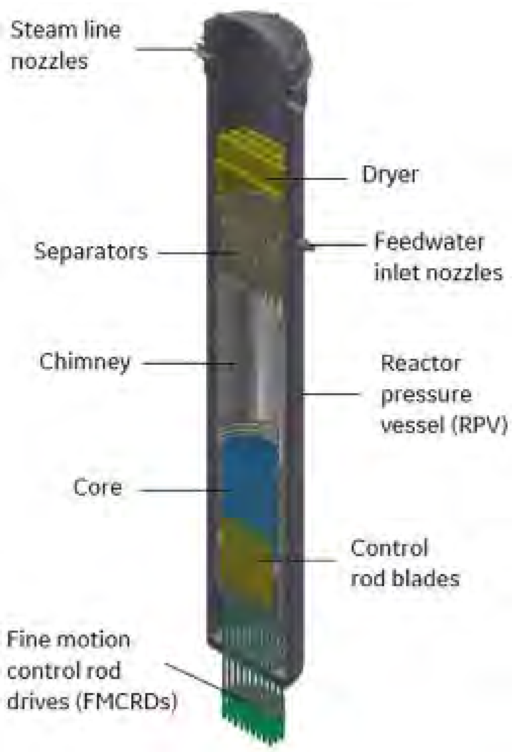
10. Waste Management and Disposal Plan

Removed assemblies are stored in the used fuel pool for initial cool-down and later moved to an on-site dry-cask storage. The plant design includes sufficient on-site storage space for all the spent fuel produced during the 60-year life of the plant. Final disposal is expected to be in a national fuel repository when available.

11. Development Milestones

2003	Initial concept developed and integral test facility operational
2007	NuScale Power was formed
2012	Twelve-reactor simulated control room was commissioned
2016	Design certification application was submitted to the US Nuclear Regulatory Commission
2020	NuScale design certification review completion
2023	Start fabrication/construction of first full-scale NuScale power plant (NPP) in the United States
2027	First commercial NuScale plant targeted to be operational in Idaho

BWRX-300 (GE-Hitachi Nuclear Energy, USA and Hitachi-GE Nuclear Energy, Japan)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	GE-Hitachi Nuclear Energy, United States and Hitachi-GE Nuclear Energy, Japan
Reactor type	Boiling water reactor
Coolant/moderator	Light-water / light-water
Thermal/electrical capacity, MW(t)/MW(e)	870 / 270-290
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	7.2 / Direct Cycle
Core Inlet/Outlet Coolant Temperature (°C)	270 / 287
Fuel type/assembly array	UO_2 / 10x10 array
Number of fuel assemblies in the core	240
Fuel enrichment (%)	3.40 (avg) / 4.95 (max)
Core Discharge Burnup (GWd/ton)	49.5
Refuelling Cycle (months)	12-24
Reactivity control mechanism	Rods and Solid Burnable Absorber (B_4C, Hf, Gd_2O_3)
Approach to safety systems	Fully passive
Design life (years)	60
Plant footprint (m^2)	8400
RPV height/diameter (m)	26 / 4
RPV weight (metric ton)	485
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Open fuel cycle utilizing standard BWR fuel
Distinguishing features	Natural circulation BWR utilizing RPV isolation valves and isolation condenser system that enable dry containment and elimination of safety relief valves
Design status	Pre-licensing initiated in UK, Canada, US

1. Introduction

The BWRX-300 is a design-to-cost 300 MW(e) water-cooled, natural circulation SMR utilizing simple, natural phenomena driven safety systems. It is the 10th generation of the Boiling Water Reactor (BWR) and represents the simplest, yet most innovative BWR design since the General Electric began developing nuclear reactors in 1955. The BWRX-300 is an evolution of the U.S. NRC-licensed 1520 MW(e) ESBWR. The design has been developed with a strict adherence to a philosophy which follows the IAEA Defense-in-Depth guidelines.

2. Target Application

The target applications of BWRX-300 include base load electricity generation, load following electrical generation within a range of 50% to 100% power, district heating and process heat up to 287°C. It is designed to provide clean, flexible energy generation that is cost competitive with natural gas fired plants.

3. Main Design Features

(a) Design Philosophy

Evolved from the licensed ESBWR, the BWRX-300 design optimizes the cost of construction, operation, maintenance, staffing and decommissioning.

(b) Nuclear Steam Supply System

The BWRX-300 is designed utilizing the proven supply chain of the ABWR and the natural phenomena driven safety features from the ESBWR, for its primary circuit, or nuclear boiler system. Components in this system include the Reactor Pressure Vessel (RPV), fine motion control rod drives (FMCRDs), control blades, chimney, separators, and dryer. The RPV and the chimney height are scaled optimally to the thermal output and natural circulation of the BWRX-300. The nuclear boiler system delivers steam from the RPV to the turbine main steam system; and delivers feedwater from the condensate and feedwater system to the RPV. It also provides overpressure protection of the reactor coolant pressure boundary (RCPB).

(c) Reactor Core

The BWRX-300 contains 240-bundles utilizing currently available, industry proven GNF2 fuel. The core lattice configuration has equal spacing on the control rod and non-control rod sides which provides increased shutdown margin that can accommodate variations in burnup imposed by load following. GNF2 fuel bundles contain a 10x10 array of 78 full length fuel rods, 14 part length fuel rods and two large central water rods. The fuel bundles are arranged in a near cylindrical configuration located inside a core shroud. The coolant flows upward through the core. The BWRX-300 reactor core comprises fuel assemblies, control rods, and nuclear instrumentation.

(d) Reactivity Control

Reactivity control is provided by control rods loaded with either B_4C or Hf neutron absorbers and burnable neutron absorber loaded in the fuel rods. The control rods are moved using the FMCRDs that are successfully deployed in the ABWR and part of the ESBWR design. FMCRDs have two independent means of moving the control rods–motor driven fine control for reactivity control during normal operation and hydraulic rapid insertion during a scram.

(e) Reactor Pressure Vessel and Internals

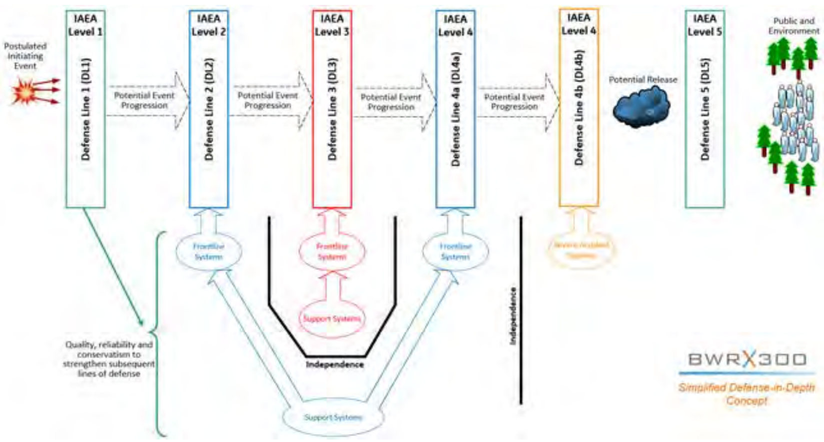
The BWRX-300's RPV assembly consists of a pressure vessel with removable head, internal components and appurtenances. The major reactor internal components include the core, core plate, top guide, control rod guide tube, control rod drive housings, and orificed fuel supports, chimney, steam separator, and steam dryer assembly.

(f) Reactor Coolant System

The reactor coolant system (RCS) is natural circulation driven and provides cooling of the reactor core in all operational states and all postulated off normal conditions. The BWRX-300 leverages natural circulation modelling and operational information from the ESBWR and the Dodewaard BWR in the Netherlands. The relatively tall RPV permits natural circulation driving forces to produce abundant core coolant flow.

4. Safety Features

The basic BWRX-300 safety design philosophy is built on utilization of inherent margins (e.g., larger structure volumes and water inventory) to eliminate system challenges and can easily accommodate transients. Natural driven phenomena safety-related systems mitigate accidents without the need for AC power. The BWRX-300 defence-in-depth (D-in-D) concept uses Fundamental Safety Functions (FSF) to define the interface between the defense lines and the physical barriers. In a given plant scenario, if the FSFs are performed successfully, then the corresponding physical barriers will remain effective.

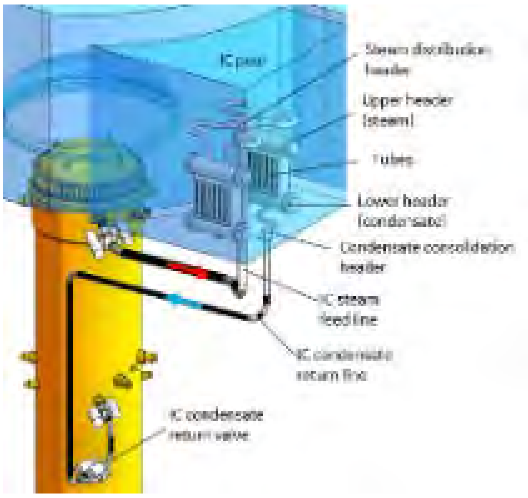


(a) Engineered Safety System Approach and Configuration

There are two passive cooling systems for BWRX-300. The first is the isolation condenser system (ICS) which removes decay heat after a reactor isolation event. The other is the passive containment cooling system (PCCS) which removes the decay heat and maintains the containment within its pressure limits for design basis accidents.

(b) Decay Heat Removal System

The ICS removes decay heat after any reactor isolation during power operations and provides overpressure protection in accordance with ASME BPV code, Section III, Class 1 equipment. The ICS consists of three independent loops, each containing a heat exchanger, with capacity of 33 MW(t). The ICS tubes condense steam from the RPV on the inside surface and transfer heat on the outside surface to the IC pool which is vented to the atmosphere. This steam condensation and gravity allow BWRX-300 to cool itself for a minimum of 7 days without power or operator action.



(c) Containment System

The containment surrounds the RPV, FMCRDs, piping systems and isolation valves. The design pressure and temperature of the containment are within values used in the operating BWR fleet, the ABWR and ESBWR. The RPV isolation vales limit the steam released to the containment during a postulated large break LOCAs to the point that the suppression pool utilized in almost all previous BWRs is not needed. Postulated small break LOCAs are copped by the ICS, PCCS and dry containment for a minimum of 72 hours without AC power or operator action. The PCCS is a passive system consisting of several low-pressure, heat exchangers in the containment. The heat in the containment is transferred to the reactor cavity pool which is located above the containment upper head and is filled with water during normal operation. The reactor cavity pool is vented to the atmosphere. The PCCS heat exchangers are always in service when the reactor is operating.

5. Plant Safety and Operational Performances

The D-in-D approach utilized in the BWRX-300 results in an internal event cord damage frequency less than 10^{-7} per year and a large release frequency less than 10^{-8} per year. The lessons learned and best practices of the BWR fleet have been applied to the BWRX-300 with the expected available factor greater than 95%. The BWRX-300 is capable to operate under the load following range of 50 to 100% with a ramp rate of 0.5% per minute. Normal manpower during operation is expected to be 75 people for all shifts. Cycle lengths between 12 and 24 months can be accommodated with refuelling durations between 10 and 20 days.

6. Electrical Systems

The BWRX-300 electrical system is a completely integrated power supply and transmission system for the power plant. It is divided into subsystems based on safety classification. Each subsystem has appropriate levels of hardware and software quality for the systems it powers, to provide reliable power to various plant electrical loads, and a transmission path for the main generator to the utility switchyard/grid.

7. Instrumentation and Control Systems

The BWRX-300 I&C system (also referred to as the Distributive Control and Information System or "DCIS") is a completely integrated control and monitoring system for the power plant. The I&C comprises three main platforms. Each platform has appropriate levels of hardware and software quality (corresponding to the systems they control), and provides control, monitoring, alarming and recording functions.

8. Plant Layout Arrangement

The reference site for BWRX-300 fits within a 170 m x 280 m footprint, which includes the power block, switchyard, cooling tower, site office, parking lot, warehouse, and other supporting facilities.

(a) Reactor Building

The reactor building extends below grade where the primary containment and RPV mostly reside. The refuelling floor is above grade level. The Reactor Building contains all of the safety related components in the plant. The underground construction of the reactor building utilizes proven techniques in the mining and shaft sinking industries to minimize the use of concrete and engineered backfill.

(b) Balance of Plant

The balance of plant of BWRX-300 includes the turbine building, radwaste building, control building and yard. The turbine receives high-quality dry steam directly from the RPV, the condenser system cooled by the ultimate heat sink and the feedwater system that returns the coolant into the reactor. The balance of plant also has process systems composed of the plant service water system (PSWS), the component cooling water system (CCW), the makeup water system (MWS), the condensate storage and transfer system (CSTS), and the chilled water system (CWS).

9. Design and Licensing Status

Licensing activities for the BWRX-300 have been initiated in three countries. Pre-licensing application activities are underway in the United States. A combined Phase 1 and 2 Vendor Design Review is underway in Canada. In the UK, the BWRX-300 completed a BEIS-funded Mature Technology evaluation. One of the design goals is also to be able of being licensed internationally with initial operation dates in 2027 and 2028.

10. Fuel Cycle Approach

The BWRX-300 has the same open fuel cycle as operating BWRs. During every refuelling outage, 15% to 25% of the bundles in the core are replaced fresh fuel. Fresh fuel stays in the core for several cycles until it is discharged. When removed from the core, used fuel is stored in the fuel pool inside the reactor building for six to eight years when it is then transferred to storage casks that can be removed from the reactor building and stored outside.

11. Waste Management and Disposal Plan

The BWRX-300 utilizes the lessons learned and best practices from the decades of operational experience of the BWR fleet in waste management and disposal. Waste minimization is one of the focus areas during the design phase to ensure that the unit has minimal environmental impact. Waste that is generated will be segregated for optimal treatment, storage and final disposal. Gaseous wastes will be removed from the steam system by steam jet air ejectors and passed through charcoal beds for absorption.

12. Development Milestones

2017	Conceptual design started
2018	BEIS funded Mature Technology Evaluation the UK ONR
2019	Pre-application activities initiated with the US NRC
2020	Combined Phase 1 and 2 Vendor Design Review initiated with CNSC in Canada
2022	First construction permit applications expected to be submitted
2024	Start of construction for first unit expected
2027	Commercial operation of first unit expected

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

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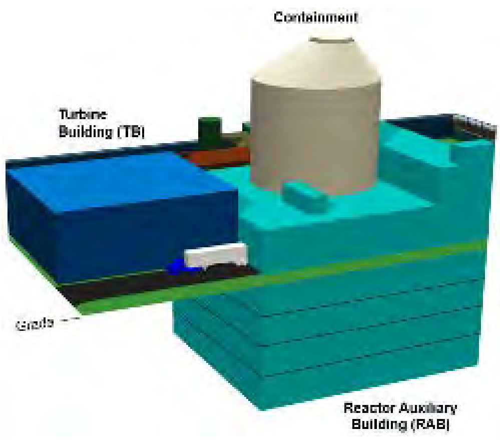
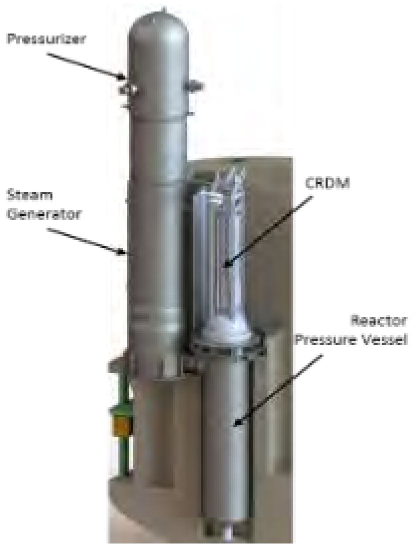


SMR-160 Nuclear Island



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
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 3.4
Core Inlet/Outlet Coolant Temperature (^oC)	229 / 321
Fuel type/assembly array	UO_2 pellet / square array
Number of fuel assemblies in the core	57
Fuel enrichment (%)	4.95 (maximum)
Core Discharge Burnup (GWd/ton)	45 (maximum, initial design)
Refuelling Cycle (months)	24 nominal (flexible)
Reactivity control mechanism	Control rods and soluble boron
Approach to safety systems	Fully Passive; utilize varied phenomena and redundancy.
Design life (years)	80
Plant footprint (m^2)	20 500
RPV height/diameter (m)	15 / 3
RPV weight (metric ton)	295 (with fuel and internals)
Seismic Design (SSE)	0.3g, derived from the NRC Regulatory Guide 1.60
Fuel cycle requirements / Approach	Approximately 1/3 batch fraction removed each refuelling outage
Distinguishing features	Defence-in-Depth with passive safety cooling systems and active non-safety systems; critical components below grade.
Design status	Development for a Preliminary Safety Analysis report. in support of commercial project and pre-licensing engagements. Completed Phase 1 of vendor design review with CNSC in Canada.

1. Introduction

The SMR-160 has been developed by Holtec International as an advanced PWR small modular reactor producing 525 MW thermal power or 160 MW electric power. The plant design incorporates robust passive safety systems to achieve a highly reliable design that protects owner's investment from all postulated accidents, sabotage, or inadvertent human actions. The SMR-160 design is 'walk-away safe' – no operator actions are necessary to cope with design basis accidents and safely reject decay heat. 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. 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.

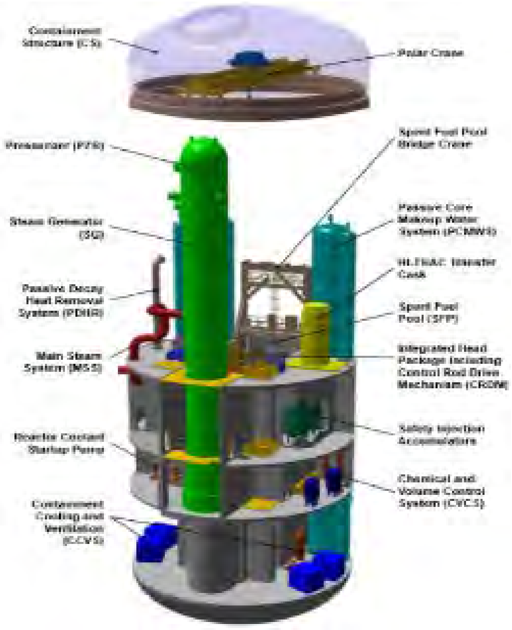
2. Target Application

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 siting in water scarce locations using 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. Main 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 during design basis accidents, while ensuring the SMR-160 design remains inherently securable, fabricable, constructible, and economically competitive in world-wide markets.



SMR-160 Containment Internals

(b) Nuclear Steam Supply System

The SMR-160 is a pressurized water reactor with a naturally circulating reactor coolant system (RCS) 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 contains both the hot leg and the cold leg in concentric ducts. 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 eliminates high-pressure turbine stages.

(c) Reactor Core

The SMR-160 employs an efficient reactor core design that uses a traditional reload shuffle. The reactor core contains standard length 17x17 PWR fuel assemblies just like those currently available from several commercial suppliers, along with typical control rod assemblies. An owner can be assured of a diverse supply chain for critical reactor components like fuel, rod cluster control assemblies, and control rod drive mechanisms. The reactor vessel internals support the reactor core, the control rod assemblies, and the control rod drive shafts within the RPV. The core is designed for a nominal two-year cycle with flexibility for shorter or longer cycles depending upon utility requirements. The SMR-160 core is based on proven PWR technology and operational characteristics and is designed to ensure large margin to thermal-mechanical fuel limits.
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 adjusting soluble boron and movements of control rod assemblies (CRAs). CRAs are positioned by 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.

(d) 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.

(e) Reactor Coolant System

The SMR-160 RCS operates purely by natural circulation, as reactor coolant circulates entirely as a result of the density difference in the primary water and the height of the RPV and steam generator. This ability to circulate is present as long as the fuel assemblies in the core produce heat. There are no reactor coolant pumps in the system. The RCS consists of three major components as shown in the figure above, a Reactor Pressure Vessel (RPV), a Steam Generator (SG), and an integral Pressurizer.

(f) 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.

(g) 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 varied pathways for rejection of decay heat. All safety systems are located inside the robust containment enclosure structure, rendering them secure and safe from external threats. All makeup water needed for a postulated loss of coolant accident (LOCA) is inside containment making the containment isolable during a LOCA reducing possible dose to the public and effects on the environment. Another large inventory of water within a reservoir between the containment enclosure structure and the containment structure provides long-term post-accident coping and allows the decay heat removal function to transition to air cooling for an unlimited coping period after a design basis accident.

(a) Engineered Safety System Approach and Configuration

SMR-160 relies on passive and redundant safety systems that also operate by natural circulation. The passive safety systems ensure safe shutdown can be maintained and decay heat removal occurs for an unlimited period without the need for power, make-up water, or operator actions. If available, active non-safety systems can be used by operators to mitigate events and preclude the need for using the engineered safety systems. 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 and makeup to the RCS during postulated accidents. The system uses passive means such as natural circulation for core cooling and compressed gas expansion and gravity injection for core makeup without the use of active components such as pumps. The PCCS is comprised of four major subsystems:

- Primary decay heat removal system (PDHR)
- Secondary decay heat removal system (SDHR)
- Automatic depressurization system (ADS)
- Passive core make-up water system (PCMWS)

The PDHR directly cools the primary coolant by re-routing the reactor coolant through a heat exchanger and rejecting the heat to a second loop full of water. The second loop rejects its heat to the large annular reservoir (AR) around containment. The SDHR provides an alternative 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 the sealed containment 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 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 potential hazards. In addition to preventing the release of radioactive fission products to the environment, the containment system acts as a large passive heat exchanger. The containment system is partially embedded, with approximately 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.

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 I&C/HSI design. MELTAC is a proven technology, with over 300 reactor-years of operating 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.

7. Plant Layout 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 house all safety systems and the spent fuel pool and share a common basemat with the reactor auxiliary building.

Plant Layout of SMR-160

(b) Reactor Auxiliary Building

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

(c) Balance of Plant

The steam turbine and associated systems are housed within the turbine building structure at grade level. The SMR 160 features an axial/side exhaust steam turbine, optionally configured for air cooled condensation. SMR-160 is also adaptable to process applications such as desalination and district heating. The SMR-160 electric power system consists of the main generator, main transformer, auxiliary transformers, non-safety diesel generators and Class 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. Fuel Cycle Approach

The SMR-160 fuel cycle is designed to discharge approximately one third of the fuel assemblies in the core each refuelling cycle, along with shuffling of a portion of the remaining fuel assemblies. The spent fuel is stored briefly in the spent fuel pool, which is uniquely protected within the same containment as the reactor. New fuel assemblies are delivered using Holtec International’s HI-STORM system, which has decades of operating experience throughout the global light water reactor fleet. This allows the SMR-160 to eliminate significant complex handling equipment. The HI-STORM system has received multiple licensing approvals from the U.S. Nuclear Regulatory Commission.

10. Waste Management and Disposal Plan

High level waste management and disposal for the SMR-160 uniquely benefits from the integration of Holtec International’s dry storage technologies. After removal of spent fuel from the spent fuel pool within a Multi-Purpose Canister called an MPC-37, all spent fuel for the life of the plant can be stored on-site within an array of HI-STORM UMAX modules (an underground vertical storage cask design). The MPC-37 is a dual-purpose canister licensed for transportation off-site within the HI-STAR 190 transportation overpack.

11. Development Milestones

2012	Conceptual design of SMR-160 commencement
2015	Conceptual design completed for SMR-160
2020	Preliminary design completed for SMR-160
2021	Ready for commercialization using a construction permit based process

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

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
NSSS Operating Pressure (primary/secondary), MPa	15.5
Core Inlet/Outlet Coolant Temperature (ºC)	294 / 324
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies in the core	89
Fuel enrichment (%)	< 5
Core Discharge Burnup (GWd/ton)	> 62
Refuelling Cycle (months)	24
Reactivity control mechanism	CRDM, boron
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	65 000
RPV height/diameter (m)	28 / 3.7
Seismic Design (SSE)	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 can also 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. Main 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 U_{235}. Approximately 40% 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 the instrumentation and electrical penetrations to facilitate removal during refuelling.

(f) Reactor Coolant System

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 by 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.

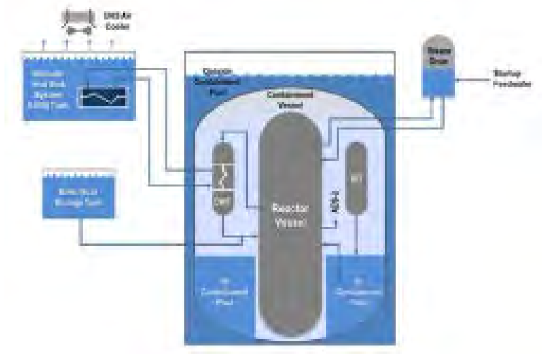
(a) Engineered Safety System Approach and Configuration

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%.

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 Layout 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 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% steam bypass capability to handle turbine trip, both preventing a need for a reactor scram.

Plant Layout of Westinghouse SMR.

(c) 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.

(d) Electric Power Systems

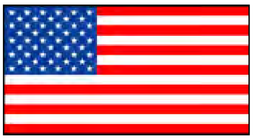
The Westinghouse SMR onsite power system consists of a main ac power system and a direct current (DC) power system. The main ac power system is a non-Class 1E system and does not perform any safety-related functions. The plant dc power system is composed of the independent Class 1E and non-Class 1E DC 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



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

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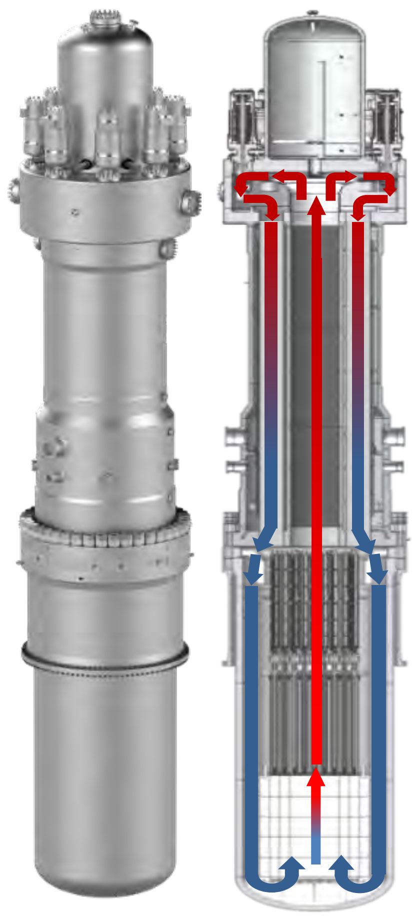


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Generation mPower, LLC, United States of America
Reactor type	Integral PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	575 / 195
Primary circulation	Forced circulation
NSSS Operating Pressure (primary), MPa	14.8
Core Inlet/Outlet Coolant Temperature (°C)	290.5 / 318.9
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	69
Fuel enrichment (%)	< 5
Core Discharge Burnup (GWd/ton)	< 40
Refuelling Cycle (months)	24
Reactivity control mechanism	Control rods
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	157 000
RPV height/diameter (m)	27.4 / 4.15
Seismic Design (SSE)	0.3g
Distinguishing features	In-vessel control rod drives mechanism
Design status	Conceptual design

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. Main 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 by the strong negative moderator temperature coefficient by control of the secondary side feedwater flow rates.

(e) 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.

(f) 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.

(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 refuelling and allows for inspection and maintenance in parallel with fuel exchange.

(h) Pressurizer

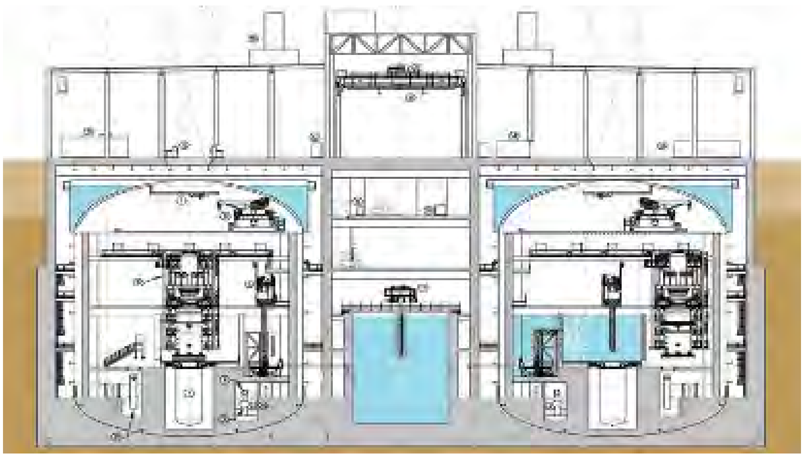
The integrated electrically heated pressurizer located at the top of the RPV maintains a nominal 14.8 MPa. Reactor coolant pressure is controlled by the heaters and steam space spray.

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 refuelling 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 refuelling 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 centers is crucial to the plant's capability to remove heat by natural circulation following a loss of forced circulation. By vertically separating these two components within an integral pressure vessel, the design of the mPower reactor encourages this natural convection heat removal rather than requiring engineered pump-driven systems. The mPower design includes analogous circuits to remove heat from the secondary and from the containment systems. In the former instance, the system is a non-safety, 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 Layout 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% 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

WATER COOLED
SMALL MODULAR REACTORS
(MARINE BASED)

KLT-40S (JSC "Afrikantov OKBM", Russian Federation)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", Rosatom, Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	150 / 35
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	12.7
Core Inlet/Outlet Coolant Temperature (°C)	280 / 316
Fuel type/assembly array	UO_2 pellet in silumin matrix
Number of fuel assemblies in the core	121
Fuel enrichment (%)	18.6
Core Discharge Burnup (GWd/ton)	45.4
Refuelling Cycle (months)	30-36
Reactivity control mechanism	Control rod driving mechanism
Approach to safety systems	Active (partially passive)
Design life (years)	40
Plant footprint (m^2)	4320 (Floating NPP)
RPV height/diameter (m)	4.8 / 2.0
RPV weight (metric ton)	N/A
Seismic Design (SSE)	9 point on the MSK scale
Distinguishing features	Floating power unit for cogeneration of heat and electricity; no onsite refuelling; spent fuel take back
Design status	Connected to the grid in Pevek in December 2019. Entered full commercial operation in May 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. Main Design Features

(a) Design Philosophy

KLT-40S is the reactor for Akademik Lomonosov FNPP, 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

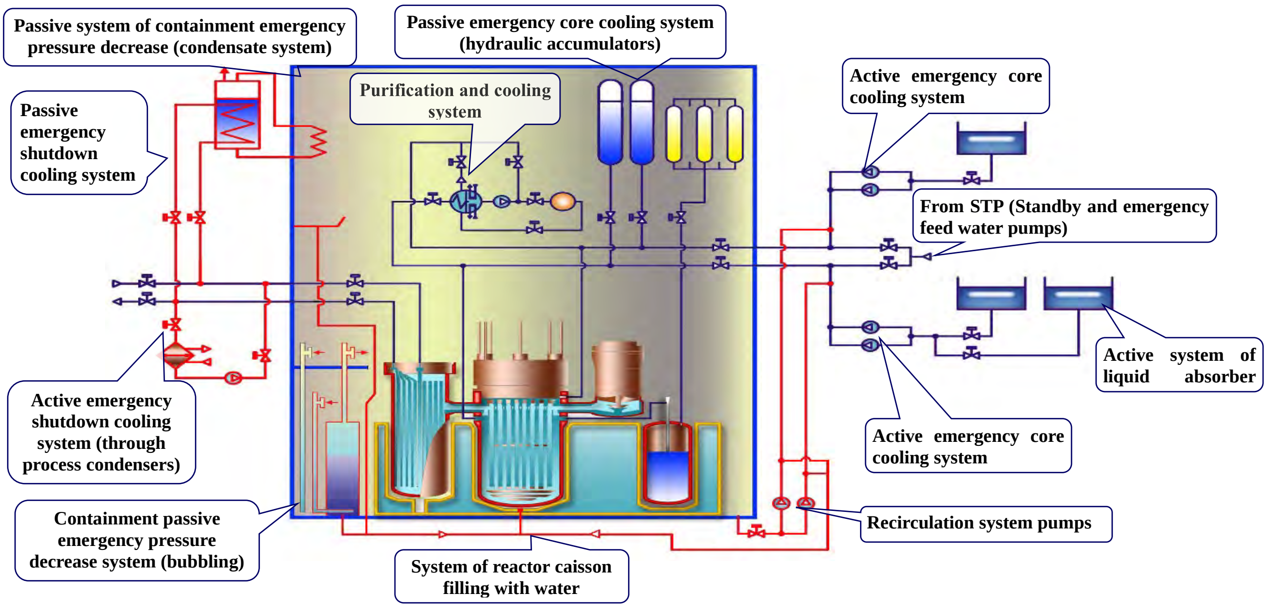
The active components of the protection system are scram actuators for six (6) groups of the control rods.

(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 hours. 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 Layout 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.

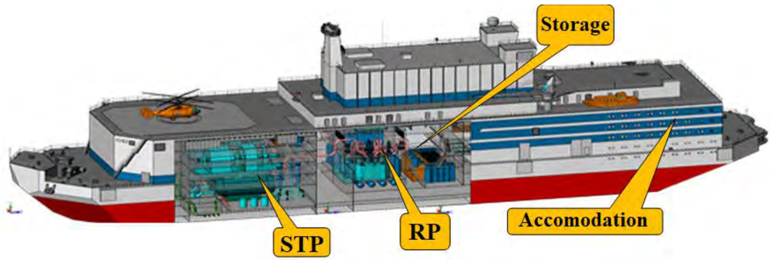
(a) Reactor Building

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. The KLT-40S is a modified version of the commercial KLT-40 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. The keel of the FNPP carrying the KLT-40S, the Akademik Lomonosov in the Chukotka region, was laid in 2007. The construction of Akademik Lomonosov was completed in 2017. The Akademik Lomonosov has started commercial operation in December 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
Summer 2019	Transportation of FPU to the town of Pevek
December 2019	Connected to grid on 19th of December in Pevek
May 2020	Fully commissioned in Pevek on 22nd of May

RITM-200M (JSC "Afrikantov OKBM", Russian Federation)

BASIC TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", Rosatom, 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
NSSS Operating Pressure (primary/secondary), MPa	15.7 / 3.83
Core Inlet/Outlet Coolant Temperature (°C)	277 / 318
Fuel type/assembly array	UO_2 (metal-ceramic fuel) pellet/hexagonal
Number of fuel assemblies in the core	241
Fuel enrichment (%)	< 20
Core Discharge Burnup (GWd/ton)	-
Refuelling Cycle (months)	Up to 120
Reactivity control mechanism	Control and protection system rod drive mechanism
Approach to safety system	Combined (active and passive) system
Design life (years)	60
NPP footprint (m^2)	3360
RPV height/diameter (m)	8.6 / 3.45
RPV weight (metric ton)	265
Seismic Design (SSE)	0.3g
Fuel Cycle Requirements or Approach	Without on-site refueling
Distinguishing features	Integral reactor, in-vessel corium retention, double containment
Design status	6 prototype reactors were manufactured and installed on icebreakers (two are in the process of testing)

1. Introduction

JSC "Afrikantov OKBM"'s RITM series reactors - RITM-200 and RITM-200M are the state-of-the-art development in SMR line. They have been designed by the JSC "Afrikantov OKBM" and have incorporated all the best features from its predecessors. Floating NPPs equipped with RITM reactors are available for commercial implementation in medium/long term. RITM-200M reactor is a development from the RITM line with refuelling cycle increased up to 10 years.

2. Target Application

The RITM-200M design was developed for the Optimized Floating Power Unit (OFPU). The OFPU is a power facility in the form of a compact non-self-propelled vessel, having two RITM-200M reactor plants. The Floating power units based on RITM-200M ensure that needs of small settlements or industrial enterprises are covered, as well as the power expanding when need for electrical power is increasing or transfer of the energy source to a new deployment site upon disappearance of their necessity (e.g. upon completion of the development of mineral deposits). OFPU can provide electricity to domestic and industrial consumers. OFPU can also be used for heat supply and water desalination purposes when installing additional equipment. Such power units will become a powerful factor of stability in the development of the region not covered by the single energy system and requiring reliable and economically competitive energy sources.

3. Main Design Features

(a) Design Philosophy

RITM series reactors are the evolutionary development of the reactors (OK-150, OK-900 and KLT-40 series) for Russian nuclear icebreakers with a total operating experience of more than 60 years (more than 400 reactor-years). Due to incorporation of steam generators into the reactor pressure vessel, the reactor system and containment is very compact compared to the KLT-40S. The RITM design makes it possible to increase electric output (by 40% more) and to reduce dimensions (by 45% less) and the mass (by 35% less) in comparison with KLT-40S. The integral reactor configuration almost eliminates the classic large loss-of-coolant accident (LOCA). Active and passive safety systems are applied in the concept to achieve the necessary safety and reliability level due to principles of redundancy, physical separation, and functional independence.

(b) Nuclear Steam Supply System

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

(c) Reactor Core

In the RITM series, a low enriched cassette core is applied similar to the KLT-40S, which ensures long-term operation without refuelling and meets international non-proliferation requirements. The core consists of 241 fuel assemblies with 20% enrichment. The service life of the core is up to 10 years.

(d) Reactivity Control

Control rods are used for reactivity control. A group of CPS drive mechanisms is provided for compensation of the excessive reactivity during start-up, operation at power and emergency shutdown of the reactor. Safety rods are provided for quick reactor shutdown and its holding in subcritical state in case of an accident. The design of control and safety rods has been developed on the basis of the drives proven in KLT-40S reactor plant.

(e) Reactor Coolant System

The reactor pressure vessel (RPV) is a thick-walled cylindrical pressure vessel with an all-welded bottom cover and a removable top cover. The reactor plant is designed as an integral vessel with main circulation pumps (MCP) located in separate external hydraulic chambers connected by nozzles. It also includes four steam generator cassettes. Each of the four SGs has three rectangular cassettes; and the four main circulation pumps are installed in a cold loop of the circulation circuit and are divided into four independent circuits. The SGs generate steam of 295°C at 3.82 MPa flowing at 261 (280) t/h. The MCPs are single-stage vane type and have a sealed asynchronous motor with one winding.

(f) Steam Generator

In the RITM series reactors, once-through SGs are applied. The configuration of the steam generating cassettes makes it 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 large-break accidents of primary pipeline. 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 system is based on the defence-in-depth principle combined with the inherent safety features and use of passive systems. The inherent safety features enable to carry out automatic regulation of power flow density and automatic reactor 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 conceptualization of a ‘passive reactor’, resistant to possible abnormalities. RITM optimally combines passive and active safety systems to ensure normal operation and sustainability during design basis accidents.

- Passive pressure reduction and cooling systems area available (system reliability is verified by tests);
- The pressurizer system is divided into two independent sections to minimize the potential coolant leakage;
- Main circulation loop of the primary circuit is located in a single vessel;
- The steam header of primary coolant circulation is added providing safety of the plant during SG and MCP failures.

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

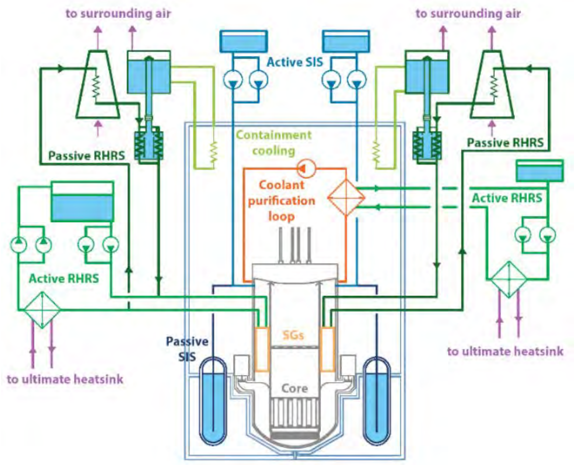
(a) Approach to and Configuration of the Engineered Safety System

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 trains 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. CPS rods drop into the core by gravity or using a spring when the electromechanical clutches are de-energized ensuring that the reactor will shut down even in case of total plant black out.

(b) Decay 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 coolant purification loop.
- Two passive safety loops with natural coolant circulation through steam generators from water tanks. Evaporated in steam generators, water condenses in air cooled heat exchangers and in water heat exchanger and then flows back to steam generators. After complete water evaporation from the tanks connected to water heat exchangers, the air-cooled exchangers continue provide cooling for unlimited time. Combination of air and water heat exchangers enables to minimize dimensions of the heat exchangers and water tanks.



RITM Series Safety System

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

(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 train and consists of:

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

In combination with the residual heat removal system the passive safety trains provide 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 inside containment with overall dimensions of 6.6m×6.4m×16.2m 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 containment integrity is ensured by overpressure relief valve, containment cooling system, and a passive autocatalytic recombiner.

5. Plant Safety and Operational Characteristics

The main characteristics are:

- OFPU construction and first fueling in the country of origin;
- Transportation to operation site through the territorial sea of transit countries;
- Power and heat production at operation site in host country (up to 10 years until refueling);
- Return to the country of origin for maintenance and refueling;
- Maintenance and refueling in the country of origin;
- Radwaste management in the country of origin
- Return to operation site in the customer's country.

6. Monitor and Control Systems

An automated control system is provided in the RITM based nuclear power plant to monitor and to control plant processes. This system possesses necessary redundancy with regard to safety function fulfilment and ensures both automated and remote control of the power plant.

7. General Layout of the Plant

The Optimized Floating Power Unit with RITM-200M reactor plant is designed to supply electricity and desalinated water to coastal or isolated territories, offshore installations, islands, and archipelagos. The OFPU may be rapidly delivered to the site by sea. The only need for its start-up is docking pier and onshore power transmitting infrastructure. The design of the vessel is also being developed to ensure that it is positioned in the open sea without coastal structures. In this case, the energy is supplied to the shore by cable.

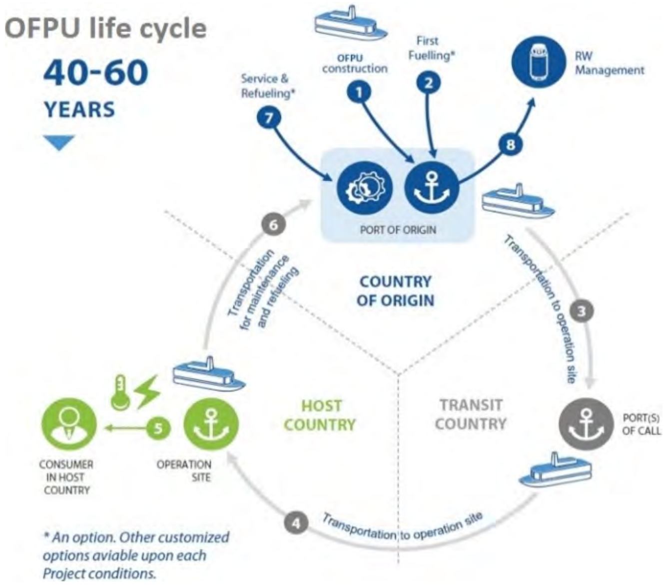
The OFPU with RITM-200M

8. Approach to the Fuel Cycle

The OFPU is delivered to the site with fresh fuel in its reactors. After completion of the fuel cycle, the OFPU returns to the exporting country together with the spent fuel in its reactors. All operations for production, post-reactor maintenance and reprocessing of spent nuclear fuel are performed in the exporting country.

9. Waste Management System and Waste Disposal Plan

Waste storage occurs within the OFPU (in the water area the waste is neither stored nor processed nor disposed of). The relevant waste of the plant is compact, has a low activity level and it is reliably isolated from the biosphere. Absence of its impact on marine organisms in the deployment water area has been confirmed.



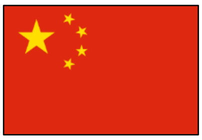
OFPU life cycle

10. Design State and Licensing Status

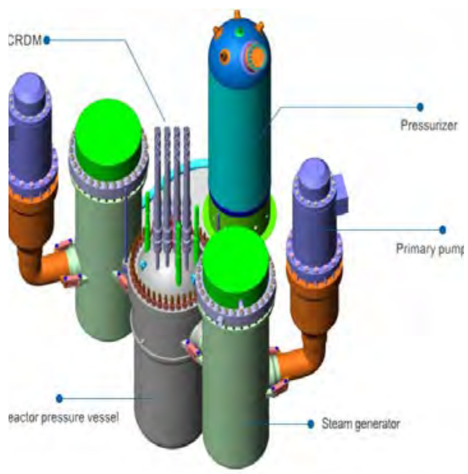
The RITM series design has been developed in conformity with Russian laws, standards and rules for nuclear power plants and safety principles developed by the world community and IAEA recommendations. In the RITM series design, optimal combination of passive and active safety systems is applied. Reactors are currently manufactured and installed on board nuclear icebreakers; the OFPU design is under development.

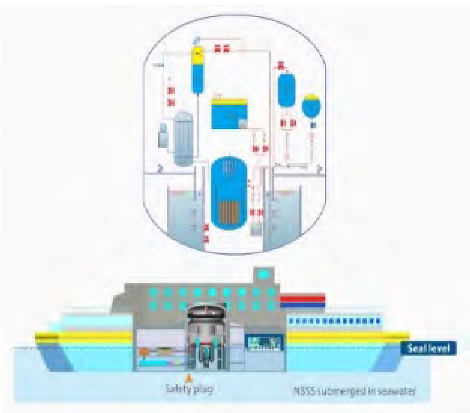
11. Development Milestones

2016	First RITM-200 was installed on board *Arktika* icebreaker
2020	*Arktika* icebreaker is under testing
2020	Conceptual design of the OFPU with RITM-200M



ACPR50S (CGNPC, China)





Reactor Building Inside Ship

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	China General Nuclear Power Group (CGNPC), 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
NSSS Operating Pressure (primary/secondary), MPa	15.5
Core Inlet/Outlet Coolant Temperature (°C)	299.3 / 321.8
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	37
Fuel enrichment (%)	< 5
Core Discharge Burnup (GWd/ton)	< 52
Refuelling Cycle (months)	30
Reactivity control mechanism	Control rod driving mechanism (CRDM), solid burnable poison and boron solution
Approach to safety systems	Passive
Design life (years)	40
RPV height/diameter (m)	7.2 / 2.2
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. Main 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 2.2m, 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

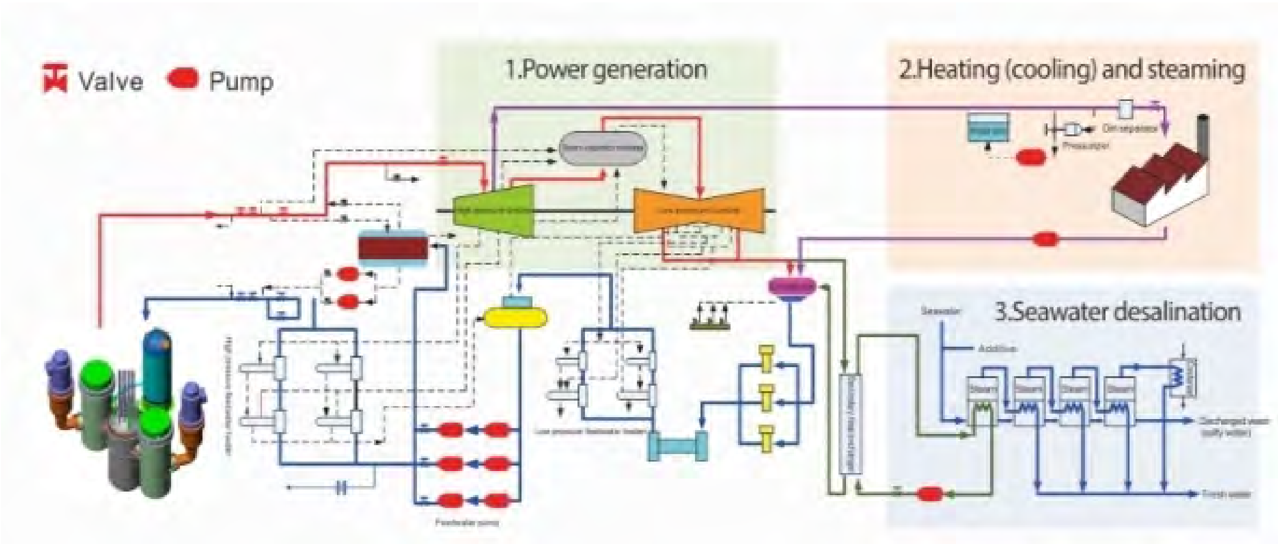
ACPR50S reactor pressure vessel is 2.2m in diameter and 7.2m high. It envelopes and fixes the core and the RPV internals, so that the fission reaction of the nuclear fuel is limited in one space.

(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 m^3, 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% nuclear power (Pn) and can be operated steadily for a long period for any power level of 20% to 100% Pn to satisfy the power demand.

6. Electric Power 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 Layout 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, radwaste treatment are installed in the onshore basement.

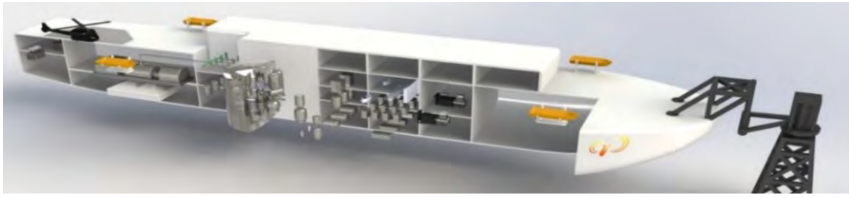
(a) Floating Platform

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 radwaste 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. An industrial demonstration plant of ACPR50S is being planned to be constructed in China.

9. Fuel Cycle Approach

The self-propelled experimental reactor adopts the whole heap refuelling strategy. All 37 fuel assemblies in the full core are discharged after a fuel cycle, then 37 new fuel assemblies are loaded. The core loading scheme adopts high leakage loading, that is, the components with high enrichment degree are arranged in the outer ring of the core, and the components with low enrichment degree are arranged in the inner ring of the core to flatten the radial power distribution. The spent fuel assemblies are placed in the spent fuel pools.

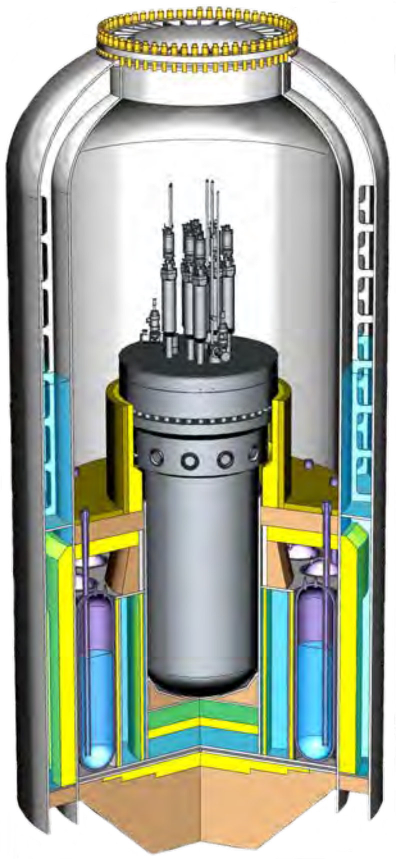
10. Waste Management and Disposal Plan

Liquid radwaste system is designed to prevent or minimize the creation of radioactive liquid effluent, and this achieved, wherever possible, by internal recycling. Gaseous radwaste system is designed to minimize the radioactivity associated with the resulting environment discharge. Solid radwaste system is designed to collect, preliminarily treat and temporarily store the solid wastes during the operation. And the wastes will be sent to disposal site for final disposal.

11. Development Milestones

2012	Starting Conceptual Design and formulating plan of theoretical tests
2014	Completion of Overall Design
2020	Completion of Preliminary Design

ABV-6E (JSC “Afrikantov OKBM”, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, Rosatom, 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
NSSS Operating Pressure (primary/secondary), MPa	16.2
Core Inlet/Outlet Coolant Temperature (°C)	250 / 325
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	121
Fuel enrichment (%)	<20
Core Discharge Burnup (GWd/ton)	N/A
Refuelling Cycle (months)	120-144
Reactivity control mechanism	Control rod driving mechanism
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	20 000 (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 to 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.

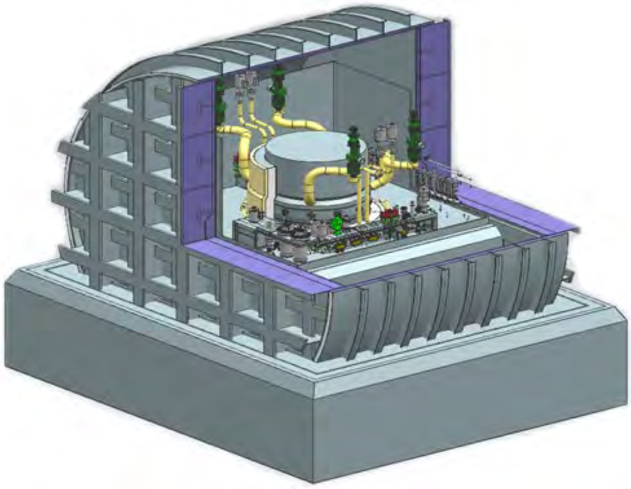
3. Main Design Features

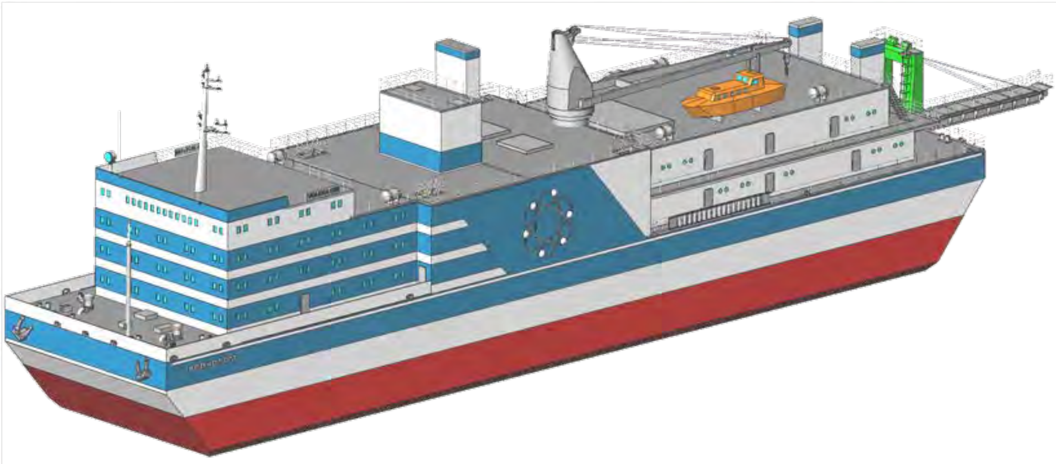
(a) Design Philosophy

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 U_{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) Decay 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

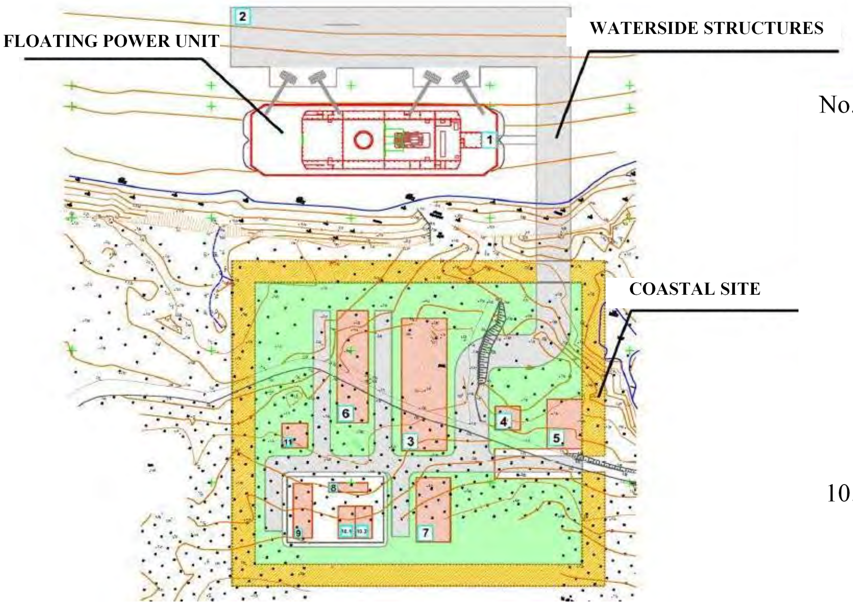
The metal-and-water shielding (MWS) tank is a substantial structure for the equipment of the RP. RPV, two pressurizers and the cooler of the purification and heat removal system are enclosed inside the dry caissons of the MWS tank. The passive reactor caisson water flooding system is designed to protect 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%N_{nom} 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 Layout 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

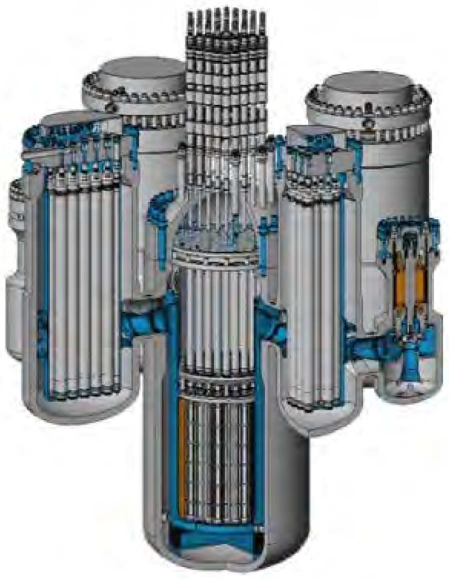
7. Design and Licensing Status

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)

VBER-300 (JSC "Afrikantov OKBM", Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", Rosatom, 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
NSSS Operating Pressure (primary/secondary), MPa	16.3
Core Inlet/Outlet Coolant Temperature (°C)	292 / 328
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	85
Fuel enrichment (%)	4.95
Core Discharge Burnup (GWd/ton)	50
Refuelling Cycle (months)	72
Reactivity control mechanism	Control rod driving mechanism and soluble boron
Approach to safety systems	Hybrid (active and passive) system
Design life (years)	60
RPV height/diameter (m)	9.3 / 3.9
Seismic Design (SSE)	0.25g
Distinguishing features	Power source for transportable Floating NPPs, 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 300 000. 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. Main 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 reactor unit incorporates the reactor and four steam generators – MCPs two-vessel units. The compact 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. 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

Sixty one control rods in combination with fuel elements mixed with burnable poison materials provide safe and reliable reactivity control during both normal and transient operations. 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 impellers.

(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 VBER-300 emergency shutdown system consists of the CRDMs, 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

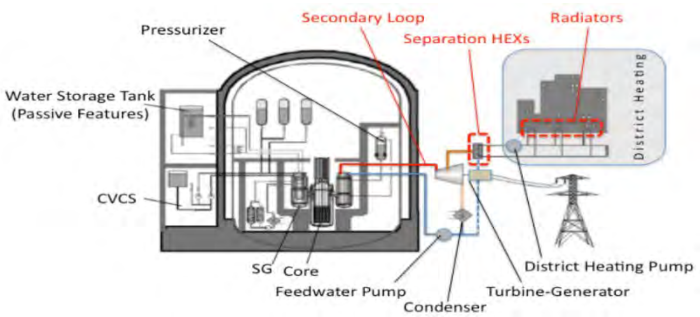
The VBER-300 safety concept is based on the defence-in-depth principles. With the modular configuration, it has increased resistance to impact loads in case of earthquake and aircraft crash. Results of strength analysis under seismic loads of up to 8 points according to the MSK-64 scale carried out for the VBER-300 RPs confirmed that the 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 ton aircraft crash on the 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 core are postulated. In the case 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 Layout 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 m in radius and that has an elliptical bottom. The height of the leak-tight enclosure is 47 m. The steel shell is designed for parameters of the maximum DBA with the excess pressure of 0.4 MPa and the temperature of 150ºC. 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.

(c) 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. It 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.

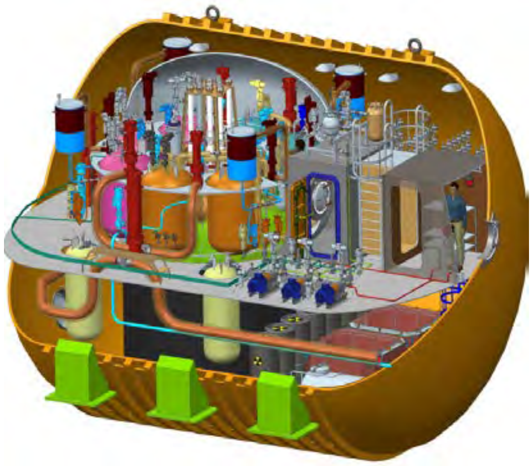
8. Fuel Cycle Approach

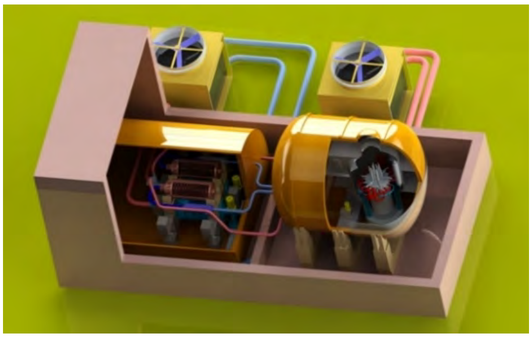
The VBER-300 design concept allows a flexible fuel cycle for the reactor core with standard VVER FAs. The fuel cycles are 3x2 years and 4x1.5 years. The number of FAs in the refuelling batch is either 15 or 30; maximal fuel burnup does not exceed 60 GWd/ton U for the cycle with 30 fresh FAs in the reloading batch and maximum initial uranium enrichment.

9. 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
2007–2008	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

SHELF (NIKIET, Russian Federation)





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 circulations
NSSS Operating Pressure (primary/secondary), MPa	14.7 / 4.7
Core Inlet/Outlet Coolant Temperature (°C)	270 / 310
Fuel type/assembly array	UO_2 pellet / hexagonal
Number of fuel assemblies in the core	163
Fuel enrichment (%)	19.7
Core Discharge Burnup (GWd/ton)	up to 160
Refuelling Cycle (months)	72 (96 for SHELF-M)
Reactivity control mechanism	Control rod driving mechanism
Approach to safety systems	Combined active and passive
Design life (years)	60
Plant footprint (m^2)	8000
RPV height/diameter (m)	Power capsule: 14 / 8 (12 / 8 RF only) RPV: 3 / 1.2
RPV weight (metric ton)	340
Seismic Design (SSE)	SSE 8 (MSK-64)
Fuel Cycle Requirements / Approach	6 years (8 years for the modernized SHELF-M)
Distinguishing features	Power source for users in remote and hard-to-reach locations as both floating and submerged nuclear power plants
Design status	Detailed design underway

1. Introduction

A power unit with the SHELF reactor is designed as a local power source for users in remote and hard-to-reach locations. SHELF is an integral-PWR based 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. The engineering design of SHELF is similar to the marine propulsion nuclear power plants. The power unit is delivered as a single module with all of its components accommodated inside a high-strength containment vessel. 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.

With the capability for long-term unattended automated operation, 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 up to 8000 hours. Within the period of automated operation, the scheduled maintenance is conducted once a year for a duration of 15 days.

2. Target Application

SHELF is designed for a source of power in remote and hard-to-reach locations with decentralized power supplies including the sites on Arctic shore regions. The power of the single-unit plant is 6.6 MW(e) from 28.4 MW(t). Depending on the consumer requirements, SHELF plant can provide 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. For land-based use, 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 for submerged deployment is provided by the sea water.

3. Main 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 reactor components are assembled inside a cylindrical power capsule with 8 m in inner diameter and 14 m long. The reactor equipment is installed in the rear portion of the capsule and the TGP equipment is installed in its front. Besides the power capsule, there is a compact module that houses the auxiliary systems including an air conditioning system to maintain the ambient temperature in the module less than 50^{o}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.

(b) Nuclear Steam Supply System

The reactor uses a traditional two-circuit heat removal system. The reactor's steam removal system is designed to transport the steam generated in the reactor to two turbines. In normal operation, steam is supplied from the reactor to each of the turbines via steam pipelines. Shut-off valves are provided on steam lines both in the sealed volume and outside it. In addition, the reactor plant provides an emergency cooling system. The emergency cooling system is passive and does not require a command to activate.

(c) 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 fuel composition are governed by the maximum fuel load needed for the reactor core life with less than 20% enrichment; fuel composition consists of uranium dioxide in a silumin matrix in the form of cylindrical fuel elements. The ammonia addition is used to generate hydrogen in water to prevent corrosive oxidative radiolysis products generation. The reactor operating time with one fuel load is 5.6 years, with no scheduled maintenance outages for maintenance, and the reactor core life is 40 000 h. This factor is favourable for core self-regulation and safety improvement. Reactor refuelling is performed on the specialized enterprise basis.

(d) Reactivity Control

The core contains two independent reactor shutdown systems. Functionally, the rods are divided into emergency protection rods and control rods. The material of the absorber is boron carbide and titanium diboride. Control rods are grouped in clusters to reduce the number of actuators. Additionally, an emergency system for filling the core with boron carbide solution is provided.

(e) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) accommodates the core and the reactor internals, including heat exchanger (HX) and emergency HX, each consists of 4 sections. The RPV has an elliptical bottom, cylindrical shells, and two reactor cover (central and peripheral). The RPV outer diameter is 1300 mm and height 3000 mm.

(f) 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.

(g) Power Conversion System

The SHELF two-circuit integral-PWR consists of a reactor and associated systems required for its normal operation, emergency cooling, emergency protection and maintenance in a safe condition. The primary circuit system removes heat from the reactor core and transfer it to the secondary circuit fluid in the SG. The secondary circuit system generates superheated steam from feedwater and transfers heat to the turbine generator plant. The secondary circuit system comprises a steam generator installed inside the reactor vessel and pipelines with valves (outside the vessel).

(h) 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.

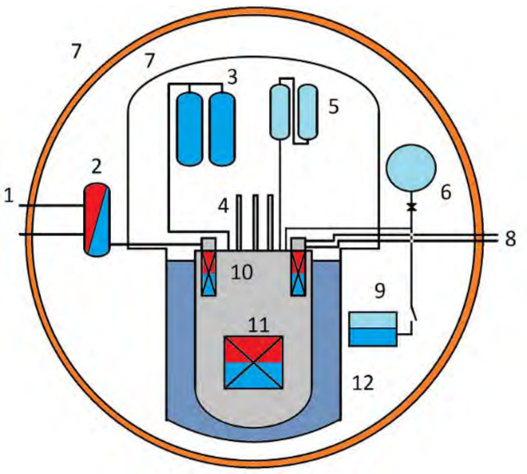
(i) Pressurizer

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.

4. Safety Features

SHELF achieves high level of reactor safety 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:
i. Structure of CPS drive actuators;
ii. Decay heat removal system (DHRS); Emergency core cooling system (ECCS);
4. Safety system reliability:
High reliability level of safety systems is achieved through implementing the following principles:
i. 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.
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



(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).

(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) 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.

(d) Containment System

Reactor is located inside several steel containments. The containment serves to localize accidents and withstands a total pressure in the reactor coolant. They form an additional barrier to leakage of radioactive materials into the environment while limiting the coolant loss during a reactor vessel break.

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 40 000 effective hours.

6. Instrumentation and Control Systems

The automated process control system (APCS) of a nuclear plant with the SHELF reactor is to control the major and auxiliary electricity generation processes in all modes of the unit operation:

1. Normal operation comprises of phased automated initiation, operation at steady power levels in a range of 20 to 100% N_{nom} with forced primary coolant circulation, operation at steady power levels in a range of 20 to 40 % N_{nom} 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.
2. 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.
3. Emergency: Emergency deactivation in the event of reactor facility parameters deviating beyond the safe operation limits or in the event of equipment failures leading to the safe operation limits being violated.

7. Plant Layout 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.

8. Design and Licensing Status

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

9. Fuel Cycle Approach

The duration of the campaign reactor core is 6 years (8 years for the modernized version of SHELF-M).

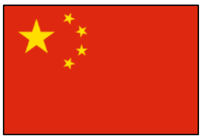
10. Waste Management and Disposal Plan

Fuel handling is based on the traditional scheme implemented for the marine-based prototype reactor. Fuel processing and disposal will take place at a specialized enterprise.

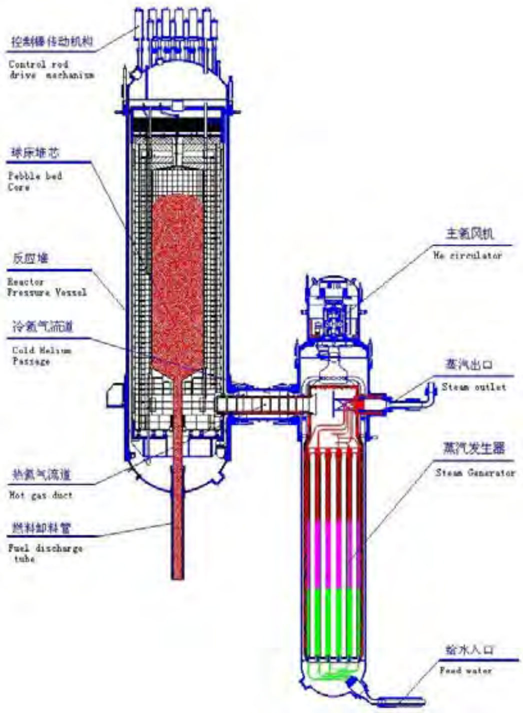
11. Development Milestones

2012-2016	Preliminary studies and technological innovation (using previously developed patents).
2017-2019	Pre-conceptual design and technology validation.
2019-2021	Design phase.
2022-2024	Detailed Design
2025 - 2028	Projected deployment (start of construction) time.
2030	Operation testing

HIGH TEMPERATURE
GAS COOLED
SMALL MODULAR REACTORS



HTR-PM (Tsinghua University, China)



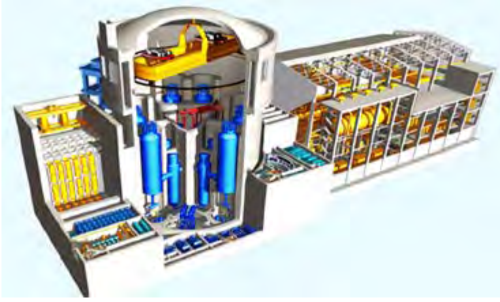
MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	INET，Tsinghua University, People's Republic of China
Reactor type	Modular pebble bed high temperature gas-cooled reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	2x250 / 210
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	7 / 13.25
Core Inlet/Outlet Coolant Temperature (ºC)	250 / 750
Fuel type/assembly array	Spherical elements with coated particle fuel
Number of fuel assemblies in the core	420 000 (in each reactor module)
Fuel enrichment (%)	8.5
Core Discharge Burnup (GWd/ton)	90
Refuelling Cycle (months)	On-line refuelling
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Combined active and passive
Design life (years)	40
Plant footprint (m^2)	--
RPV height/diameter (m)	25 / 5.7 (inner)
RPV weight (metric ton)	800
Seismic Design (SSE)	0.2g
Fuel cycle requirements / Approach	LEU, open cycle, spent fuel intermediate storage at the plant
Distinguishing features	Inherent safety, no need for offsite emergency measures
Design status	Finalizing construction, startup commissioning test of primary circuit in 2020

1. Introduction

In 1992, the China Central Government approved the construction of the 10 MW(t) 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 next 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 9 December 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 already installed. The power plant is scheduled to start power generation in 2021.

2. Target Application



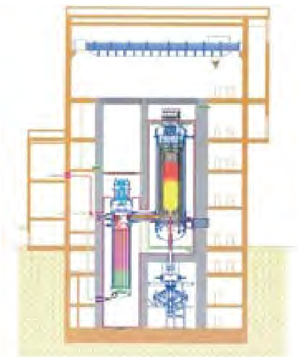
The HTR-PM is a commercial demonstration unit for electricity production. The twin reactor modules driving a single turbine configuration was specifically selected to demonstrate its feasibility. Following the HTR-PM demonstration plant, commercial deployment of HTR-PM based on batch construction is planned. Units with multiple standardized reactor modules coupling to one single steam turbine, such as 200, 600 or 1000MW(e) are envisaged.

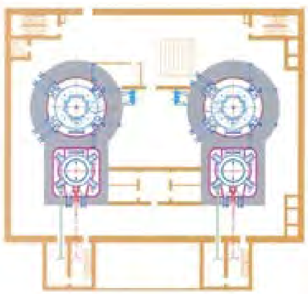
A standard design has been finished for the 600 MW(e) multi-module HTR-PM600 nuclear power plant, which consists of six reactor modules coupling to one steam turbine. Each reactor module has the same design as the HTR-PM demonstration plant, with independent safety systems and shared non-safety auxiliary systems. The footprint of a multi-module HTR-PM600 plant is not substantially different from that of a PWR plant generating the same power. The figure shows the 600 MW(e) HTR-PM600 nuclear power plant layout. Future sites have been identified for possible deployment.

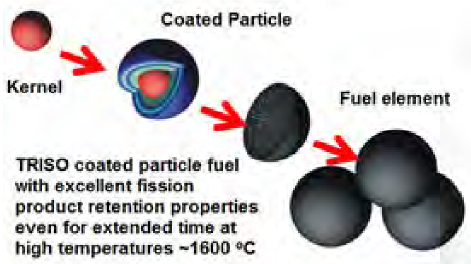
3. Main Design Features

(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) that contains 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 parameters is 13.25 MPa/567°C at the steam turbine entrance.







(b) Reactor Core and Power Conversion Unit

The primary helium coolant works at 7.0 MPa with the rated mass flow rate of 96 kg/s. Helium coolant enters the reactor in the bottom area inside the RPV 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

Illustrated above, fuel elements are spherical ones. Every fuel element contains 7 grams of heavy metal. The enrichment for the equilibrium core is 8.5% of U_{235}. 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, 5 cm in diameter. Surrounding the fuel containing graphite matrix is a 5 mm thick graphite layer.

(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 performed using 24 control rod assemblies, and 6 SAS shutdown systems serve as a reserve shutdown system. The control rods are used as a regulating group during normal plant operation and for emergency shutdown. Furthermore, turning off the helium circulator is also efficient for reactor trip. Drop of all control rods can achieve long term shutdown.

The SAS system is used to reduce the shutdown temperature for the purpose of in-service inspection and maintenance. 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 is designed with the following safety features: (1) radioactive inventory in the primary helium coolant is very small during normal operation conditions, and even if released there is no need to take any emergency measures; (2) for any reactivity accident or loss of coolant accident, the rise of the fuel elements' temperature will not cause a significant additional release of radioactive substances; (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 ensure safe operation and limit 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 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 transfer tubes, a dumping system is designed to minimize the amount of water ingress into the primary circuit.

(b) Reactivity control

The on-line refuelling leads to a small excess reactivity, the overall temperature coefficient of reactivity is negative, and two independent shutdown systems are available.

(c) Reactor Cooling Philosophy

Normally the reactor is cooled by steam generating system. Under accident conditions, the main helium blower 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. And the fuel temperature increase in this phase will compensate accident reactivity and shutdown the reactor automatically via negative temperature feedback. 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 by transferring it through the concrete structure of reactor cavity while the temperatures of fuel elements are under design limit.

(d) 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 cavity inside reactor building and some auxiliary systems such as sub-atmosphere ventilation, burst disc and filters.

(e) Chemical control

Water and steam ingress is limited by plant design (pipe diameters, SG lower than core and water / steam dumping system) while massive air ingress are practically eliminated (small pipes, connection vessel; no chimney effect). After water ingress accident, the way to remove humidity from primary circuit are provided.

5. Plant Safety and Operational Performances

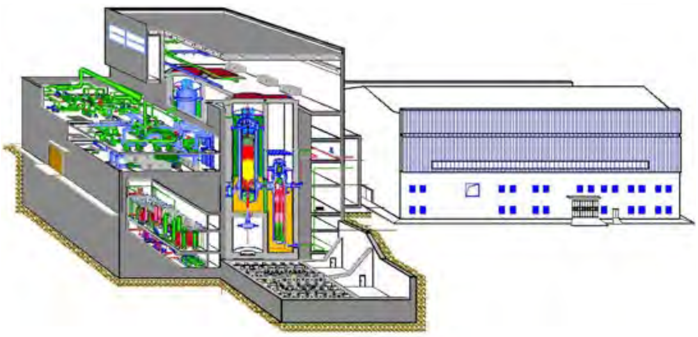
The HTR-PM demonstration power plant is under construction. Due to online fuel loading mode of HTR-PM, better availability factor can be expected compared with other power plants operating in a mode of periodic fuel loading.

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 Layout 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. Fuel Cycle Approach

The air-cooled spent fuel canisters are placed in the spent fuel storage building with concrete shields. The canister can be placed in a standard LWR transport cask and be transported if necessary. HTR-PM currently adopts open fuel cycle. After the intermediate storage of spent fuel elements, the final storage in geological deposits can be carried out in the open cycle. In the closed cycle, the spent fuel elements would be dismantled, and the nuclear fuel can be reprocessed in normal reprocessing facilities (when the amount of spent fuel reaches certain level and reprocessing technology is economically available).

10. Waste Management and Disposal Plan

The technologies of cleaning the liquid waste and of the off-gases are similar to those used in normal PWR plants, although the amount of liquid waste from HTR-PM is much smaller. Waste with low or medium level activity resulting from the operation is conditioned following different process technologies, which have been established with high efficiency in nuclear industry. Different auxiliary material and solid residue will be put into casks for intermediate storage. These waste can be underwent a final storage after conditioned.

11. 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
Q4/2020	Startup commissioning test of primary circuit
2021	Commercial / Demonstration operation

STARCORE (StarCore Nuclear, Canada, United Kingdom and United States)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	StarCore Nuclear, Canada
Reactor type	HTGR
Coolant/moderator	Helium/Graphite
Thermal/electrical capacity, MW(t)/MW(e)	Block One: 35 / 14 Block Two: 50 / 20 Block Three: 150 / 60 One to six modules/plant
Primary circulation	Forced
NSSS Operating Pressure (primary/secondary), MPa	7.4 / 6.7
Core Inlet/Outlet Coolant Temperature (°C)	280 / 750
Fuel type/assembly array	TRISO Prismatic
Number of fuel assemblies in the core	Block 1: 90; Block 2: 126; and Block 3: 324
Fuel enrichment (%)	15
Core Discharge Burnup (GWd/ton)	60 GWd/t / 6 %
Refuelling Cycle (months)	> 60
Reactivity control mechanism	2 x 6 Rotary/Automatic deployment when gas flow lost
Approach to safety systems	Passive; Containment 1: TRISO; Containment 2: RPV/ETS-1 Helium System; Containment 3: Silo
Design life (years)	40 – 60
Plant footprint (km^2)	1 (up to 1 Hectare)
RPV height/diameter (m)	Block One and Two: 7.4 / 2.6 Block Three: 9 / 3.2
RPV weight (metric ton)	Block One: 75; Block Two: 90; Block Three: 150
Seismic Design (SSE)	0.25g
Fuel cycle requirements / Approach	LEU/Temporary storage in Silo at plant
Distinguishing features	RPV 30 Metres below grade in hardened silos
Design status	Pre-conceptual/conceptual design

1. Introduction

Founded in 2008, StarCore had the first major Technical Review at Argonne National Laboratory in 2013, and the last major Due Diligence (technical and business) review in 2019. StarCore is teamed with TREDIC Corporation (UK), a global infrastructure developer to export plants from Canada to all compliant Nations around the Globe. StarCore provides risk free, inherently safe, Generation IV Small Modular Reactor (SMR) power to off-grid and edge-of-grid locations. Our operations can provide clean electricity for industrial and consumer use, high temperature thermal energy for down-stream minerals processing, desalinated or purified water for irrigation or for those people without clean water sources, and wide-band internet for medical and educational use.

StarCore's business model is predicated on a 'Build, Own, Operate and Decommission' (BOOD) basis, generating revenue and profit from Power Purchase Agreements (PPAs) and off-take arrangements with both national and local governments, and private enterprise. The StarCore business plan therefore brings with it a very low financial burden and StarCore will be responsible for obtaining all the necessary licenses and certification.

2. Target Applications

Serving remote communities to provide energy access and combat energy poverty is an important part of StarCore's vision. In the developing countries there are more than two billion people who are either without electricity or who get power from diesel generators at very high cost. Many countries lack supplies of clean water and desalination-only plants are attractive to them. We are aware that we cannot just bring energy to a remote community – we must also bring some type of environmentally suitable industry, so the people have a future. We include 100 km HVDC transmission lines in our cost estimates, and so we can connect plants to residential and manufacturing sites. Remote mines are also an ideal market for StarCore, since they tend to be in remote locations and energy costs are high.

3. Main Design Features

(a) Design Philosophy

StarCore is designed to operate in the harshest environment in remote locations anywhere in the world. To this end the reactors are contained in steel-walled concrete silo structures 30 metres below grade. Each reactor silo has two supporting silos with their access hatches in the refuelling room, also below grade. This room is filled with helium at low pressure during refuelling operations, which are carried out by an automatic system.
StarCore plants have between one and six reactors per plant, giving us a range of outputs between 14 MW(e) and 360 MW(e). The plants are load-following, and can also output High Temperature Gas or Steam, with the outputs dynamically changed as required.
The plants meet all planned Remote Siting Requirements, which include that they must be: Inherently Safe; Passively Secure; Load Following; Fully Automated; Have a Remote Shutdown (Intervention) Capability; and after suitable qualification (subject to local regulations) be operated with a zero-radius exclusion zone.
The plants also have 69 KV 100 km HVDC transmission lines included in the cost of the plant. The StarCore Build-Own-Operate-Decommission Business Plan includes all capital costs, licensing, operating and decommissioning logistics, thus eliminating the cost of entry to all countries and communities. The HVDC is inverted to local frequency standards.

(b) Reactor Core

The prismatic core is made up of hexagonal graphite blocks that are 360 mm across the flats and 793 mm long. Cylindrical fuel compacts (26 mm diameter and 39 mm long) are inserted into holes drilled in the graphite blocks, and burnable (neutron) poison elements will also be inserted as needed. The helium flow will be through vertical holes drilled in the blocks. The number of blocks used depend on the output of the core.
The helium coolant enters at the bottom of the core, flows up the outside including through the reactivity control mechanisms, and then down through the core prismatic blocks. There are automatic bi-stable valves at the inlet and outlet, which seal off the core in the event of pressure loss.

(c) Reactivity Control

The core has vertical rotary reactivity control rods with a neutron reflector on a semi-circular rotating half-cylinder. There are two sets of 6 control rods each with a different deployment mechanism that deploys the controls if helium flow or pressure is lost. The reactivity control mechanisms use the helium pressure differential across the core to stay stowed; in the event of pressure or flow loss they automatically deploy.

(d) Reactor Pressure Vessel and Internals

The primary containment boundary is the TRISO fuel microsphere; the second is the Energy Transport System (ETS-1) with the nuclear system containing the RPV, piping, pumps and first stage intermediate heat exchanger. IHX-1. There are helium scrubbers in ETS to remove trace elements of radioactive dust and tritium.

(e) Reactor Coolant System

The first stage coolant Energy Transport System (ETS-1) is helium at 7.3 MPa; this then transfers energy to ETS-2 through IHX-1 which contains nitrogen at 6.7 MPa. Nitrogen is used in this system at this pressure to minimize delta-P across IHX-1 and allow compact exchanger designs to be sued; in addition, the pressure gradient ensures that any gas migration will be in the direction of ETS-2.
The Nitrogen in ETS-2 then passes to an aero-derivative gas turbine that has an annular heat exchanger in place of the usual burner cans, and the turbine exhaust gas at 300 ºC can be used for district heating or cooling. The energy in ETS-2 can also be sent to a heat exchanger that provides high-temperature gas or steam to external plants.

(f) Safety Features

The TRISO fuel exhibits a very strong negative temperature coefficient. As the fuel temperature increases the neutron energy also increases; this effect reduces the neutron cross sections and lowers the number of fissions and thus the power level. The result of removing all reactivity controls and shutting off the primary and secondary cooling systems will result in the core becoming stable with an output of 600 kW(t), which will be dissipated to the outside of the silo through vestigial fins into iso-thermal layer of the surroundings. The prismatic core is attractive in this regard since the thermal pathways are better than those in pebble bed cores. StarCore also has automatically deployed reactivity controls and inlet and outlet shutoff valves that are deployed if helium pressure or flow is lost. In a worst-case accident, where all control mechanisms fail, and the helium is exhausted to atmosphere the reactor will not suffer any catastrophic failures or radiation release. A core meltdown type of event simply cannot happen; in the worst case the core will remain several hundred degrees C below TRISO microsphere failure temperatures.

4. Plant Safety and Operational Performance

Most nuclear plant control systems today rely on operators to determine the correct course of action in complex circumstances; nearly all reactor accidents and failures have been the result of incorrect operator techniques. This is not practical for remote locations, and the StarCore Reactor Plant will be fully automatized using on-site hardware and software.

StarCore's remote control technology will provide full-time monitoring and the ability to shut down the plant from StarCore Central or regional administrative centres by satellite links (2xGEO and 1xLEO) and reduce on-site personal to only maintenance workers. This design for fully automated operation with 'remote monitoring and intervention (shutdown)' is the StarCore Automated Reactor System (STARS) and has been the subject of an independent review by the former Atomic Energy Canada Limited (AECL) and the Canadian National Laboratories (CNL).

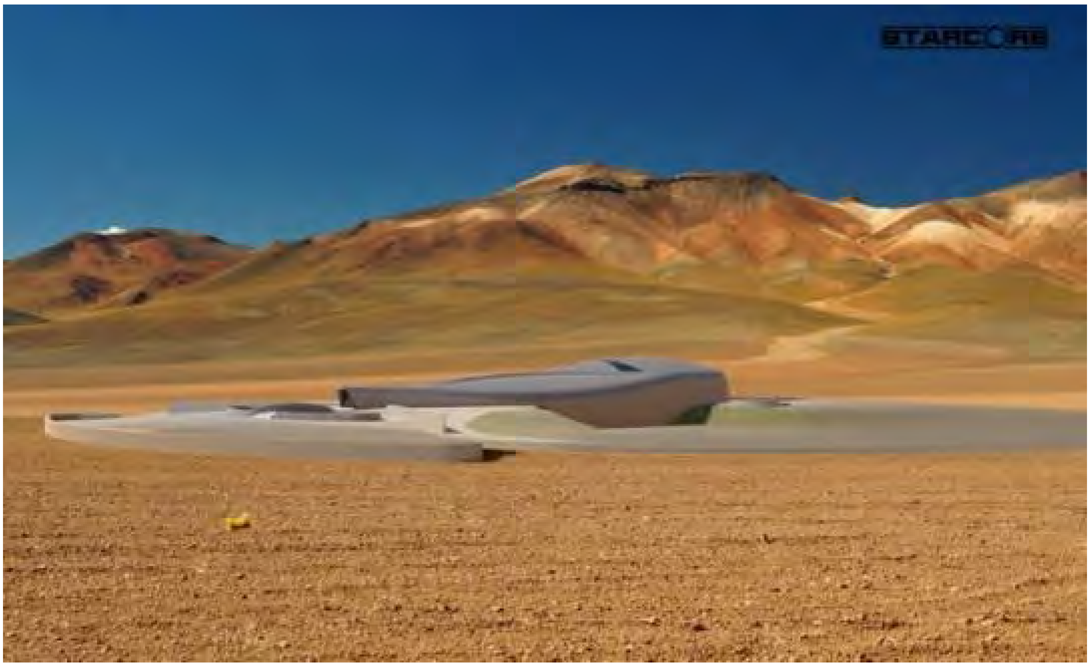
5. Instrumentation and Control Systems

StarCore owns the Intellectual Property to a modern fully automated control system design (the STARS HyperVector Control System) previously used in many safety-critical aerospace systems. There are many benefits that this control system technology brings, including automatic failure prediction for every system or component in an arbitrarily complex application; alarms that uniquely identify any specific failures that have occurred or are predicted; controls that prevent wrong commands or actions ever being taken, and automatic responses to arbitrarily complex failures.

The system is named after the manifold in n-dimensional state space that define the operational limits of the systems and components; these are represented by n-vectors, or HyperVectors, defined as complex data in the imaginary plane. The states are defined for every component in the plant, recognize system operations, and predict - in real time - any failures that may occur by calculating the state vector and time-to-state-operational-boundary for all components.

6. Plant Layout Arrangement

The plant has from two to six hardened silos at the base of the turbine hall; up to three turbine halls can be accommodated in each plant. The main body of the plant is constructed of high-performance modules that are bonded into a single monocoque structure and is designed to be capable of resisting standard man-portable weapons such as RPGs. The plant is designed to meet a Beyond Design Basis Accident (such as an aircraft or bomb) that destroys all above-ground facilities without causing any nuclear contamination.



7. Design and Licensing Status

StarCore has completed the initial design needed to develop Supplier Approval Procedures which resulted in an Approved Supplier List and a Hardware Readiness Assessment. The StarCore Team includes direct-hired StarCore personnel responsible for the overall management of the entire program, including all design authority for commissioning, maintaining, operating, refuelling and decommissioning of the reactor plants.

8. Fuel Cycle Approach

The anticipated core lifetime is more than 5 years. At the end of every fifth year of operation, the graphite fuel prismatic blocks and spent fuel will be removed from the reactor pressure vessel and will be stored for 12 months in an on-site underground fuel storage silo before being transported to a permanent repository site. The spent fuel will be transported from the plant site to the repository site using a certified, existing fuel transfer cask. Each transfer cask will hold six graphite fuel blocks, assembled in a Fuel Cartridge and which will be replaced as one unit. StarCore will use a once-through LEU cycle initially, with plans to pursue R&D to move to a fuel recycling capability depending on political agreements. We also plan to investigate TRU use in MOX TRISO fuel to start recycling LWR waste in due course.

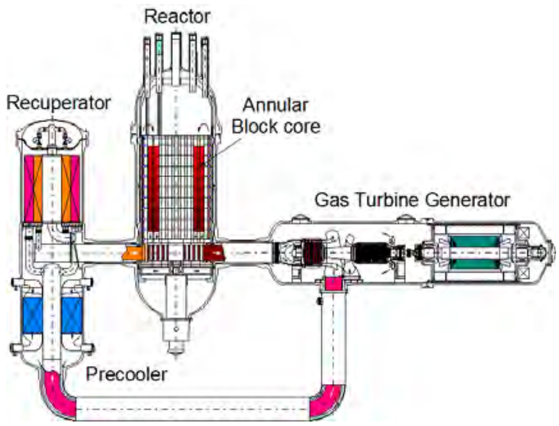
9. Waste Management and Disposal Plan

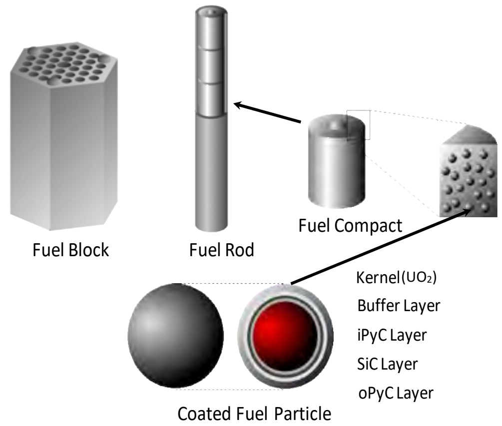
The plant will be decommissioned by StarCore at the end of its life, and all decommissioning cost is already built into the StarCore Financial Plan. This work will include: removal of the reactor fuel and shipment to a spent fuel storage facility; the spent fuel will be managed and disposed of; removal of the reactor vessel and shipment off site and disposed of; disposal of reactor components that cannot be reused; removal of all equipment with radioactive components at the plant site and shipment to a disposal site; demolition of all above ground facilities at the plant site and shipment of the materials off-site to a disposal facility; and entombment of the below ground facilities at the plant site by backfilling them with concrete.

10. Development Milestones

209- 2013	Preliminary studies and initial pre-conceptual design.
2013-2017	Pre-conceptual design phase, technology validation and vendor contracts and qualification
2017-2020	Fund raising and PPA contracts
2021-2016	Projected deployment (start of construction to commissioning)

GTHTR300 (JAEA Consortium, Japan)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JAEA, MHI, Toshiba/IHI, Fuji Electric, KHI, NFI, Japan
Reactor type	Prismatic HTGR
Coolant/moderator	Helium / graphite
Thermal/electrical capacity, MW(t)/MW(e)	<600 / 100~300
Primary circulation	Forced by gas turbine
NSSS Operating Pressure (primary/secondary), MPa	7 / 7
Core Inlet/Outlet Coolant Temperature (^{o}C)	587-633 / 850-950
Fuel type/assembly array	UO_2 TRISO ceramic coated particle
Number of fuel assemblies in core	90
Fuel enrichment (%)	14
Core Discharge Burnup (GWd/ton)	120
Refuelling Cycle (months)	48
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Active and passive
Design life (years)	60
Plant footprint (m^2)	~250x250 (4-reactor plant)
RPV height/diameter (m)	23 / 8
RPV weight (metric ton)	~1000
Seismic Design (SSE)	>0.18g automatic shutdown
Fuel cycle requirements / Approach	Uranium once through (initially)
Distinguishing features	Multiple applications of power generation, cogeneration of hydrogen, process heat, steelmaking, desalination, district heating
Design status	Pre-licensing basic design completed

1. Introduction

The 300 MW(e) Gas Turbine High Temperature Reactor (GTHTR300) is a multi-purpose, inherently-safe and site-flexible small modular reactor (SMR) that Japan Atomic Energy Agency (JAEA) is developing for commercialization in 2030s. As a Generation-IV technology, the GTHTR300 offers important advances when compared to current light water reactors. The reactor coolant temperature is significantly higher in the range of 850-950^{o}C. Such high temperature capability as proven in the JAEA's HTTR test reactor operation enables a wide range of applications. The design employs a direct-cycle helium gas turbine to simplify the plant by eliminating water and steam systems while generating power with enhanced efficiency of 45-50%. 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 in close 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^{o}C) heat rejection from the gas turbine cycle, making inland and remote siting possible without the need of a large source of cooling water.

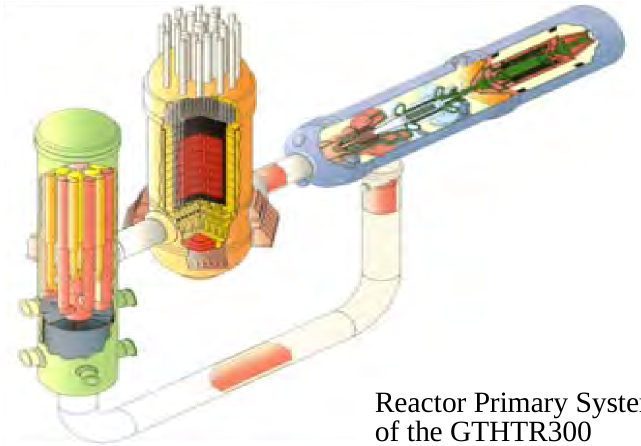
2. Target Applications

Typical applications include electric power generation, thermochemical hydrogen production, desalination cogeneration using waste heat, and steel production. The reactor thermal power may be rated up to 600 MW(t). The maximum hydrogen production per reactor is 120 t/d, 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. Main 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.



Reactor Primary System
of the GTHTR300

(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 950ºC test reactor, the balance of plant key technologies needed for the commercial SMR are separately being developed and tested. The activities include 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 950ºC 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 1600ºC. Approximately 10 000 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. 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 reflector blocks on inner and outer rings of 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:

- Ceramic coated particle fuel maintains containment integrity under a temperature limit of 1600ºC.
- Reactor helium coolant is chemically inert and thus absent of explosive gas generation or phase change.
- Graphite-moderated reactor core provides 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

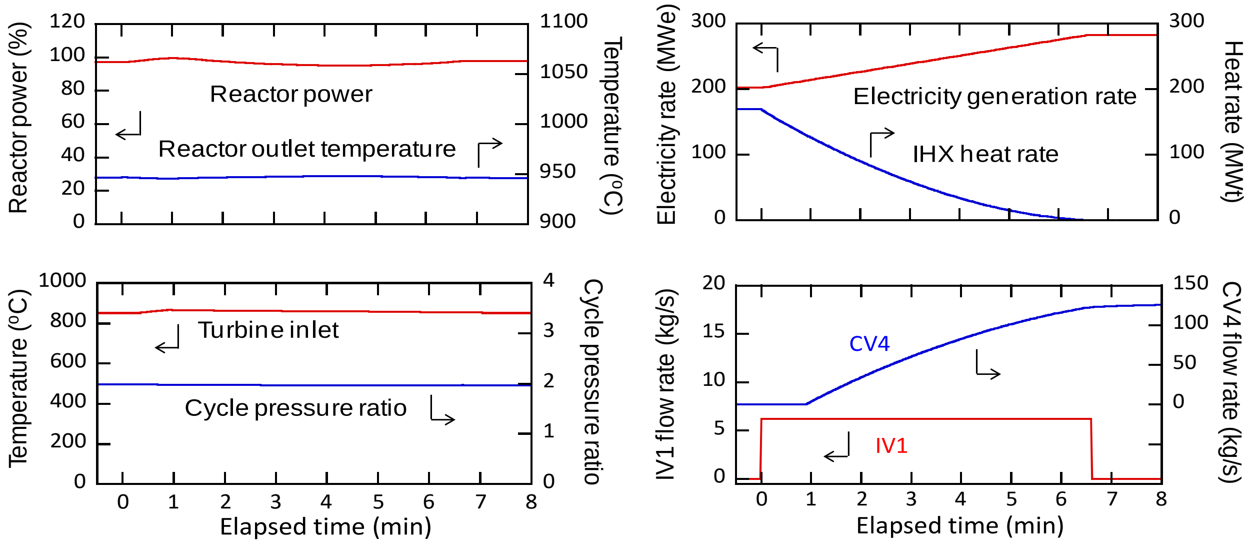
GTHTR300 design removes the core decay and residual heat from the outside surface of the reactor vessel by the natural convection and radiation, and to transfer it to an ultimate heat sink in the operational states and in the accident conditions so that the design limits for fuel, the reactor coolant 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 reactor core.

5. Plant Safety and 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 base cogeneration state, control is reversed by reducing primary coolant inventory and closing bypass valve VC4.



6. Instrumentation and Control Systems

The instrumentation and control system consists of reactor and process instrumentations, control systems, safety protection system, and engineered safety features actuating systems. The six (6) fundamental controls are for turbine bypass, inventory, reactor outlet temperature, turbine inlet temperature, process heat supply rate and IHX differential pressure controls. They 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 Layout 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. Fuel Cycle Approach

The design is applicable to fuel cycle options including UO_2, MOX, and Pu-burning.

10. Waste Management and Disposal Plan

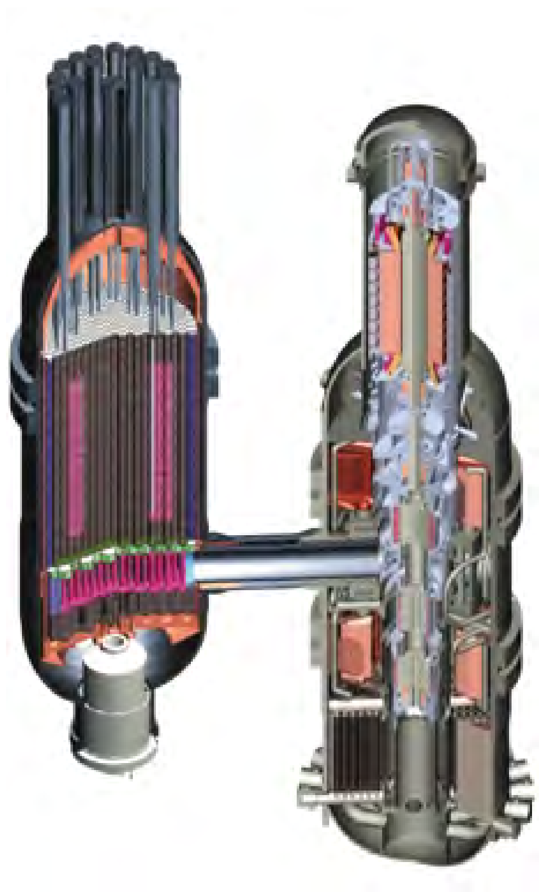
The design is applicable to options of direct disposal or reprocessing for spent fuel.

11. Development Milestones

The design is at pre-licensing stage with various development and demonstration activities in Japan as follows:

2003	Basic design of GTHTR300 completed
2004	Begin development of key technologies
2005	Design for cogeneration plant GTHTR300C
2014	IS process continuous H_2 production test facility construction
2015	Basic design for HTTR-connected gas turbine and H_2 plant (HTTR-GT/H_2) for system demonstration
2020s~	HTTR-GT/H_2 test plant construction and operation (planned)
2030~	Construction of lead commercial plant (planned)

GT-MHR (JSC “Afrikantov OKBM”, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, Russian Federation
Reactor type	Modular Helium Reactor
Coolant/moderator	Helium /graphite
Thermal/electrical capacity, MW(t)/MW(e)	600 / 288
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	7.2 / -
Core Inlet/Outlet Coolant Temperature (°C)	490 / 850
Fuel type/assembly array	Coated particle fuel in compacts, hexagonal prism graphite blocks of 0.36 m
Number of fuel assemblies in the core	~1020
Fuel enrichment (%)	14-18% LEU or WPu
Core Discharge Burnup (GWd/ton)	100-720 (depends on fuel type)
Refuelling Cycle (months)	25
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Hybrid (active and passive)
Design life (years)	60
Plant footprint (m^2)	9110
RPV height/diameter (m)	29 / 8.2
RPV weight (metric ton)	950
Seismic Design (SSE)	8 points (MSK 64)
Fuel cycle requirements / Approach	Standard LEU or WPu / No recycling; high fission product retention
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 a HTGR with a Brayton power conversion cycle to produce electricity at high efficiency. As the reactor unit can produce high coolant outlet temperatures, the modular helium reactor system can also efficiently produce hydrogen, e.g. by high temperature electrolysis or thermochemical water splitting.

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

The modular helium reactor design proved 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 can produce high coolant outlet temperatures, the modular helium reactor system can also efficiently produce hydrogen, e.g. by high temperature electrolysis or thermochemical water splitting.

3. Main Design Features

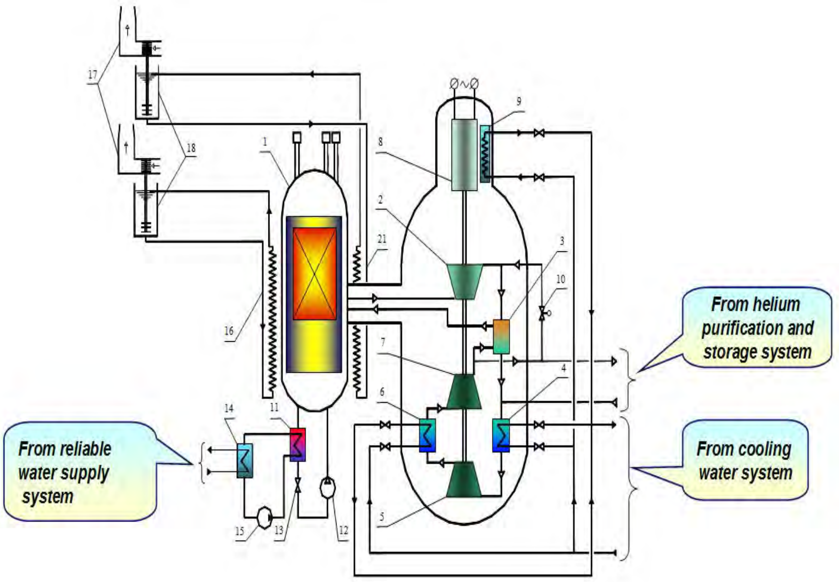
(a) Design Philosophy

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 pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with thousands of compacts inserted into the fuel channels of the hexagonal prismatic graphite blocks or fuel assemblies. The coated particles will contain almost all fission products with temperatures up to 1600ºC. 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) but alternative cycles including the disposition of plutonium were also studied in detail.

(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 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) 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 assemblies stacked 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.

(e) 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, 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, what 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) Use 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 during 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 to 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.

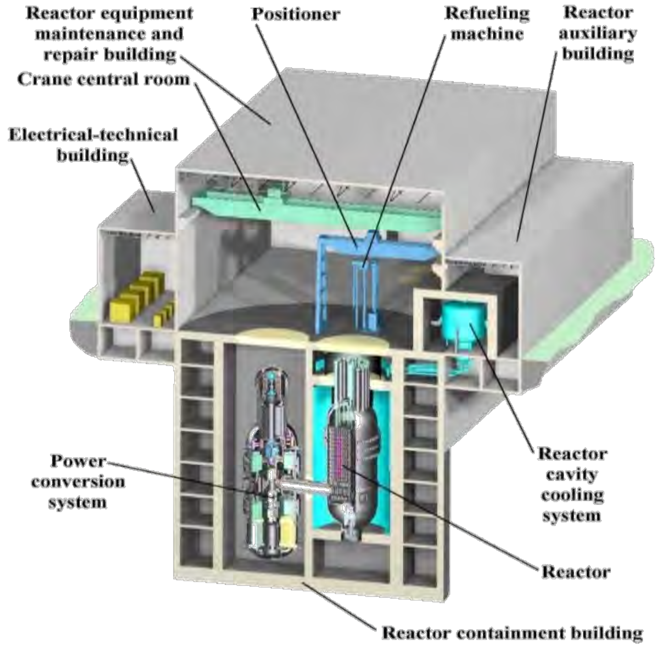
6. Instrumentation and Control Systems

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 Layout Arrangement

The plant layout is shown on the right.



8. Design and Licensing Status

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

9. Fuel Cycle Approach

The GT-MHR fuel cycle approach is a once through mode without reprocessing. 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 at fuel reloading. These measures are also applied at spent fuel management. Spent fuel shows good proliferation resistance characteristics, producing less materials of proliferation concern (total plutonium and ^{239}Pu) per unit of energy produced.

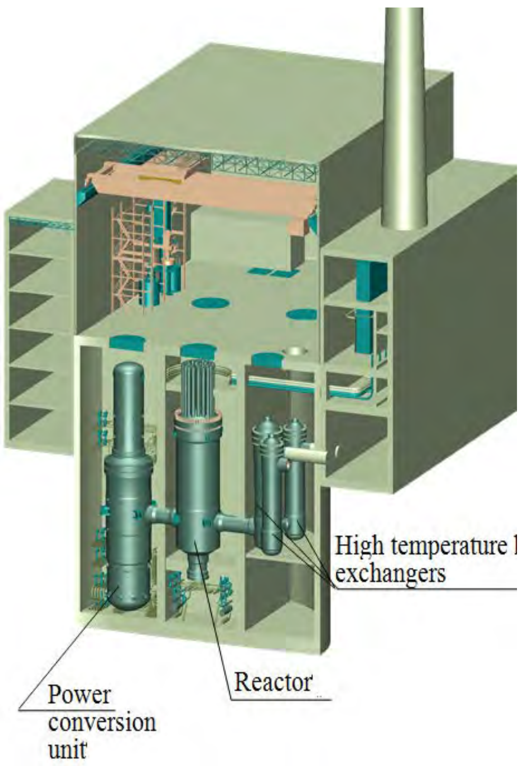
10. Waste Management and Disposal Plan

Facilities for long-term storage of spent nuclear fuel (SNF) and solid/solidified radioactive waste (RW) are included in the complex of a GT-MHR commercial 4-unit NPP. The capacity of the designed SNF storage is determined from the condition of capability to store fuel unloaded from the NPP for 10 years. The estimated total construction volume of the SNF reception and storage compartments is around 150 000 m^3. The capacity of solid/solidified RW storage facility is designed to provide storage of waste generated during the 10-year period of NPP operation. After 10 years of storage at the NPP site, SNF and RW are to be removed for final underground disposal. Radiochemical SNF reprocessing is considered as an option for future only.

11. 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 (JSC "Afrikantov OKBM", Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", 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
NSSS Operating Pressure (primary/secondary), MPa	7.5 / -
Core Inlet/Outlet Coolant Temperature (ºC)	578 / 950
Fuel type/assembly array	Coated particle fuel in compacts, hexagonal prism graphite block of 0.36m
Number of fuel blocks	~1020
Fuel enrichment (%)	< 20 LEU
Core Discharge Burnup (GWd/ton)	125
Fuel cycle (months)	30
Reactivity control mechanism	Control rods
Approach to safety systems	Hybrid (active and passive)
Design life (years)	60
RPV height/diameter (m)	32.8 / 6.9
RPV weight (metric ton)	950
Seismic design	8 points (MSK 64)
Fuel cycle requirements / Approach	Standard LEU / No recycling; high fission product retention
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. Main 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.

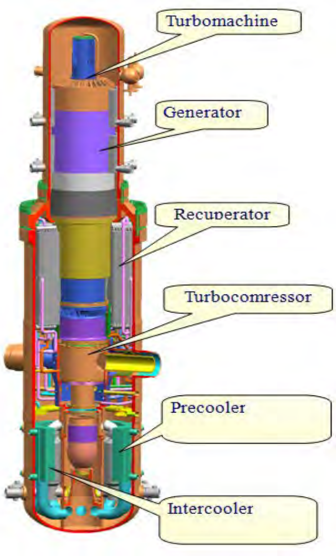
(b) Reactor Core and Fuel Characteristics

Coated particle fuel is used. The fuel kernel (U oxide) is coated by a first porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with thousands of compacts inserted into the fuel channels of the Hexagonal Prism graphite blocks or fuel assemblies.

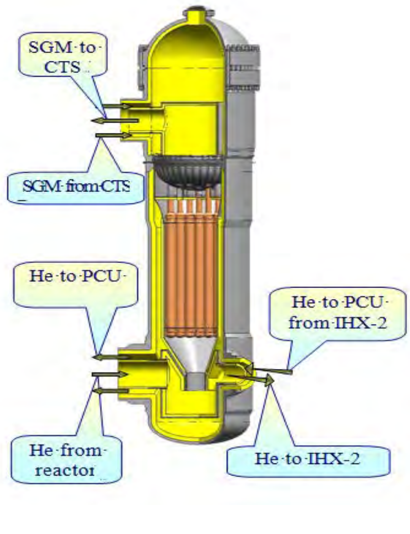
The coated particles will contain almost all fission products with temperatures up to 1600°C. About 1 billion fuel particles of the same type were manufactured and tested in Russia. The standard fuel cycle is to utilize low enriched uranium (LEU) in a once through mode. The MHR-T show good proliferation resistance characteristics. It produces less total plutonium and ^{239}Pu (materials of proliferation concern) per unit of energy produced. The fuel form presents formidable challenges to diversion of materials for weapons production, as either fresh or as spent fuel.

(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. The gas turbine cycle of power conversion with the helium turbomachine, heat exchanger and intercooler provides a thermal efficiency of 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) 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.

(f) 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 are as follows:

- Using helium as the coolant. During operation, helium is not affected by phase transformations. It does not dissociate and has good heat transfer properties. Helium is chemically inert. It 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 maintain core configuration integrity;
- 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 In case of LOCA with failures of all active circulation systems and power sources, operation limits of the fuel are not exceeded.

(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

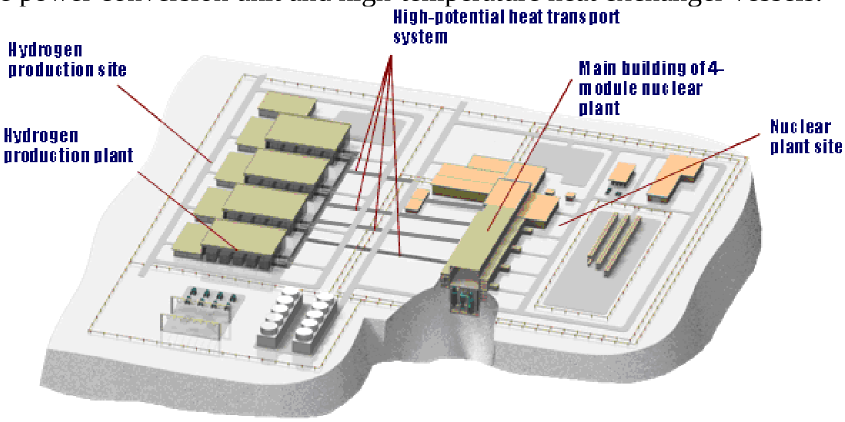
All safety systems are designed with two channels. Fulfilment of the regulatory requirements on safety, proven by a compliance with both deterministic and probabilistic criteria, is secured by an exclusion of the active elements in a channel or by applying the required redundancy of such active elements 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.

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. Currently, safety issues is the major area of R&D with the emphasis on mutual influence of nuclear and hydrogen production components of the facility.

9. Fuel Cycle Approach

The MHR-T fuel cycle approach is a once through mode without reprocessing. Appropriately shielded containers are provided to protect the personnel against radiation impacts during dismantling of the reactor unit components at fuel reloading. These measures are also applied at spent fuel management.

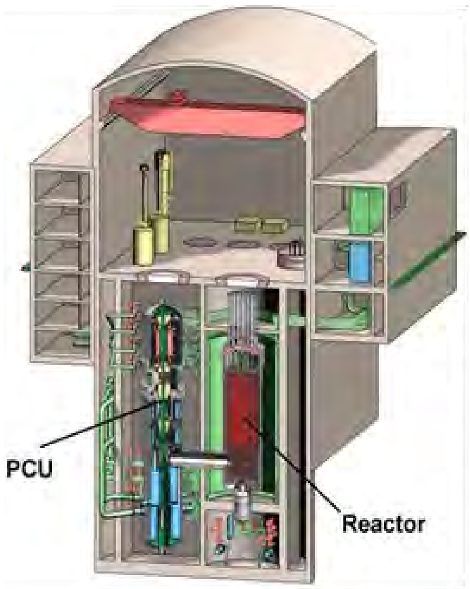
10. Waste Management and Disposal Plan

Facilities for long-term storage of spent nuclear fuel (SNF) and solid/solidified radioactive waste (RW) are included in the complex of a MHR-T commercial 4-unit NPP. The capacity of the designed SNF storage is determined from the condition of capability to store fuel unloaded from the NPP for 10 years. The estimated total volume of the SNF reception and storage compartments is around 150 000 m^3. The capacity of solid/solidified RW storage facility is designed to provide storage of waste generated during the 10-year period of NPP operation. After 10 years of storage at the NPP site, SNF and RW are to be removed for final underground disposal.

11. 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
2020	Development of basic design solutions for chemical-technological part of the facility

MHR-100 (JSC “Afrikantov OKBM”, Russian Federation)



MHR-100 GT layout

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, 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
NSSS Operating Pressure (primary/secondary), MPa	4-5 / depends on configuration
Core Inlet/Outlet Coolant Temperature (^oC)	490-553 / 795-950 (depends on configuration)
Fuel type/assembly array	Hexagonal prism 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
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Hybrid (active and passive)
Design life (years)	60
Plant footprint (m^2)	Depends on configuration
RPV height/diameter (m)	Similar to 1000 MW(e) LWR
RPV weight (metric ton)	Similar to 1000 MW(e) LWR
Seismic Design (SSE)	8 points (MSK 64)
Fuel cycle requirements / Approach	Once through U; Pu and Th cycle also possible
Distinguishing features	A multipurpose reactor for cogenerations of electricity, heat and hydrogen; high-temperature heat supply to oil refinery plant
Design status	Conceptual design

1. Introduction

The designs are based on the global experience in the development of experimental HTGR plants. Russia has more than 40 years of 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. They consist mainly of cogeneration plants producing about 40% of electric power and 85% of heat generation. Analysis shows that SMRs 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 study of the power market development and demands, pre-conceptual work is performed for commercial MHR100 with modular helium reactor and several power conversion layouts for various power-industrial applications. The following options of MHR100 for industrial applications were studied:

- Electricity and district heat productions by core thermal power conversion to electric one in direct gas-turbine Brayton cycle – MHR-100 GT;
- Electricity and hydrogen generations by high-temperature steam electrolysis – 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		
Parameters	**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

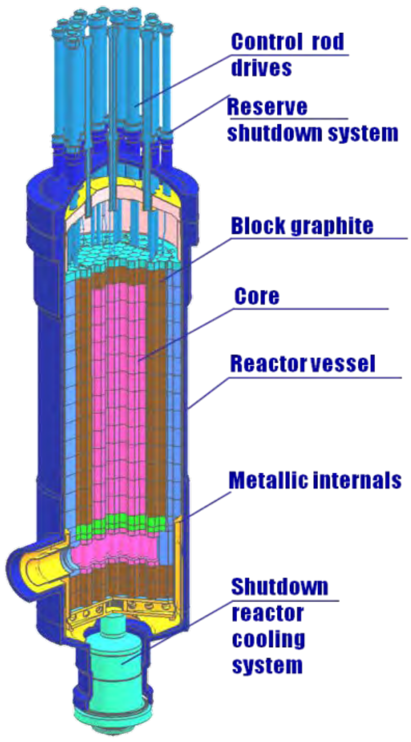
3. Specific Design Features

(a) Design Philosophy

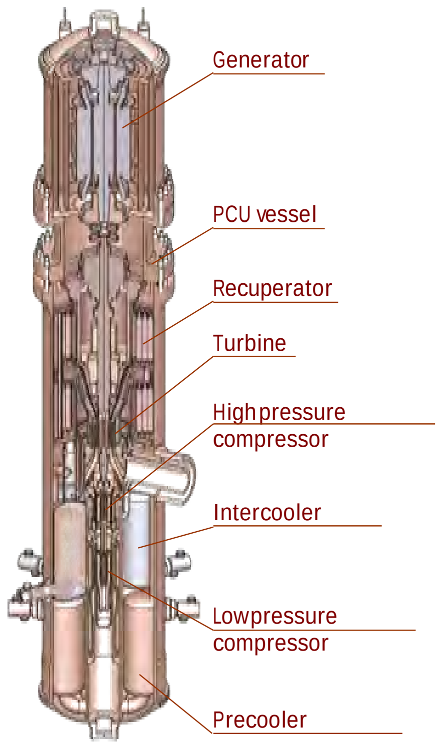
The reactor power and its design are universal for all the different power and process heat options with only the coolant parameters that are different. The reactor unit power level (215 MW(t)) was selected according to: (i) the regional power industry and district heat supply needs; (ii) the manufacture 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 pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with hundreds of compacts inserted into fuel channels of each hexagonal prism graphite block (0.2 m x 0.65 m height). 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) Reactivity Control

The two (2) independent reactivity control systems are used to perform reactor emergency shutdown and maintenance in a sub-critical state: (i) Electromechanical reactivity control system based on control rods in the 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.

(e) Reactor Pressure Vessel and Internals

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

4. Safety Features

Safety objectives for the MHR-100 are achieved, first of all, by relying on the *inherent safety features*. These design features ensure thermal, neutronic, chemical and structural stability of the reactor. Safety is ensured by passive principles of system actuation. The decay and accumulated heat is removed from the core through RPV to reactor cavity cooling system and then to atmosphere by natural heat conductivity, radiation, and convection. In LOCA condition with failures of all active circulation systems and power sources, operation safety limit of the fuel is not exceeded.

(a) Engineered Safety System Approach and Configuration

In addition to the inherent features, the MHR-100 incorporates safety systems based on: (i) Simplicity of both system operation algorithm and design; (ii) Natural processes for safety system operation under accident conditions; (iii) Redundancy, physical separation and independence of systems; (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 DBA.

(c) Containment Function

Passive localization of 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 external loads due to 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%vol/d day at 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.

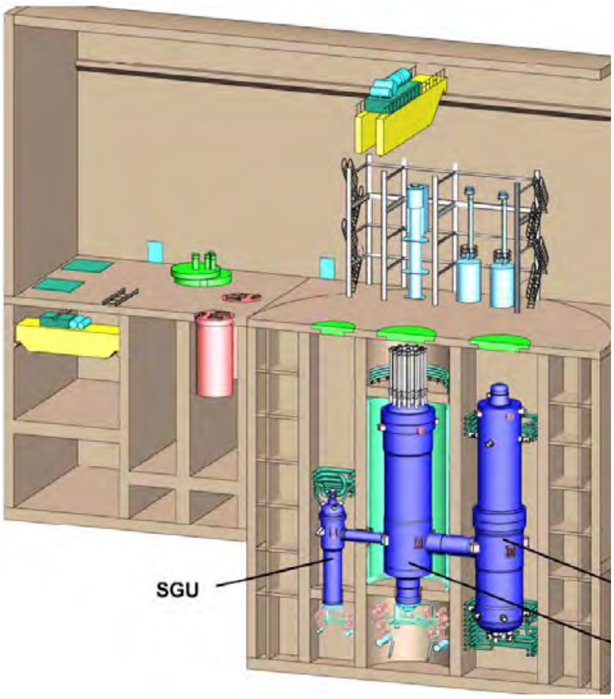
7. Design Variants and Plant Arrangements Based on the Modular MHR-100

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 UO_2-based coated fuel particles, high burnup and possibility to dispose the spent fuel blocks without additional reprocessing;
- High-performance high-temperature compact heat exchangers, high-strength casings of heat-resistant steel;
- Direct gas-turbine cycle 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.

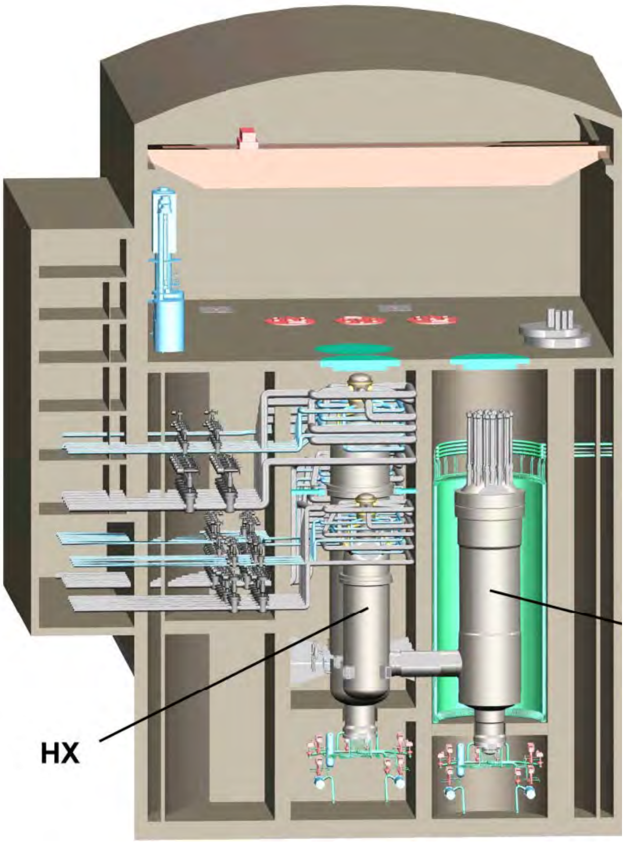


MHR-100 SE

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.

MAJOR TECHNICAL PARAMETERS			
MHR-100 SE		**MHR-100 SMR**	
Parameters	**Values**	**Parameters**	**Values**
Reactor heat capacity (MW)	215	Reactor heat capacity (MW)	215
Useful electric power of generator (MW)	87.1	Helium temperature at reactor inlet/outlet (°C)	450 / 950
Net power generation efficiency (%)	45.7	Helium flow rate through the reactor (kg/s)	81.7
Helium temperature at reactor inlet/outlet (°C)	553 / 850	Helium pressure at reactor inlet (MPa)	5.0
Helium flow rate through the reactor (kg/s)	138	Steam-gas mixture pressure at HX inlet (MPa)	5.3
Helium pressure at reactor inlet (MPa)	4.41	**HX-TCF 1**	
Expansion ratio in turbine	2.09	HX 1 capacity (MW)	31.8
Generator/TC rotation speed (rpm)	3000/ 9000	Helium/steam-gas mixture flow rate (kg/s)	12.1 / 43.5
Helium flow rate through turbine (kg/s)	126	Steam-gas mixture temp. at inlet/outlet (°C)	350 / 650
Helium temperature at PCU inlet/outlet (°C)	850 / 558	**HX-TCF 2**	
SG power (MW)	22.3	HX 2 capacity (MW)	58.5
Helium flow rate through SG (kg/s)	12.1	Helium/steam-gas mixture flow rate (kg/s)	22.2 / 60.9
Helium temperature at SG inlet/outlet (°C)	850 / 494	Steam-gas mixture temp. at inlet/outlet (°C)	350/750
Steam capacity (kg/c)	6.46	**HX-TCF 3**	
Steam pressure at SG outlet (MPa)	4.82	HX 3 capacity (MW)	125
		Helium/steam-gas mixture flow rate (kg/s)	47.4/101
		Steam-gas mixture temp. at inlet/outlet (°C)	350/870

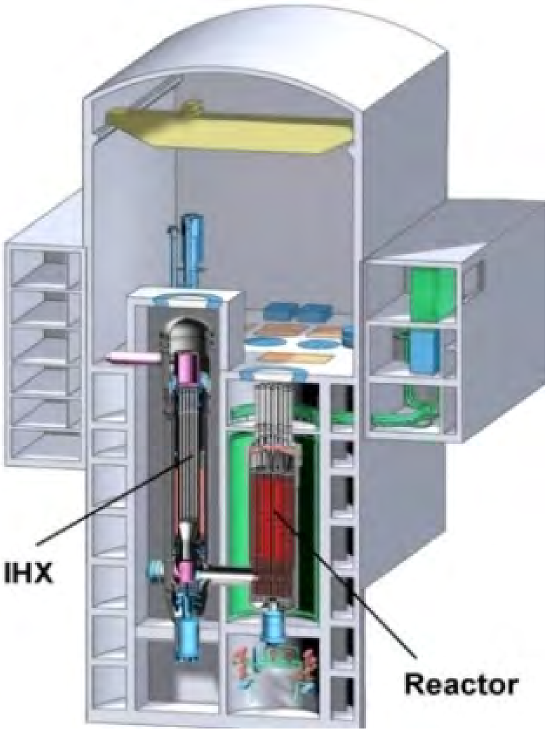


MHR-100 SMR

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. Studies of safety issues, with the emphasis on mutual influence of nuclear and hydrogen production components of the facility.

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



MHR-100 OR

9. Fuel Cycle Approach

The MHR-100 fuel cycle approach is a once through mode without reprocessing. 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 at fuel reloading. These measures are also applied at spent fuel management.

10. Waste Management and Disposal Plan

Facilities for long-term storage of spent nuclear fuel (SNF) and solid/solidified radioactive waste (RW) are included in the complex of a MHR-T commercial 4-unit NPP. The capacity of the designed SNF storage is determined from the condition of capability to store fuel unloaded from the NPP for 10 years. The capacity of solid/solidified RW storage facility is designed to provide storage of waste generated during the 10-year period of NPP operation. After 10 years of storage at the NPP site, SNF and RW are to be removed for final underground disposal. Radiochemical SNF reprocessing is considered as an option for future only.

11. Development Milestones

2014	Conceptual design completed
2018	Feasibility study of plant application for large-scale hydrogen production
2020	MHR-100-SMR is taken as the basis for near-term development of non-electricity nuclear applications in Russia

PBMR®-400 (PBMR 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
NSSS Operating Pressure (primary/secondary), MPa	9 (direct cycle / no secondary steam)
Core Inlet/Outlet Coolant Temperature (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
Reactivity control mechanism	Control rod insertion, negative temperature coefficient
Approach to safety systems	Active
Design life (years)	40
Plant footprint (m^2)	4200 (main structures only)
RPV height/diameter (m)	30 / 6.2 (inner)
RPV weight (metric ton)	1250 (with vessel head)
Seismic Design (SSE)	0.4g PGA for main power system design
Fuel cycle requirements / Approach	Uranium once through; spent fuel stored in tanks in the facility.
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 that are based on a direct Brayton cycle aims for higher efficiencies. The maximum achievable power levels for the reactor was increased in several design steps to achieve a set target for installed cost/kW that would be 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 reprioritization 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

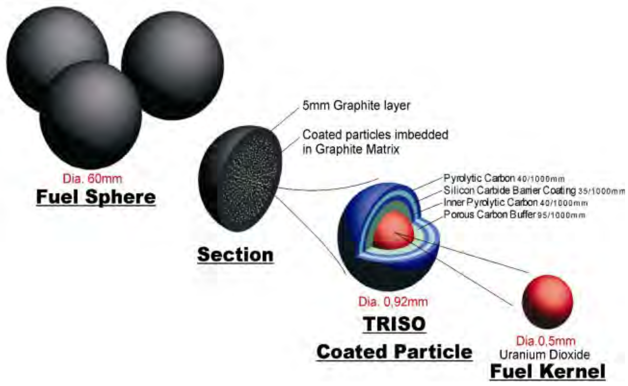
(a) Design Philosophy

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^{0}C) 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 pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. 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 turbine, compressors and power generator in a single shaft. 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

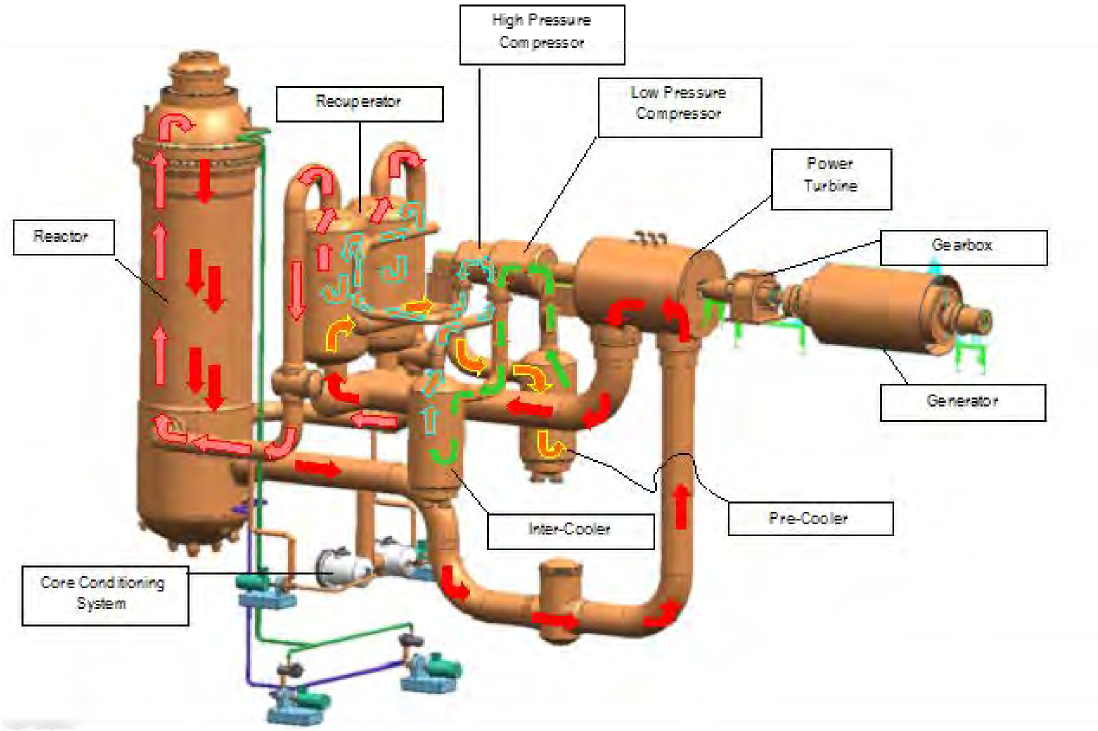
Excess reactivity is limited by continuous refuelling while adequate passive heat removal ensures an inherent safe design with no event with significant fission product release being possible. Adequate reactivity control and long-term cold shutdown capability is provided by two separate and diverse systems 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 back 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 modification 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.

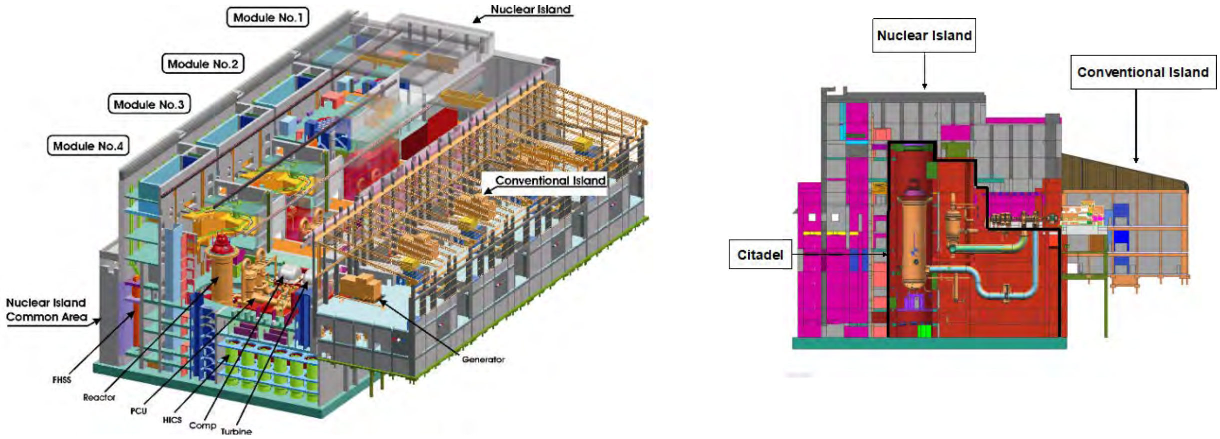
5. Plant Safety and Operational Performances

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 1.0 x 10^{-5} per reactor year while maintaining an expected capacity factor of 95%.

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. Fuel Cycle Approach

Once through uranium cycle was planned and analysed; pebble bed reactors are flexible to accommodate other fuel cycles (plutonium or thorium) too.

10. Waste Management and Disposal Plan

The Waste Handling System is designed to handle, store and discharge low- and medium-level liquid and solid radioactive waste generated during normal operation, maintenance activities, and upset conditions of the PBMR®-400; including preparation for the final disposal. During final decommissioning, the spent fuel spheres are removed from the Spent Fuel tanks and conveyed to a point where they can be loaded into the Spent Fuel Transport Casks suitable for final disposal at a designated site.

11. 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.; Alternative process heat markets and designs explored.
2010	Project closure.
2018	Project in care and maintenance.

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

Reactor System Configuration of AHTR-100

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 (ºC)	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
Fuel cycle requirements / Approach	Initially once through Uranium
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

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 (THTR300) 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.

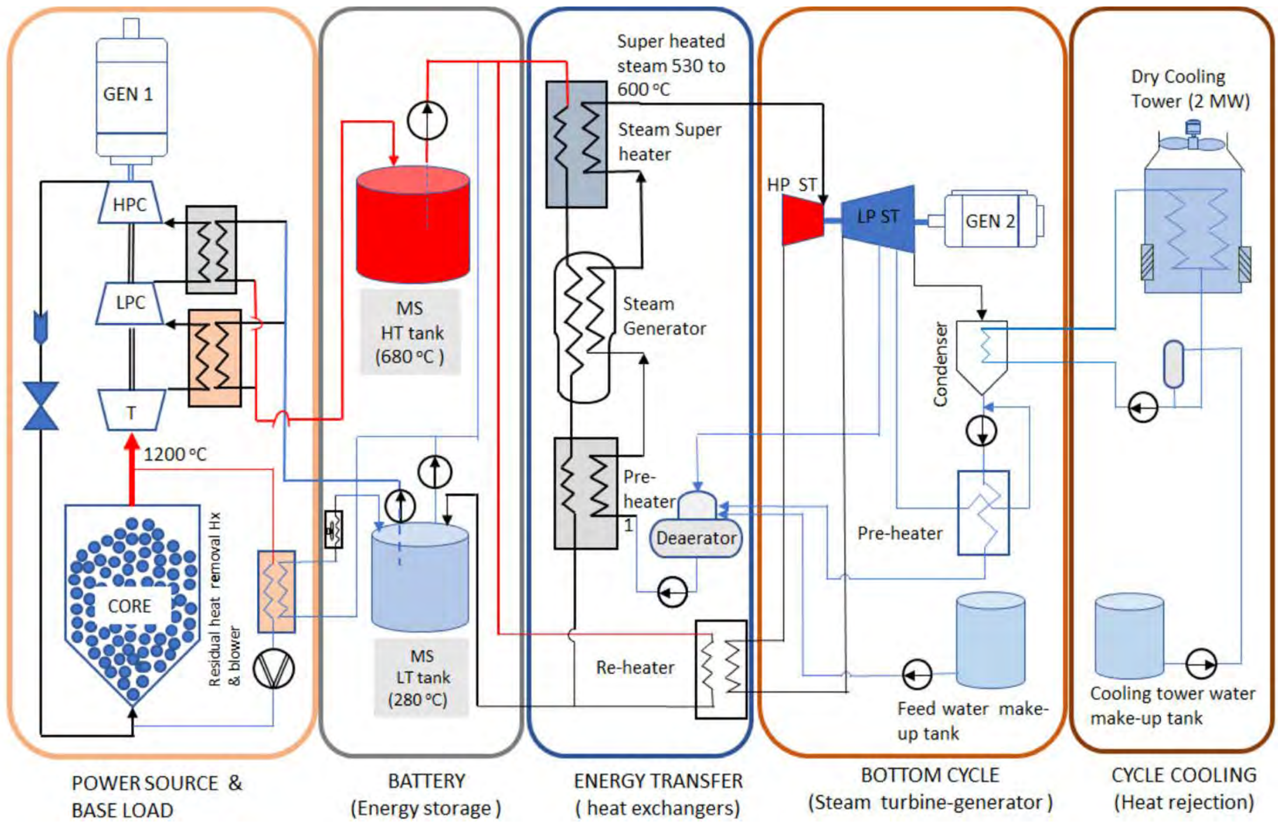
Given these potential advances available with current international technology, but not planned to be applied by other programs, significant performance improvements could be achieved over the performance envisaged for the original PBMR®.

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 below Figure.



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 (1200ºC) 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 pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. About 13 330 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%.

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 deisgn and construction of demonstration components.

8. Fuel Cycle Approach

Once through uranium cycle is planned and analysed; pebble bed reactors are flexible to accommodate other fuel cycles (plutonium or thorium) too.

9. Waste Management and Disposal Plan

The Waste Handling System would be based on the PBMR®-400 experience and designed to handle, store and discharge low- and medium-level liquid and solid radioactive waste generated during normal operation, maintenance activities, and upset events; including preparation for the final disposal. During final decommissioning, the spent fuel spheres are removed from the interim storage and conveyed to a point where they can be loaded into the Spent Fuel Transport Casks suitable for final disposal at a designated site.

10. 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.
2019	R&D put on hold pending funding availablility.

HTMR100 (STL Nuclear (Pty) Ltd., South Africa)

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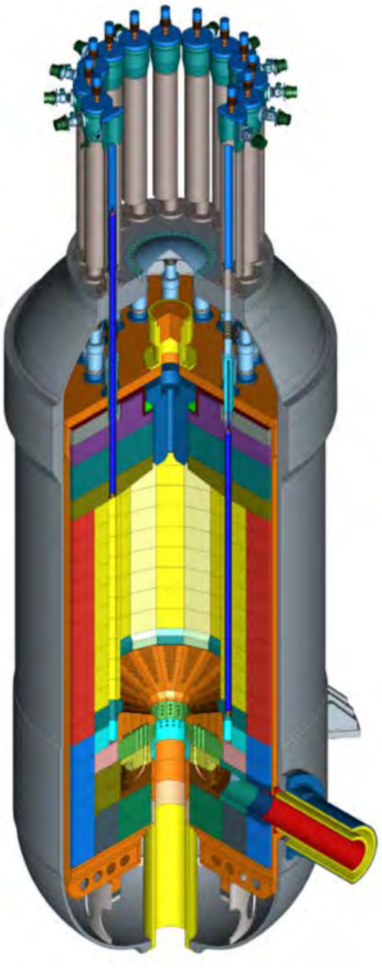


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	STL Nuclear, South Africa
Reactor type	HTGR (Pebble Bed)
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	100 / 35 single module plant
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	4 / -
Core Inlet/Outlet Coolant Temperature (°C)	250 / 750
Fuel type/assembly array	TRISO particles in pebbles: LEU, Th/LEU, Th/HEU or Th/Pu
Number of fuel assemblies in the core	~ 150 000 pebbles; around 125 to 150 pebbles/day throughput
Fuel enrichment (%)	Various, see description below
Core Discharge Burnup (GWd/ton)	80-90
Refuelling Cycle (months)	Online fuel loading
Main reactivity control mechanism	Absorber rods in the reflector
Approach to safety systems	Passive
Design life (years)	40 full power years
Plant footprint (m^2)	5000 buildings only
RPV height/diameter (m)	15.7 /5.6 (flange outer diameter)
RPV weight (metric ton)	155
Seismic Design (SSE)	0.3g for generic site; (0.5g under consideration)
Fuel cycle requirements / Approach	Various options – see text below
Distinguishing features	No core meltdown, no active engineered safety systems, spent fuel in acceptable form
Design status	Conceptual design

1. Introduction

The HTMR100 (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(e). 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 HTMR100 can supply 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 (later also upgradable to very high temperature). The HTMR100 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 HTMR100 reactors.

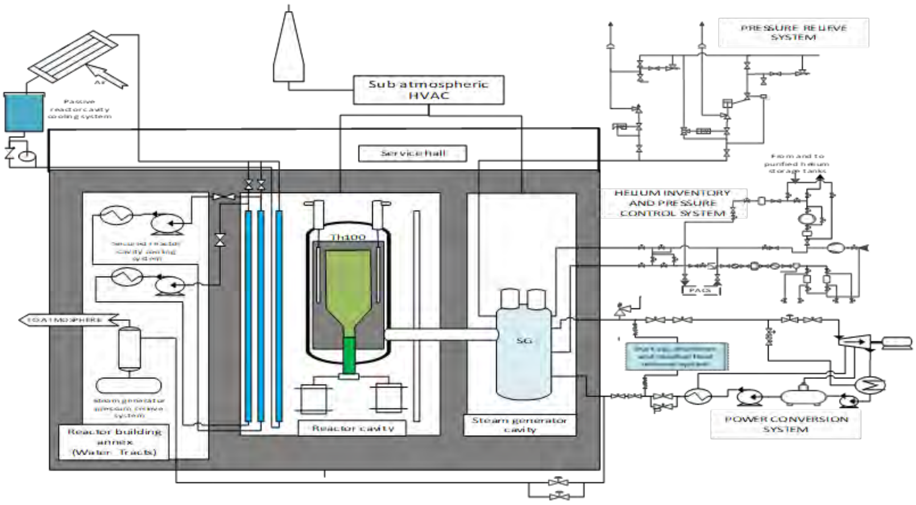
3. Main Design Features

(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 relatively 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 HTMR100 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 HTMR100 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 HTMR100 reactor (see below for details). A Fuel Qualification and Test programme will be conducted on the fuel prior to loading of the reactor. The HTMR100 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 B_4C absorber material, sandwiched between an inner and an outer tube segment. The inner tube segment allows cooling helium gas to flow form the top down in the circular channels. Each rod is equipped with a position indicator which measures the position of the rod over its entire positioning range and with position indicators for the upper and lower limit positions.

(f) Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel (RPV) is constructed in compliance with the ASME III subsection NB code. It comprises two main components reactor of 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 can withstand 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) Engineered Safety System Approach and Configuration

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 increases 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 Performances

The central consideration is the demand for high availability of process steam supply and/or electricity generation. To reduce or minimize the NSSS daily or weekly load changes of the reactor, the preference is to change the ratio between steam supply and electricity supply. Excess steam and/or electricity can be utilized in the desalination plants to provide water as a sellable commodity earning additional revenue. This allows the plant to operate virtually continually at full power very close to the plant availability.

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 HTMR100 system structure define control and instrumentation are plant control, data and instrumentation system, equipment/investment protection system and protection system.

7. Plant Layout Arrangement

(a) Reactor Building

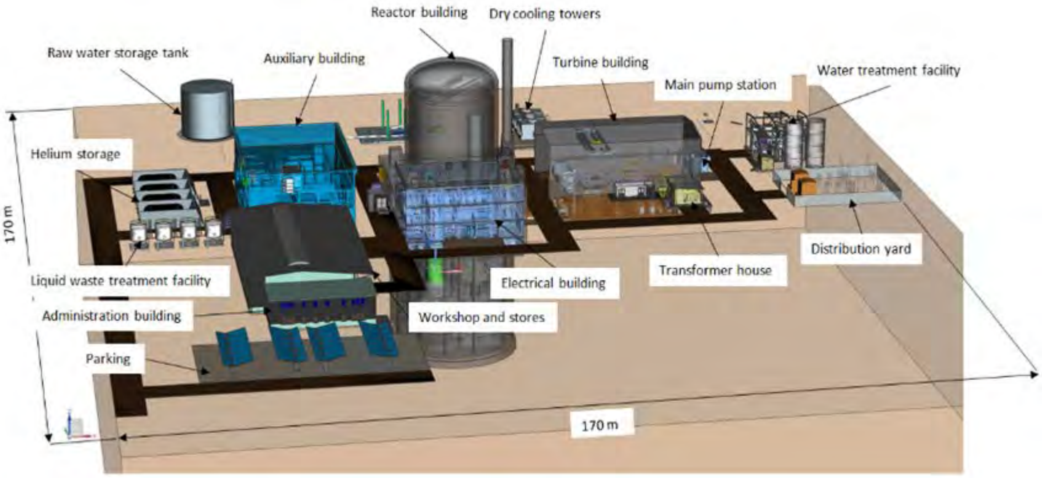
The reactor building contains the safety equipment that provides the necessary functions for the safe shutdown of the reactor under all design basis conditions. The reactor building is partially submerged below ground level such that the reactor and steam generator cavities are completely protected against 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 HTMR100.

(b) Balance of Plant Building

The Turbine Building provides the foundation and housing for the Power Conversion System, including other support systems such as the compressed air, water sampling, HVAC, Voltage Distribution Systems, permanent 11kVAC and 400VAC diesel generator sets and steam safety valves.

The Electric Building 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 was initiated with the aim of commencing the pre-assessment for licensing in order to reach design certification status at the end of the conceptual phase.

9. Fuel Cycle Approach

The reactor design can accommodate various fuel types with different fuel cycles, including mixtures of thorium and plutonium or thorium and uranium. Studies have shown the reactor can utilise: (i) 10% LEU (7-10g HM/sphere); (ii) Th/LEU mix of 50% LEU (20% enriched) and 50% Th (10-12g HM/sphere); (iii) Th/HEU mix with 10% HEU (93% enriched) and 90% Th (10-12g HM/sphere); and (iv) Th/Pu mix with 15% reactor grade Pu by mass, (12g HM/sphere). Reprocessing of the HTMR100 fuel elements is not intended

10. Waste Management and Disposal Plan

The HTMR100 fuel elements can be stored and disposed of a fuel spheres but available technology needs to be assessed for volume reduction.

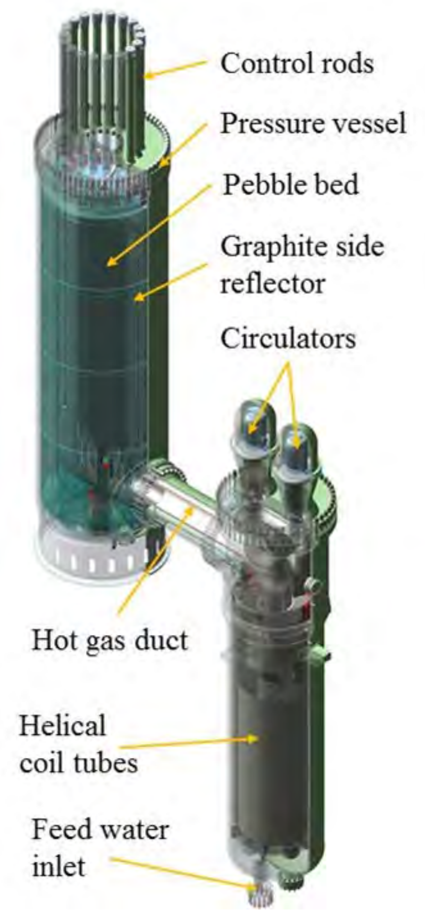
Disposal of spent spherical fuel elements from the HTMR100 is executed in the following sequence: (i) Direct transfer of spent fuel elements into a flask inside the cast iron high energy spent fuel casks (Hi-cask); (ii) Immediately after filling the H-cask they are sealed and transferred to the spent fuel cool-down facility on site; (iii) Once cooled down, the flask filled with fuel is transferred from the H—cask to a low energy spent fuel concrete cask (Low-Cask); iv) The Lo-Cask is transported to the low energy on-site interim storage facility; v) For offsite transport the flask is transferred to a shipping/transport cask for shipping to an ultimate repository.

Approximately 55 000 fuel elements will be discharged for on full-power year of operation and only one or two flasks containing physically damaged fuel spheres, singled out by the fuel unloading machine (failed fuel separator), may be required in the lifetime of the core. As in normal and accident conditions the coated particles maintain its excellent fission product retention capabilities and fission products are almost entirely retained within the fuel element kernels. Also, the release of these nuclides into the cask or flask atmosphere from the number of fuel element particles with defective SiC coatings is very low.

11. Development Milestones

2012	Project started
2021	Preparation for Pre-license application
2022	Conceptual design completed

Xe-100 (X Energy, LLC, United States of America)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	X Energy, LLC, United States of America
Reactor type	Modular HTGR
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	200 / 82.5
Primary circulation	Forced helium circulation
NSSS Operating Pressure (primary/secondary), MPa	6.0 / 16.5
Core Inlet/Outlet Coolant Temperature (ºC)	260 / 750
Fuel type/assembly array	UCO TRISO/pebbles
Number of fuel assemblies in the core	220 000 pebbles per reactor
Fuel enrichment (%)	15.5
Core Discharge Burnup (GWd/ton)	165
Refuelling Cycle (months)	Online fuel loading
Reactivity control mechanism	Thermal feedback & control rods
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	340m x 385m (4 reactor modules with 4 turbines)
RPV height/diameter (m)	16.4 / 4.88
RPV weight (metric ton)	274
Seismic Design (SSE)	0.5g
Fuel cycle requirements / Approach	Uranium once through (initially)
Distinguishing features	Online refuelling, core cannot melt and fuel damage minimized by design, independent radionuclide barriers, potential for advanced fuel cycles
Design status	Basic design development

1. Introduction

The Xe-100 is a pebble bed high-temperature gas-cooled reactor with 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 165 000 MWd/tHM.

2. Target Application

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

3. Main 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 ~220 000 graphite pebbles fuel elements each containing ~18 000 UCO TRISO coated particles. The core is graphite moderated with online refueling capability. The advantage of online

refuelling 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 165 000 MWd/tHM. At full power approximately 173 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 > 3920°C (4200K) and fuel temperature never exceeds 1100°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 comprises four main subsystems/components: new fuel loading system; fuel unloading and recirculation system; fuel burnup-measurement system; and spent fuel handling and 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 = 1000 pcm), 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

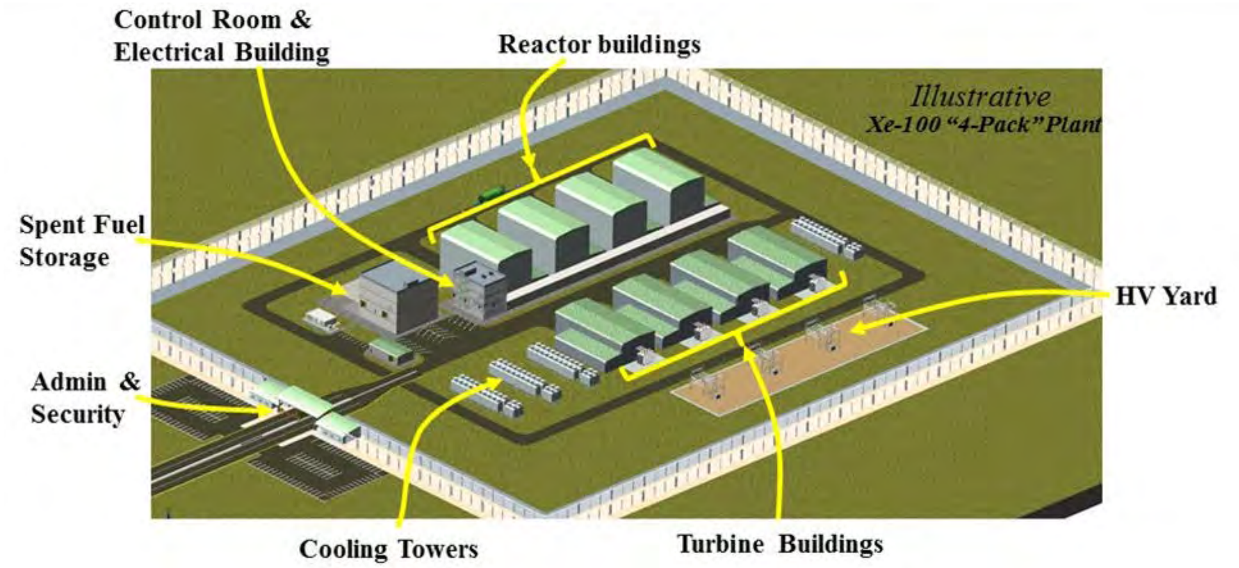
(i) Key safety features to limit plant transients:
The RCSS insertion depth during normal operation binds around 1.4 $\delta_{k\text{-}eff}$. Any spurious signal that would cause full withdrawal of the RCSS would therefore only translate to a higher temperature that would remain below an allowable value 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 Systems

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 Layout Arrangement

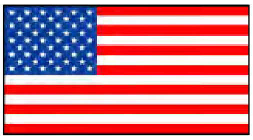


8. Design and Licensing Status

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	Applications submitted to the U.S. Nuclear Regulatory Commission
2025	Start of Construction



SC-HTGR (Framatome Inc., United States of America)

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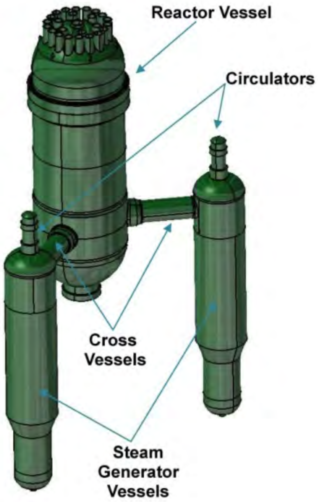
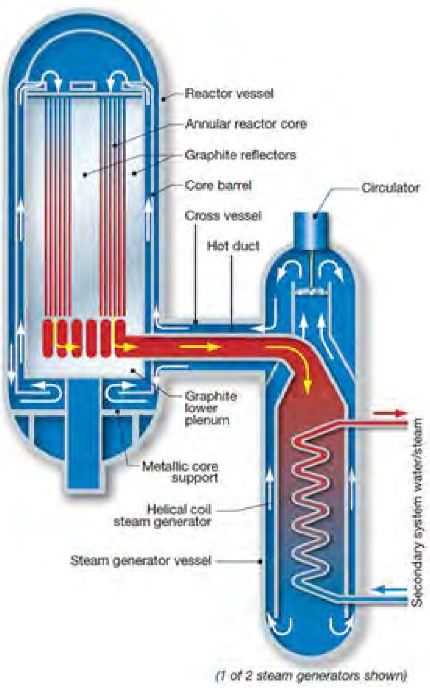




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Framatome Inc., United States of America
Reactor type	Prismatic block HTGR
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	625 / 272
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	6 / 16
Core Inlet/Outlet Coolant Temperature (°C)	325 / 750
Fuel type/assembly array	UCO TRISO particle fuel in hexagonal graphite blocks
Number of fuel assemblies in the core	Annular core, 102 column, 10 blocks / column
Fuel enrichment (%)	14.5 average / 18.5 maximum
Core Discharge Burnup (GWd/ton)	165
Refuelling Cycle (months)	½ of the core replaced every 18 months; 21 days outage
Reactivity control mechanism	Control rods (gravity insertion) Independent reserve shutdown system (gravity insertion) large negative temperature coefficient
Approach to safety systems	Active/Passive
Design life (years)	80
Plant footprint (m^2)	10 000
RPV height/diameter (m)	24 / 8.5
RPV weight (metric ton)	880
Seismic Design (SSE)	0.5g
Fuel cycle requirements / Approach	LEU once-through fuel cycle / options for later consideration.
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	Conceptual 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 co-located 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 KWU/INTERATOM HTR-MODUL.

2. Target Application

The SC-HTGR produces high temperature steam suitable for numerous applications including industrial process heat, H_2 production, 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 is designed to produce high temperature steam for serving a wide variety of near-term markets. The steam cycle is also well suited to cogeneration of electricity and process heat.

3. Main Design Features

(a) Design Philosophy

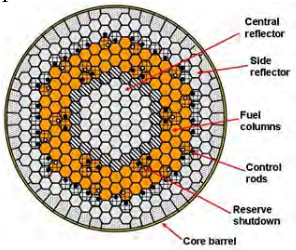
The SC-HTGR is designed around proven helium-cooled, graphite moderated reactor technology, and passive decay heat removal, the heart of which is the safety characteristics of TRISO coated fuel particles.

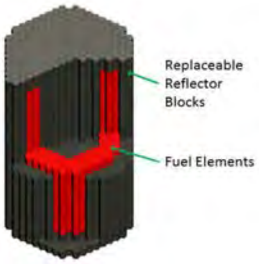
(b) Reactor Core and Power Conversion Unit

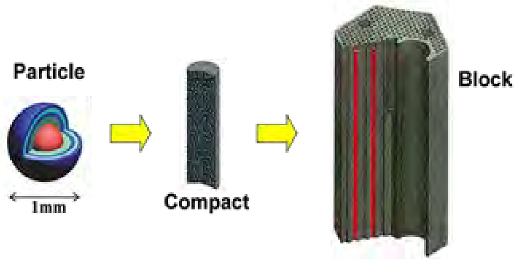
The reactor inlet and outlet temperatures are 325°C and 750°C, respectively. These temperatures were selected primarily to support the desired steam outlet conditions for the target markets. These temperatures also allow the use of SA-508/533, a standard PWR vessel material, for the primary system vessels without requiring cladding, separate cooling or special thermal protection. For the reference four-module steam cycle concept plant, the single reactor module 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 barriers 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.







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(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 Reactor module design can be scaled from 625 MW(t) to 16 MW(t) per module using the same fuel blocks in scaled arrangements.

(e) Cycle Length and Fuel Management

The core cycle length for the reference 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 for the optimization of core power distribution in three dimensions and can also be used to support effective fuel utilization, proliferation resistance and waste reduction.

(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 and active reactivity control systems provide further confidence of the ability to shut down the reactor.

(g) Fuel Handling System

Refuelling is performed using robotic systems with the primary coolant boundary intact. Following shutdown, the primary system temperature rapidly is reduced followed by the helium inventory reduction to slightly sub-atmospheric. Refuelling access is then gained through the control rod drive penetrations at the top of the reactor vessel. The robotic refuelling operation is then commenced using predetermined fuel and reflector 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 (EAB) 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.

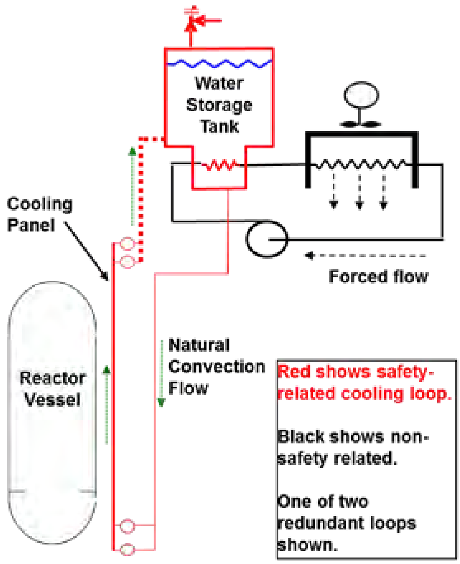
The primary radionuclide retention barrier in the SC-HTGR consists of the three ceramic coating layers surrounding the fuel kernel that forms a coated fuel particle radionuclides retention mechanism. The coating system constitutes a micro-scale pressure vessel around each kernel that has been engineered 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 active cooling. Helium, the reactor coolant and heat transport medium, is chemically inert and neutronically transparent. Helium will not change phase during normal operation or accidents.

The SC-HTGR is designed to passively remove decay heat from the core regardless of whether or not the primary coolant is present. The concrete walls surrounding the reactor vessel are covered by the Reactor Cavity Cooling System panels, which provide natural circulation cooling during both normal operation and accidents, so there is no need for the system to 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.

(b) Engineered Safety System Approach and Configuration

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

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 scenarios.

5. Plant Safety and Operational Performances

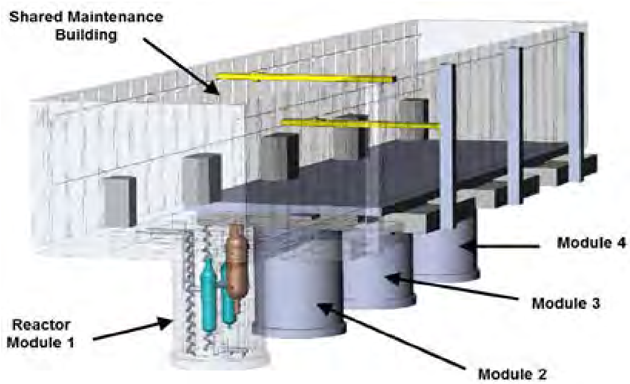
The nominal electricity generation performance of the SC-HTGR system has been evaluate, taking into account preliminary efficiency estimates for the helium circulators, feedwater pumps, turbine, generator and other plant electrical loads. The net electrical output from each 625 MW(t) reactor module is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

7. Plant Layout 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-license application has been completed.

9. Fuel Cycle Approach

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. The TRISO coated particle fuel can also be recycled when and if such process is mandated and economically viable.

10. Waste Management and Disposal Plan

Due to higher thermal efficiencies the storage and disposal requirements, which largely depend on fission product decay heat, will be lessened by about 50% for HTGRs as compared to LWRs. The spent fuel capacity is 10 years. Coated particles represent excellent radioactivity containment characteristics also for long term storage or disposal, volume reduction and reprocessing is also options studied in the past.

11. Development Milestones

2022	Conceptual Design
2024	Preliminary Design
2024	Start of pre-licencing vendor design review in the USA.
2027	FOAK plant engineering design complete; Secure necessary licenses in the USA.
2027	Start construction of a first full-scale NPP module in the USA.
2033	Commercial operation

HTR-10 (Tsinghua University, China)

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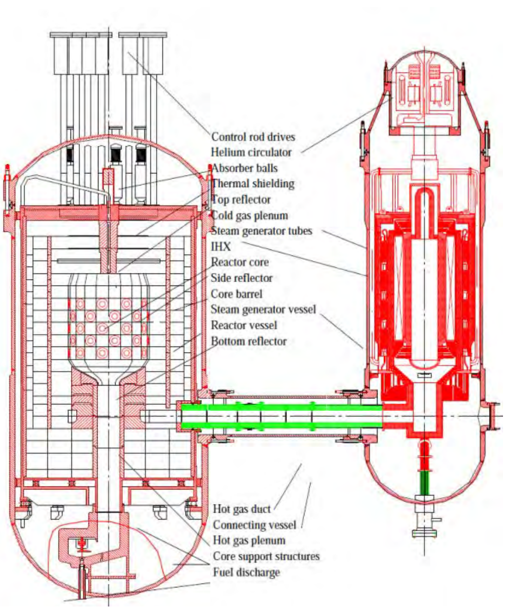


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	INET, Tsinghua University, People's Republic of China
Reactor type	Pebble bed modular high temperature gas-cooled test reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 / 2.5
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	3 / 4
Core Inlet/Outlet Coolant Temperature (^{o}C)	250 / 700
Fuel type/assembly array	Spherical elements with TRISO particles fuel (UO_2 kernel)
Number of fuel assemblies in the core	27 000 spherical fuel elements
Fuel enrichment (%)	17
Core Discharge Burnup (GWd/ton)	80
Refuelling Cycle (months)	On-line refuelling
Reactivity control mechanism	Control rod insertion/ negative temperature feedback
Approach to safety systems	Combined active and passive
Design life (years)	20 (test reactor)
Plant footprint (m^2)	100x130
RPV height/diameter (m)	11.1 / 4.2
RPV weight (metric ton)	167
Seismic Design (SSE)	3.3 m/s^2
Fuel cycle requirements / Approach	17% enriched LEU is needed for such a small test reactor; Small amount of material to be included in national programme.
Distinguishing features	To verify and demonstrate the technical and safety features; and to establish an experimental base for process heat applications
Design status	Operational

1. Introduction

In 1992, the China Central Government approved the construction of the 10 MW(t) 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. Afterwards, INET conducted many experiments using 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.

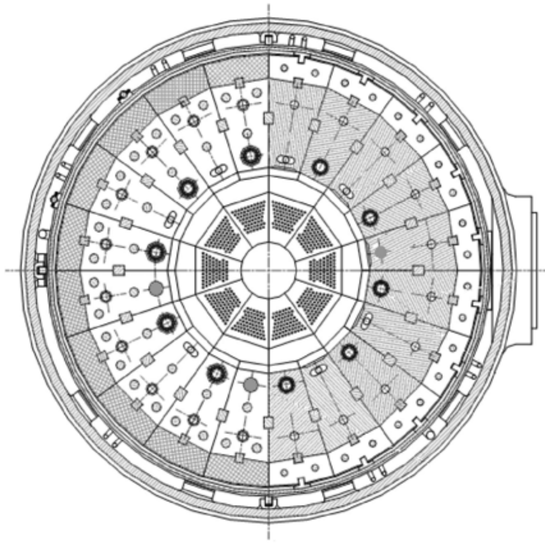
2. Target Application

The HTR-10 is a major project on the energy sector within the Chinese National High Technology Programme, serving as the first major step of the development of modular HTGR in China. Its main objectives are to: (1) explore the technology in the design, construction and operation of HTGRs; (2) establish an irradiation and experimental facility; (3) demonstrate the inherent safety features of modular HTGR; (4) test electricity and heat co-generation and closed cycle gas turbine technology; and (5) perform research and development work on nuclear process heat application. The aims of this project are to demonstrate the inherent safety features of the HTGR modular design and test the technologies of electricity generation, district heating as well as process heat application with modular HTGR.

3. Main Design Features

(a) Design Philosophy

The primary pressure boundary consists of reactor pressure vessel, steam generator pressure vessel and hot gas duct pressure vessel which connects the above two vessels. This arrangement can make the maintenance and inspection of the components easier and mitigate the accident result of water ingress into reactor core if the steam generator heat transfer tubes might fail.



(b) Reactor Core

The reactor core volume is 5m^3, 1.8 m in diameter and the mean height is 1.97 m. About 27 000 spherical fuel elements with 60 mm in diameter will be filled up in the reactor core, the enrichment of fuel is 17% and the mean discharge burn up is designed to be 80 000 MWd/tU. The reactor core is entirely constructed by graphite materials, no metallic components are used in the region of the core. At the funnel bottom of the reactor core, there is a fuel-element discharging tube with a diameter of 500 mm and a length of 3.3 m. At the tube end the special fuel discharge facility singularise the fuel to be unloaded through a 65 mm diameter pipe that penetrates the reactor pressure vessel.

(c) Reactor Pressure Vessel and Internals

The pressure vessel unit consists of the reactor pressure vessel, the steam pressure vessel and the hot gas duct pressure vessel. The upper part of the reactor pressure vessel is a cover which is connected via eighty bolts, and its lower part is a cylindrical shelf with a lower closure head. A metallic O-ring and an Ω-ring are used for sealing between the upper and lower parts. The tube nozzle for irradiation channels and the control rods driving system are mounted on the cover.

(d) Reactor Coolant System

Cold helium channels are designed within the side reflector for the helium primary coolant to flow upward after entering the reactor pressure vessel from the annular space between the connecting vessel and the hot gas duct. Helium flow reverses at the top of reactor core to enter the pebble bed, so that a downward flow pattern takes place. After being heated in the pebble bed, helium then enters into a hot gas mixing chamber in the bottom reflector, and from there it flows through the hot gas duct and then on to the heat exchanging components.

(e) Steam Generator

The steam generator (SG) is a once through, modular helical tube type. Hot helium from the hot gas duct flows through its central tube to the top part of the SG and then is fed in above the SG heat transfer tubes. While flowing around the tubes, the helium releases its heat to the water/steam side, thereby cooling down from 700°C to 250°C. The cold helium flow is then deflected to the inlet of the helium blower and returns to the reactor along the wall of the pressure vessel. The water flows through the helical tubes from the bottom to the top. The feed water temperature is 104°C and the steam temperature at the turbine inlet is 435°C. The SG mainly consists of the pressure vessel, the steam generator tube bundle modules and the internals.

(f) Helium Circulator

The helium circulator is a key component for high temperature helium cooled reactors and therefore an important component to develop and test in the HTR-10. The helium circulator assures the thermal energy transfer from the reactor core to the steam generator and operates at 3.0 MPa and at 250°C. The circulator is integrated into the steam generator vessel and installed on top of the SG. The helium circulator was designed and manufactured by INET at Tsinghua University and the Shanghai Blower Works Co., Ltd.

(g) Fuel Characteristics

The fuel elements are the spherical type fuel elements, 6 cm in diameter with coated particles. The reactor equilibrium core contains about 27 000 fuel elements forming a pebble bed that is 180 cm in diameter and 197 cm in average height. The spherical fuel elements move through the reactor core in a multi-pass pattern.

(h) Fuel Handling System

The HTR-10 is designed to use spherical fuel elements. Its Fuel Handling System (FHS) is different from the refuelling machines of reactors using rod shaped or block shaped fuel elements. The main feature of the FHS is to charge, circulate and discharge fuel elements in the course of the reactor operation, or in other words on-line. For the initial core loading, dummy balls (graphite balls without nuclear fuel) were firstly placed into the discharge tube and the bottom cone region of the reactor core. Then, a mixture of fuel balls and dummy balls were loaded gradually to approach first criticality. The percentages of fuel balls and dummy balls were 57% and 43% respectively. After the first criticality was reached, mixed balls of the same ratio were further loaded to fill the core in order to make the reactor capable of being operated at full power. The full core (including the cone region) is estimated to have a volume of 5 m^3.

(i) Reactivity Control

There are two reactor shutdown systems, one control rod system and one small absorber ball system. They are all designed in the side reflector. Both systems are able to bring the reactor to cold shutdown conditions. Since the reactor has strong negative temperature coefficients and decay heat removal does not require any circulation of the helium coolant, the turn-off of the helium circulator can also shut down the reactor from power operating conditions. There are ten control rods placed in the side reflector. Boron carbide (B_4C) is used as the neutron absorber. Each control rod contains five B_4C ring segments which are housed in the area between an inner and an outer sleeve of stainless steel. These are then connected together by metallic joints. The inner and outer diameter of the B_4C ring is 60 mm and 105 mm respectively, while the length of each ring segment is 487 mm. There are 7 holes in the side reflector of the HTR-10 for small absorber ball system.

4. Safety Features

HTR-10 has inherent safety features common to the new generation of advanced reactors, i.e. the reactor automatically shuts down because of the negative temperature reactivity coefficients and the decay heat is passively removed from the reactor to the environment. HTR-10 is a new generation reactor whose design is based on the ideas of module reactors.

(a) Reactivity control

The on-line refuelling leads to a small excess of reactivity, the overall temperature coefficient of reactivity is negative, and two independent shutdown systems are available.

(b) Decay Heat Removal System

After shutdown, the decay heat will be dispersed from the core to outside of reactor pressure vessel via conduction, convection and radiation, even in the case of depressurized accident condition. Then the decay heat can be carried out by two independent trains of passive decay heat removal systems to environment. Two independent reactor cavity coolers are located at the surface of the reactor cavity. During an accident, the decay heat is removed to the environment by the passive heat transfer mechanisms, i.e. heat conduction, natural convection and thermal radiation.

(c) Containment Function

There are three barriers to the release of fission products to the environment, i.e. the coating layers of the TRISO coated fuel particles, the pressure boundary of the primary loop and the confinement. In any accidents the maximum temperature of the fuel elements could not exceed the temperature limit and a significant radioactivity release can be excluded. In addition, the low free uranium content of fuel elements, the retention of radioactivity by graphite matrix of fuel elements, and the negligible activated corrosion products in the primary coolant system will maintain the radioactivity of the primary coolant system at a very low level. In the depressurization accidents of the primary coolant, the impact of radioactivity release on the environment will be insignificant. Therefore, it is not necessary to provide containment for the HTR-10. Therefore, a confinement without requirement of pressure-tightness is adopted.

5. Plant Safety and Operational Performances

There are two operational phases for the HTR-10. In the first phase, the plant is operated at a core outlet temperature of 700°C and inlet of 250°C. The secondary circuit include a steam turbine cycle for electricity generation with the capability for district heating. The steam generator produce steam at temperature of 440°C and pressure of 4 MPa to feed a standard turbine-generator unit. In the second phase (not implemented yet), the HTR-10 will be operated with a core outlet temperature of 900°C and an inlet of 300°C. A gas turbine (GT) and steam turbine (ST) combined cycle for electricity generation is in preliminary design. The intermediate heat exchanger (IHX), with a thermal power of 5 MW, provides high temperature nitrogen gas of 850°C for the GT cycle. There are other options under consideration to operate HTR-10 in higher temperature mode.

6. Instrumentation and Control Systems

The control system makes use of the distribution control system (DCS). Full digitalized control room and reactor protection system are used in HTR-10.

HTR-10 Control room.

7. Plant Layout Arrangement

The HTR-10 plant includes the reactor building, a turbine/generator building, two cooling towers and a ventilation center and stack. These buildings are arranged and constructed on an area of 100 x 130 m^2. The HTR-10 plant does not contain a leak-tight pressure containing system. The concrete compartments that house the reactor and the steam generator as well as other parts of the primary pressure boundary are preferably regarded as confinement.

8. Design and Licensing Status

HTR-10 is operational.

9. Fuel Cycle Approach

For the test reactor a once through fuel cycle is initially implemented.

10. Waste Management and Disposal Plan

To be included in the national plan of test facilities.

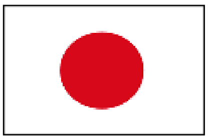
11. Research and Development Plan

From 1986 to 1990, eight (8) research topics for key technologies were defined: (i) a conceptual design and the supporting reactor physics and thermal fluid design and safety software codes; (ii) manufacturing process of the fuel spheres; (iii) reprocessing technologies for the thorium-uranium cycle; (iv) core internal graphite structure design and supporting analysis; (v) helium technology establishment, (vi) pressure vessel designs, (vii) the fuel handling design; (viii) development of special materials.

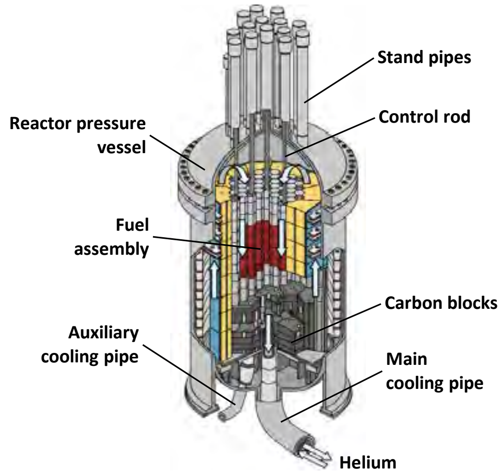
Before the commissioning, the following engineering experiments were conducted: (i) a hot gas duct performance test; (ii) measurements to establish the mixing efficiency at the core bottom (limit stratification and heat streaks); (iii) two-phase flow stability tests on the once-through steam generator; (iv) fuel handling performance test; (v) control rods drive mechanism performance; (vi) V&V of the digital reactor protection systems; (vii) measurements to confirm the neutron absorption cross-section of the reflector graphite and (viii) a performance test for the helium circulator.

12. Development Milestones

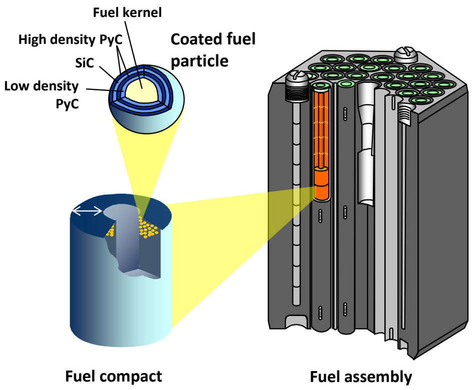
1992	Project approved
1995	Construction began
2000	First criticality
2001	HTR-PM Project is launched
2003	Commission date and full power operation
2018	Restart after upgrade of systems; Melt-wire tests to measure temperatures conducted.



HTTR (JAEA, Japan)



HTTR 30MW



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JAEA in cooperation with MHI, Toshiba, IHI, Hitachi, Fuji Electric, NFI, Toyo Tanso, Japan
Reactor type	Prismatic HTGR
Coolant/moderator	Helium / graphite
Thermal power, MW(t)	30
Primary circulation	Forced by gas circulators
Primary coolant pressure, MPa	4
Core Inlet/Outlet Coolant Temperature, ºC	395 / 850 (950 max.)
Fuel type/block array	UO_2 TRISO ceramic coated particle
Number of fuel block in core	150
Fuel enrichment, wt%	3 – 10 (6 avg.)
Average fuel discharged burnup, GWd/t_{HM}	22 (33 max.)
Refuelling Cycle, days	660 EFPD
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Active
Design lifetime, years	~20 (Operation time)
Plant area, m^2	~200m × 300m
RPV height/diameter, m	13.2 / 5.5
Seismic Design (SSE)	> 0.7m/s^2 automatic shutdown
Distinguishing features	Safety demonstration test
Status	Operational

1. Introduction

The High Temperature Engineering Test Reactor (HTTR) is Japan's first High Temperature Gas-cooled Reactor (HTGR) established in the Oarai Research and Development Institute of Japan Atomic Energy Agency (JAEA). The HTTR has superior safety features by using coated fuel-particle, graphite moderator, and helium gas coolant. With the potential of supplying high temperature heat above 900°C, HTGR can be used not only for power generation but also for process heat in several industrial fields. JAEA conducted long-term high temperature operation (950°C/50days operation) to demonstrate the capability of high temperature heat supply. It then conducted a loss of forced cooling (LOFC) test (at 30% power) to demonstrate the inherent safety feature of HTGR in 2010. The LOFC test simulates the severe accident in which the reactor coolant flow is reduced to zero and the reactor scram is blocked. The test result shows that the reactor could be shut down and kept in a stable condition without any operation management. JAEA has accumulated useful data for the development of future commercial HTGR system though the design, construction, and operation of the HTTR.

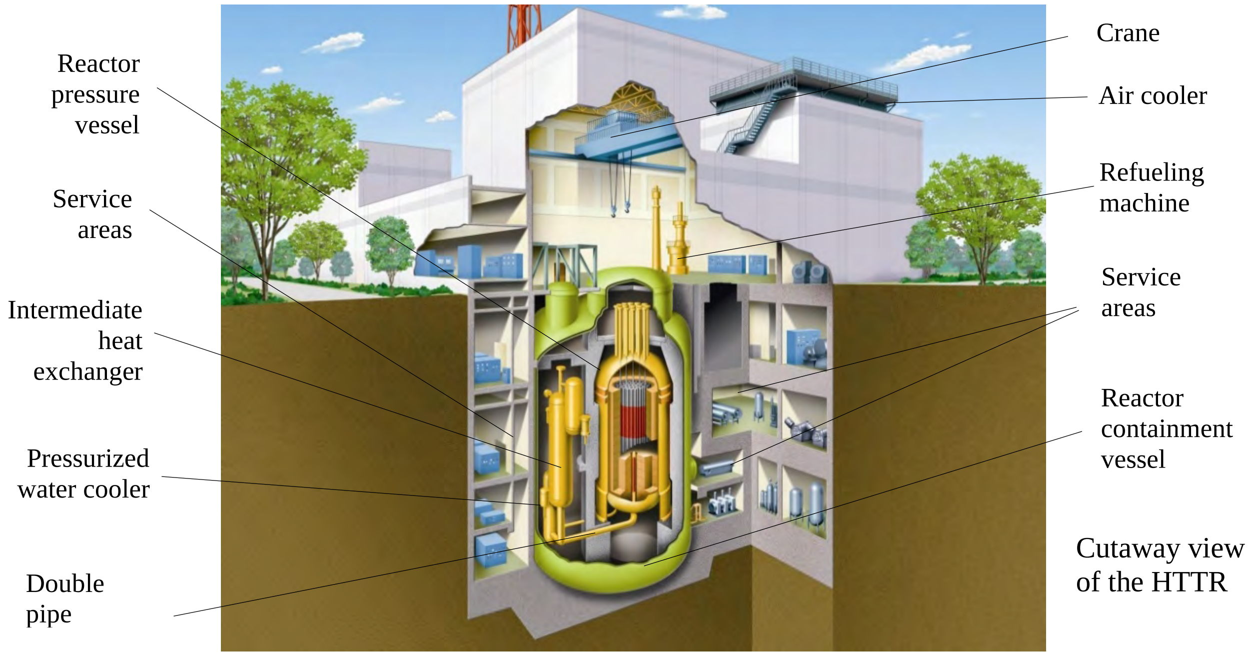
2. Target Applications

The objectives of HTTR are to: (i) stablish and upgrade the technological basis for the advanced HTGR; (ii) Perform innovative basic research in the field of high temperature engineering; and (iii) Demonstrate high temperature heat applications and utilization achieved from nuclear heat.

3. Main Design Features

(a) Design Philosophy

Illustrated in the figure below, the reactor building is designed with five levels of three underground floors and two upper ground floors. The reactor building is 18.5 m in diameter, 30 m in height. The cylindrically shaped containment steel vessel contains the reactor pressure vessel, the intermediate heat exchanger, the pressurized water cooler and other heat exchangers in the cooling system.



Cutaway view of the HTTR

The reactor core is designed to keep all specific safety features within the graphite blocks. The intermediate heat exchanger is equipped to supply high temperature clean helium gas for process heat application systems. The instrumentation and control system are designed to allow operations which simulate accidents and anticipated operational occurrences. As the HTTR is the first HTGR in Japan and a test reactor with various purposes, it incorporates specific aspects regarding safety design. JAEA established the safety design principles for HTTR in reference to the ‘Guidelines for Safety Design of LWR Power Plants’, but taking into account the significant safety characteristics of HTGR and corresponding design requirements as a test reactor.

(b) Reactor Core

The HTTR reactor consists of reactor internals and core components. The reactor internals comprise the graphite and metallic core support structures and shielding blocks. They support and arrange the core components, such as fuel blocks and replaceable reflector blocks within the reactor pressure vessel (RPV). The core components are made up of the same prismatic blocks of 360 mm width across the flats and 580 mm in height, including replaceable reflector blocks, irradiation blocks, control rod guide blocks, and fuel assembly blocks. The 2.9m in height, 2.3m in diameter core is surrounded by the permanent reflector made of graphite. The active core region consists of 30 fuel columns and 7 control rod guide columns while the reflector region contains 9 additional control rod guide columns, 12 replaceable reflector columns, and 3 irradiation columns.

(c) Fuel

The HTTR employs the TRISO (Tri-structural isotropic)-coated fuel particles (CFPs) with UO_2 fuel kernel. There are four layers surrounding the fuel kernel, including a low-density porous pyrolytic carbon (PyC) buffer layer, followed by a high-density PyC layer, a SiC layer, and an outer high-density PyC layer. Approximately 13-thousand CFPs are fabricated in a graphite matrix of fuel compact. There are 14 fuel compacts in a fuel rod. Each fuel assemblies contains 31 or 33 fuel rods.

The fabrication of the first-loading fuel for the HTTR started in June 1995. A total of more than 60-thousand fuel compacts, corresponding to about five-thousand fuel rods, were successfully produced through the fuel kernel, coated fuel particle, and fuel compact processes. The fuel rods were transferred to the reactor building of HTTR, where they were inserted into the graphite blocks to form the fuel blocks. In December 1997, 150 fuel assemblies were completely formed and stored in new fuel storage cells.

(d) Reactivity control system

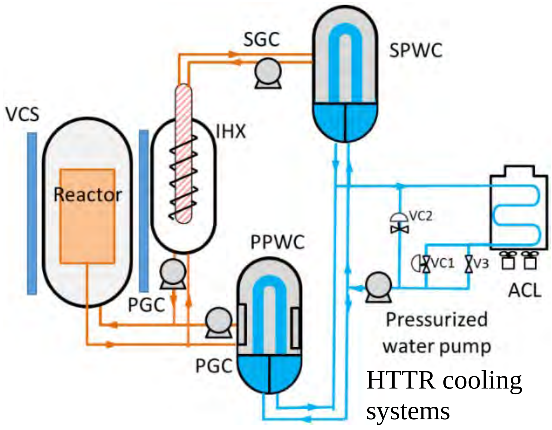
The HTTR contains two reactivity control systems, including a control rod system and a reserve shutdown system (RSS). The control rod system comprises of 16 pair of control rods made of B_4C. Each pair of control rods can move individually by control rod drive mechanisms located in standpipes at the top head closure of the RPV. In the event of a scram, the control rods can freely fall into the core by gravity. There are 7 pairs of control rod in the active core and 9 pairs in the reflector region.

The RSS is located in the standpipes along with the control rod and can be inserted into the third hole of control rod guide block. The RSS consists of driving mechanism, hopper, guide tube, etc. The hopper contains B_4C/C pellet. When the RSS is activated, the hopper is opened and the B_4C/C pellets drop into the reactor by gravity. The RSS was designed to be able to make the reactor subcritical from any operation condition at a temperature range from 27°C to 950°C.

(e) Cooling systems

The cooling systems of HTTR are composed of a main cooling system (MCS), an auxiliary cooling system (ACS), and a vessel cooling system (VCS). Under a normal condition, the heat of 30 MW from the reactor core could be removed by the MCS with two loading modes. One is a single loaded operation mode where 30MW thermal from the reactor is cooled by only the primary pressurized water cooler. Another is a parallel loaded operation mode, in which 10 MW and 20 MW thermal are separately removed by the helium-helium intermediate heat exchanger and the primary pressurized water cooler, respectively. The helium-helium intermediate heat exchanger of HTTR is operated at the highest temperature in the world.

The ACS consists of the auxiliary heat exchanger, auxiliary gas circulators, and air cooler. The heat transfer capacity of the ACS is about 3.5 MW. The ACS automatically starts up when the reactor is scrammed and the MCS is stopped abnormally. The residual heat of the core can also be removed by the VCS without the activation of ACS.



HTTR cooling systems

4. Safety Features

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

- Helium coolant is chemically stable. It does not react chemically with fuel and core structures so that hydrogen gas is not produced by chemical reaction of fuel element in accident like LWR;
- The CFPs of HTTR have excellent heat-resistant property which can bear very high temperature condition over 2200°C without any fission product release. The HTTR is designed that the fuel temperature does not exceed 1600°C in any accident to prevent fuel damage;
- Graphite-moderated reactor core provides a negative reactivity coefficient, low-power density, and high thermal conductivity. Graphite core structure also can withstand up to 2500°C without any thermal damage.

The HTTR can remove the residual heat of the core inherently because of optimized low reactor power density and graphite core structure. If the forced cooling performance was lost in an accident, decay heat of fuel transfers to reactor vessel through the core graphite structure slowly by thermal conduction and radiation. The fuel temperature is kept below the design limit of 1600°C by this safety features. The HTTR does not need to consider the immediate accident management and to provide excess emergency safety system.

5. Plant Safety and Operational Performance

Various operational tests have been conducted to confirm the plant safety and operational:

a) Pre-operational test

Pre-operational test operation of the reactor cooling system was performed from May 1996 to March 1998. At the stage of the pre-operational test without nuclear heating the helium gas was heated by the gas circulators up to about 200°C at 2 MPa. Plant control systems were also fully checked. During the pre-operational tests, several improvements in the system were made in terms of securing its safety margin and easy operation. Their performance was finally confirmed in July 1999 after completing the actual fuel loading.

b) Start-up physics test

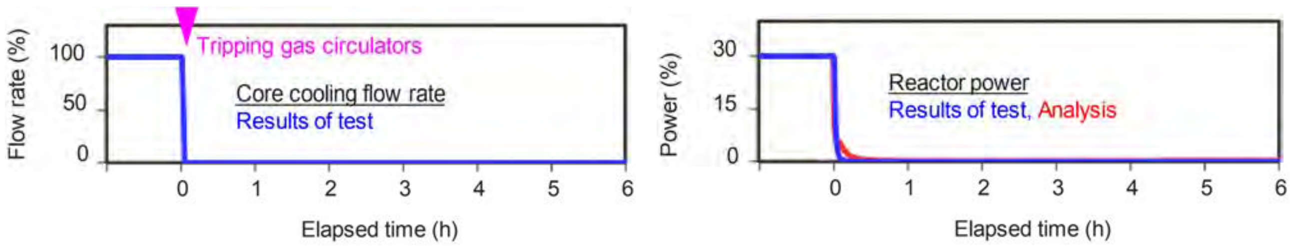
Fuel loading to the reactor started in July 1998, and the first criticality was attained on November 10th, 1998. The fuel blocks were column-wise loaded from the outer fuel columns to the inner. The first criticality was achieved successfully with 19 fuel columns loading. After that, the other inner fuel columns were loaded and the full core criticality was achieved by December 1998. In the course of fuel loading, low power physics tests were also carried out for the 21, 24, and 27 fuel columns loaded core. These tests provide useful data for designing future annular cores of advanced HTGR.

c) Rise to power test

Rise to power tests were started in September 1999 when the reactor power was increased step-by-step to 10 MW(t), 20MW(t), and then finally to 30 MW(t). The 30 MW(t) full power and 850°C high reactor outlet coolant temperature were achieved in December 2001. Certificate of pre-operation test, that is, operation permit of the HTTR was issued in March 2002. The HTTR accomplished the maximum reactor outlet coolant temperature of 950°C in April 2004 in high temperature test operation. Operation permit for the high temperature test operation was issued in June 2004.

d) Safety demonstration test

The safety demonstration tests have been implemented from 2002 in order to confirm the excellent inherent safety of HTTR. In the first phase of the safety demonstration test, the control-rod withdrawal tests, the gas circulator tripping tests, etc. have been carried out demonstrating the safety of HTTR. From 2010, the second phase of safety demonstration tests, namely loss of forced cooling test (LOFC), was carried out. The LOFC test was initiated by tripping all three helium gas circulators of the HTTR while deactivating all reactor reactivity control systems to disallow reactor scram due to abnormal reduction of the primary coolant flow rate. The test results showed that the reactor power immediately decreased to almost zero and became stable as soon as the helium gas circulators were stopped as shown in the figure below.



e) Continuous operation

In order to demonstrate the long-term operation of the heat utilization system, the HTTR was conducted Rated/Parallel-loaded 30 days, and further 50 days, continuous operation with full power. This was the first long-term operation with a reactor outlet coolant temperature over 900°C. The continuous operation test confirmed that the reactor internal structures and the intermediate heat exchanger, which are the core technologies of the HTGR, operated properly as designed value. The intermediated heat exchanger could also transfer stable high-temperature heat from the primary to secondary helium coolant.

6. Instrumentation and Control Systems

Instrumentation and control systems consist of instrumentation, control, and safety protection systems. The instrumentation includes reactor and process instrumentations to provide important parameters such as control rod position, neutron flux, temperature, pressure, flow rate, etc. for operation, monitoring, and reactor protection. There are about four-thousand sensors in the HTTR, and the signals from the sensors are centralized by the plant computer. The control systems comprise the operation mode selector, reactor power control system, and plant control system. The safety protection systems consist of the reactor protection system and engineered safety features actuating system to ensure the integrity of the core and prevent the fission products release.

7. Plant Layout Arrangement

The plant area is 200 m × 300 m in size, including the reactor building, cooling towers, exhaust stack, laboratory building, and other auxiliary facilities. The reactor building is located in the centre of the plant. The exhaust stack is on the north side of the reactor building to ventilate the air from the reactor building to the atmosphere. The laboratory building and the development building are on the west of the reactor building.

8. Design and Licensing Status

The HTTR construction started in 1991 with first criticality accomplished in 1998. Details of the operational achievements is given below. The HTTR has not been in operation since the great east Japan earthquake occurred in March 2011. On June 2020, JAEA received permission toward the restart of HTTR in conformity with new regulatory requirements by Nuclear Regulation Authority of Japan.

9. Development Milestones

1969 – 1984	Conceptual design (4 years); System integrity design (6 years) and Basic design (3 years)
1985 – 1990	Detail design (3 years); Application and permission of construction (1 year)
1991 – 1997	Construction
1998	First criticality
2001, 2002	Reactor outlet coolant temperature of 850ºC; Safety demonstration test
2004	Reactor outlet coolant temperature of 950ºC
2007; 2010	850ºC/30 days operation; 950ºC/50 days operation and Safety demonstration tests
2014	Conformity review on the New Regulatory Requirements start toward resumption of operation
2020	Permission toward the restart of HTTR

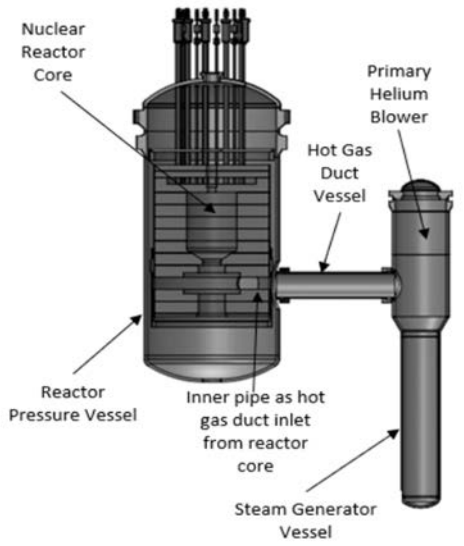
10. Future plans

Following the restart of HTTR, a number of activities are planned to be carried out including the safety demonstration tests in the OECD/NEA LOFC project, discussion on technology demonstration tests of heat utilization system consisting of helium gas turbine and hydrogen production facilities to be connected to the HTTR, operation test of fuel performance, and international cooperation and human-resource development utilizing the HTTR. As the HTTR can be used as a test bed for international cooperation, JAEA plans to launch new international projects based on the operation of the HTTR, and welcomes discussion with potential partners.

11. Reference

T. Nishihara et al., “Excellent Feature of Japanese HTGR Technologies”, JAEA- Technology 2018-004.

RDE/Micro-PeLUIt (BATAN, Indonesia)

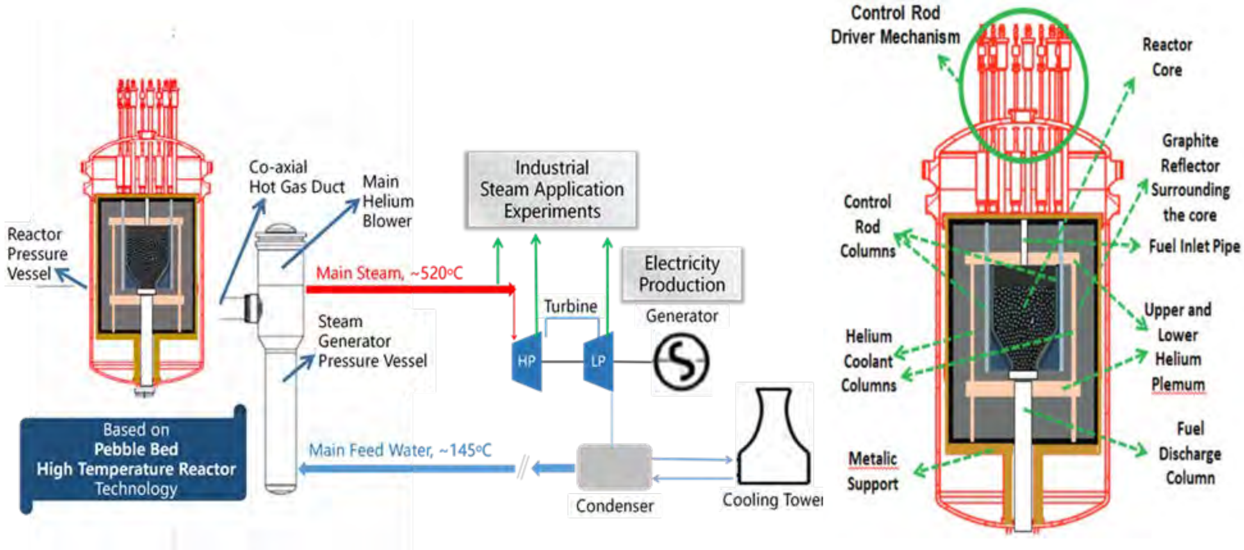


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	National Nuclear Energy Agency (BATAN), Indonesia
Reactor type	Pebble bed high temperature gas-cooled reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 / 3
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	3 / 6
Core Inlet/Outlet Coolant Temperature (°C)	250 / 750
Fuel type/assembly array	Spherical elements with coated particle fuel
Number of fuel assemblies in the core	27 000
Fuel enrichment (%)	17
Core Discharge Burnup (GWd/ton)	80
Refuelling Cycle (months)	On-line refuelling
Reactivity control mechanism	Control rod and small absorber sphere
Approach to safety systems	Combined active and passive
Design life (years)	40
Plant footprint (m^2)	~24 000
RPV height/diameter (m)	11.1 / 4.2 (inner)
RPV weight (metric ton)	180
Seismic Design (SSE)	0.26g
Fuel cycle requirements / Approach	LEU, open cycle, spent fuel intermediate storage at plant
Distinguishing features	Inherent safety, no need for offsite emergency measures
Design status	Site license issued in 2017. Detail design development is in progress as part of Design Approval phase.

1. Introduction

As one of the national programmes to support the National Medium Term Development Plant (RPJMN) for 2015-2019, BATAN started the *Reaktor Daya Eksperimental* (RDE) development programme in 2015. The main goal of the RDE Development Programme is to build the national capability to be a nuclear reactor technology developer by mastering the design, construction project management, commissioning and operation of a nuclear power reactor. Furthermore, the nuclear reactor type selected for the programme should become the prototype design to be scaled up and commercialized to contribute in enhancing the national energy supply security. The Pebble Bed Reactor (PBR) type of High Temperature Gas-cooled Reactor (HTGR) was selected as the technology for the RDE Programme. A sound safety feature and flexible applications of the PBR technology are among the reasons of this decision. PBR has a high safety level shown by a small radioactive release to the environment on any probable accident.

As part of licensing procedure, BATAN already received the RDE Site Licensing from National Nuclear Regulatory Body (BAPETEN) in January 2017. The next step of the licensing, which is in progress, is the Design Approval in which BATAN needs to submit the detail design and safety analysis report of RDE to BAPETEN. The general scheme of the RDE system is shown in the left figure below. The main components of the nuclear island system are the reactor pressure vessel (RPV) and its internal, coaxial hot gas duct, main steam blower, and the steam generator pressure vessel. The rest of balance-of-plant system is common to other power plant such as coal power plant. While parts of the RPV-internals are shown in the right side of below figure such as the core and the graphite reflector surrounding the core.



RDE Schematic system (left) and main component of the RPV Internals.

2. Target Application

The RDE is an experimental power reactor for research, which basically focuses to demonstrate the operational and safety performance of a pebble bed high temperature gas-cooled reactor (PB-HTGR). The RDE Programme will demonstrates the capability of the reactor to survive even in the most severe accident scenarios such as depressurized loss of forced cooling (DLOFC). Cogeneration experiment is another priority of the RDE to utilize its high steam temperature output. For the initial stage, a small 3 MW(e) output of RDE is not for commercial application. The RDE and its experimental apparatus will be used as the prototype reactor to be scaled up to a larger power level for a commercial application in the future, provided that there is an interest from the government or industry.

However, as the communication with industry continues, in particular with PT. Pembangunan Perumahan (PT. PP) and PT. Rekadaya Elektrika Consult (Reconsult), a direct utilization of a small 3 MW(e) as electric source for many islands in Indonesia, replacing an expensive diesel fuel, was foreseen. A joint cooperation with national industries to study and develop the techno-economic and detail design of the reactor has started. A specific product identification ‘Micro-PeLUIt’ has been given for the follow-up RDE design to be utilized in the remote area or particular industrial complex. PeLUIt stands for *Pembangkit Listrik dan Uap-panas Industri* (Indonesian) that means Nuclear Power Plant for Cogeneration of Electricity and Industrial Process Heat. On the other hand, RDE is the design identification for the reactor to be sited in the PUSPIPTEK, Serpong.

3. Main Design Features

(a) Design Philosophy

The RDE/Micro-PeLUIt is designed based on an established pebble-bed reactor principle. The design employs the TRISO-based fuel which provide a sound fission product retention capability resulting in allowable release of radioactive material to the environment in any condition of the core including the most severe postulated accident. The core and reflector are dominantly composed by a graphite which gives a good heat transfer and neutronic features. High heat conductivity and capacity of the graphite improve the heat transfer characteristic of the design. While it helps to improve the thermal neutron spectrum of in the core due to its effective neutron thermalization capability. An inert He gas as the coolant avoids any chemical or physical reactions. The nuclear island is applying a typical side by side arrangement of the RPV and Steam Generator Vessel which connected by a hot gas duct vessel. 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 10 MW(t), the helium temperatures at the reactor core inlet / outlet are 250 / 700°C, and steam parameters is 6 MPa / 520°C at the steam turbine entrance. The nuclear island is coupled with a 3 MW(e) steam turbine.

(b) Reactor Core and Power Conversion Unit

The primary helium coolant works at the pressure of 3.0 MPa. The rated mass flow rate is 4.27 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 and top helium plenum 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 700ºC and then flows to the steam generator. The hot helium then transfers its energy to the 243ºC feed water in the steam generator to have a superheated steam of 520ºC at 6 MPa flowing to the turbine to generate a ~3MW(e).

(c) Fuel Characteristics

Fuel elements are spherical ones. Every fuel element contains 5 g of heavy metal. The equilibrium core has 17% enrichment of U_{235}. Uranium kernels of ~0.5 mm in diameter are coated by three layers of pyro-carbon and one layer of silicon carbon. Coated fuel particles are dispersed in matrix graphite with 5 cm in diameter. Surrounding the fuel containing graphite matrix is a 5 mm thick graphite layer.

(d) Fuel Handling System

The operation mode of RDE/Micro-PeLUIt adopts continuous fuel loading and discharging: the fuel elements are pneumatically lifted into the upper part of the reactor, drop into the reactor core using a single fuel loading tube, then move downward across the core and through a discharging tube at the core bottom. The fuels will pass one-by-one through the singulator. The geometry of discharged fuel elements is checked in the fail-fuel separator. Failed fuel with geometrical defects will be separated and diverted into the failed fuel cask, while the good ones will continue to the burn-up measurement facility. Fuel pebbles that already reached the burnup target will be collected in the spent fuel cask while the other will be redirected back into the core. In average, a single pebble fuel will pass the core 5 times to reach the average discharge burnup target of 80 MWd/kg.

(e) Reactor Pressure Vessel and Internals

The primary pressure envelope of RDE/Micro-PeLUIt consists of the reactor pressure vessel (RPV), the steam generator pressure vessel (SGPV) and the hot gas duct pressure vessel (HGDPV), which are housed in a concrete shielding cavity. The material for the RPV is selected based on ASME Section III. The RPV consists of vessel portion, closure head and nozzles. The RPV internals including the ceramic internal and metallic internal, also the control rod and control rod drive mechanism. The metallic internal include the core barrel with guides and suppors, lower structure with bottom plate and the top thermal shield. The ceramic internal include all the bottom, side, and top reflector also the outer carbon brick layer.

4. Safety Features

The RDE/Micro-PeLUIt incorporate the established safety features of PB-HTGR design as follows: (1) Maximum temperature of the fuel is below 1400°C in any condition, even in the most hypothetical accident; (2) With that maximum temperature, the TRISO-based fuels contains all fission products to ensure a non-hazardous released to environment. Passive control safety features by the low power density, a large negative temperature coefficient, low excess reactivity (due to on-line refuelling). Passive cooling safety features supported by the physical properties of the graphite which is the dominant material in the core also by the low diameter design of the core.

(a) Engineered Safety System Approach and Configuration

The RDE/Micro-PeLUIt employ a standard engineering safety system of the PB-HTGR. This engineered safety system function to localize, control, mitigate and terminate accidents and to maintain radiation exposure levels to the public below applicable limits. It applied the principle of redundancy, high reliability, diversity, and single failure principle. It includes the ventilated low-pressure containment, reactor cavity cooling system, safety shutdown and protection systems, primary loop isolation, secondary loop isolation, emergency steam generator drainage system, and main control room habitability system.

(b) Reactivity control

RDE is equipped with two (2) independent reactivity control or shutdown system, a control rod system and a small ball shutdown system. Control rods are used for shutdown, fine temperature adjustment and trimming. Each control rod can move in the side reflector columns independently. The small ball shutdown system is provided for cold and long-term shutdowns. The small ball shutdown elements are stored above the top thermal shield and fall under gravity into reflector columns (slotted holes) by demand. The passive control capability of the reactor is supported by its strong negative reactivity feedback and a low excess reactivity.

(c) Reactor Cooling Philosophy

In normal operation, the core is cooled by the helium coolant flowing to the reactor in the rate of 4.27 kg/s and inlet temperature of 250°C. It reaches 700°C after absorbing the heat from the reactor, then the heat is transferred to the secondary cycle in the steam generator. Under accident conditions, the cooling of the core depends on its passive cooling system. After shut-down the core decay heat is dissipated passively through the core structures to the RPV due to the sound thermal characteristic of the graphite. From the RPV, the heat will be taken by the reactor cavity cooling system (RCCS) which operate passively based on natural circulation. However, the main function of the RCCS is to protect the concrete of the cavity. Even if the RCCS fails, basically the fuel temperature can be maintained below the design limit.

(d) Containment Function

Containment capability of the RDE/micro-PeLUIt design is based on multi-barrier system. The TRISO-based fuel design, in particular the SiC layer act as the first barrier which maintain almost all of the fission product. From the previous HTR-fuel test, in particular the HTR-10 duel design which adopt in RDE, the fuel able to maintain its containment capability under the temperatures of 1620ºC which is not expected for any accident scenarios. The second barrier is the primary pressure boundary which consists of the pressure vessels of the primary components. The third barrier is the ventilated low-pressure containment as part of engineered safety feature of the design.

5. Plant Safety and Operational Performances

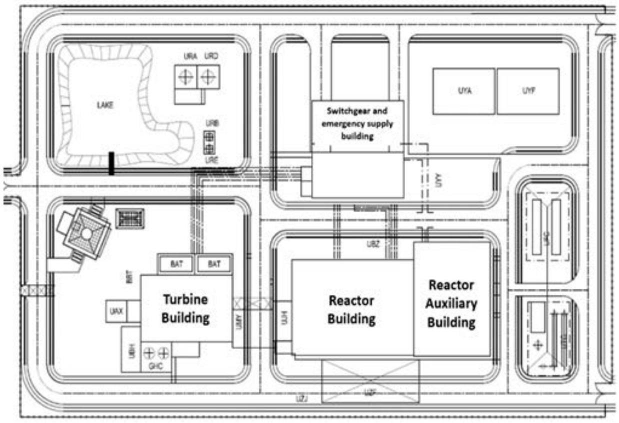
The RDE/Micro-PeLUIt is not yet constructed so no empirical safety and operational performances. However, it is expected that due to online refuelling feature a better availability factor can be expected compared with other power plants operating in a mode of periodic fuel loading. One of the priorities is to conduct experiments to evaluate plant operational safety performance using the RDE.

6. Instrumentation and Control Systems

In general, the instrumentation and control system of RDE/Micro-PeLUIt is similar to those of normal PWR plant. The Reactor Protection System (RPS) has the capability to measure important parameter related with the reactor safety including the neutron fluxes for the intermediate and power ranges, high and low Helium gas temperature, mass flow of the main helium coolant and feed water in the secondary system, and pressure in the primary and secondary system.

7. Plant Layout Arrangement

The nuclear island contains reactor building, reactor auxiliary building, as shown below. The steam turbo-generator is housed in the turbine building. The other plant auxiliary systems are also shown in figure. For the Micro-PeLUIt a more compact layout (maintaining the main components as in RDE) is under development



Plant Layout of the RDE.

8. Design and Licensing Status

The Site License was already issued by the Nuclear Energy Regulatory Agency (BAPETEN) in 2017. Before applying for a Construction License, the RDE needs to pass through the Design Approval phase which is now in progress since end of 2018. Detail design and Safety Analysis Report development are in progress of completion as part of the Design Approval requirement. On these activities, BATAN cooperate with national engineering companies, particularly PT. PP and the ReConsult.

9. Fuel Cycle Approach

In the adopted fuel cycle, the pebble fuels that already reach its burnup limit are collected in the spent fuel casks. These spent fuel casks are placed in the spent fuel storage room in the reactor building. The spent fuel storage room is equipped with the radiation and temperature monitoring and control system. The casks and the room are designed to avoid criticality of the spent fuels. The cask is designed so that it can be placed in a standard LWR transport cask and be transported if necessary. For RDE, its final storage destination will be at the nuclear waste facility of BATAN near the RDE site. In current plan, no further re-utilization or re-processing of these spent fuels.

10. Waste Management and Disposal Plan

The waste management for the RDE/Micro-PeLUIt, in particular for the liquid waste include the active liquid waste collection, low level and inactive liquid waste collection, the monitoring tanks for collecting liquid waste below the statutory limiting values for disposal, and the water extraction system for helium supporting systems. In the case of accident, the post-accident water separator is provided to collect and store the condensate. Disposal of active liquid waste is planned to be at the BATAN's nuclear waste processing center near the RDE site.

11. Development Milestones

2015	Launch of RDE Program as part of BATAN Strategic Plan and National Medium-term Development Plan
2015-2016	Development of RDE Conceptual Design and start of RDE Site Licensing Phase
2017-2018	Site License issued by BAPETEN; RDE Basic Design development started
2018-now	Review of RDE Design and RDE detail design development in cooperation with industry partners.

FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS

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
NSSS Operating Pressure (primary/secondary), MPa	Low pressure
Core Inlet/Outlet Coolant Temperature (^{o}C)	420 / 535
Fuel type/assembly array	Mixed uranium plutonium nitride
Number of fuel assemblies in the core	169
Fuel enrichment (%)	up to 14.5
Core Discharge Burnup (GWd/ton)	61.45
Refuelling Cycle (effective days)	900-1500
Reactivity control mechanism	Shim and automatic control rods ($\Delta\rho \approx 12.5\ \beta_{eff}$)
Approach to safety systems	Passive
Design life (years)	30
Plant footprint (m^2)	80 x 80
RPV height/diameter (m)	17.5 / 26
RPV weight (metric ton)	27 000
Seismic Design (SSE)	VII-MSK 64
Fuel cycle requirements / Approach	Closed fuel-cycle. It uses nitride of depleted U with Pu
Distinguishing features	High level of inherent safety due to natural properties of the lead, fuel, core and cooling design
Design status	Detailed design with potential start-up in 2026

1. Introduction

BREST-OD-300 is a lead cooled fast reactor fuelled with uranium plutonium mononitride (U-Pu)N 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. Main Design Features

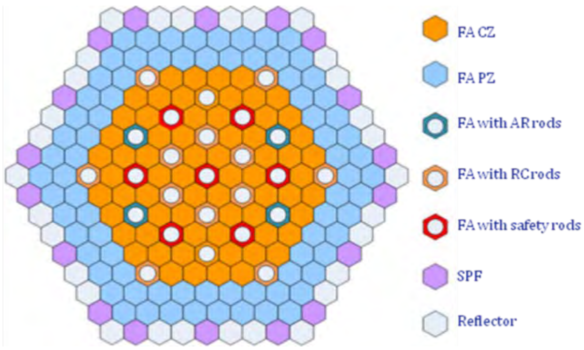
(a) Design Philosophy

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 (U-Pu)N 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 ($\Delta\rho \approx 12.5\ \beta$eff). Emergency protection rods are also provided ($\Delta\rho \approx 6.5\ \beta$eff).

(e) Reactor Vessel and Internals

An integral layout is used in the reactor facility to avoid coolant losses. The reactor vessel material is multilayer metal concrete; the lead coolant and the main components of the primary circuit are located in the reactor vessel. The central cavity accommodates the reactor core with a side reflector, the CPS rods, an in-reactor SFA storage and a reactor core barrel that separates the hot and cold lead flows. The four peripheral cavities (one for each loop) accommodate steam generators and reactor coolant pumps, heat exchangers of the emergency and normal cooldown systems, filters and other components. The cavities are hydraulically interconnected.

(f) 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.

(g) 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 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.

(h) Secondary System

The BREST-OD-300 reactor does not require secondary system.

(i) Steam Generator

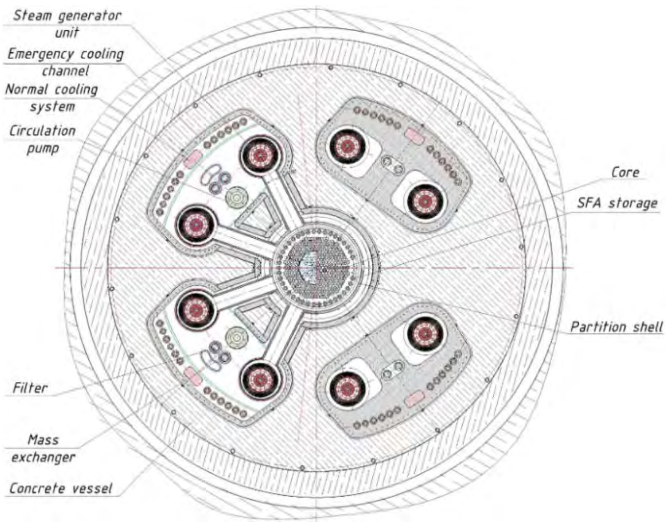
Steam generator is designed with single-walled twisted tubes.

(j) 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 and etc. Thanks to these features, 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) Engineered Safety System Approach and Configuration

Preferential use for safety: neutron-physical and physico-chemical properties of fuel, coolant, materials, as well as design solutions that allow to fully realize these properties. The lead coolant properties make it possible to implement in BREST-OD-300 fast reactor the following:

- in combination with application of (U-Pu)N fuel, complete breeding of fissile materials in the reactor core, which provides for a constant small reactivity margin preventing the disastrous effects of an uncontrolled power increase when implementing the reactivity margin because of equipment failures and personnel errors;
- to avoid the void effect of reactivity due to a high boiling point and high density of lead;
- to prevent coolant losses from the circuit in an event of the vessel damage due to high melting/solidification points of the coolant and the use of an integral layout of the reactor;
- to provide for high heat capacity of the coolant circuit which decreases a possibility of fuel damage;
- to allow for utilization of the high density of lead and its albedo properties for flattening the FA power distribution and the fuel pin temperatures respectively, as well as in the safety systems;
- to facilitate larger time lags of the transient processes in the circuit, which makes it possible to lower the requirements to the safety systems' rate of response.

(b) Decay Heat Removal System

The basic principles: no shutoff valves in the primary circuit (no circulation can be lost), a coolant circulation pattern with a free level difference (circulation is safely continued during loss of power), use of an emergency cooldown system with natural circulation and removal of heat to the atmospheric air.

(c) 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 –55ºC and 37ºC respectively. The system is passive.

(d) Containment System

The localizing function is performed by multilayer metal concrete vessel. Protection from external influences and threats is provided by the reactor building.

5. Plant Safety and Operational Performances

An innovative fast reactor BREST-OD-300 with inherent safety is being developed as a pilot and demonstration prototype for the basic commercial reactor facilities of future nuclear power with a closed nuclear fuel cycle with a view to the following:

- practical confirmation of the key design decisions used in the lead-cooled reactor facility operating in a closed nuclear fuel cycle and of the fundamental guidelines in the inherent safety concept on which these design decisions are based;
- phased justification of the reactor component service life for the creation of commercial nuclear power plants with lead-cooled reactors.

6. Instrumentation and Control Systems

In the reactor monitored temperatures, coolant levels, oxygen concentration, activity of the lead coolant and cover gas. Control and protection system is based on 2 channel and 3 sets.

7. Plant Layout 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.

8. Design and Licensing Status

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

- 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.

9. Fuel Cycle Approach

Establishment of CNFC for full utilization of energy potential of natural raw uranium. Mixed nitride fuel with high density and thermal conductivity allows to ensure full reproduction of fuel in the core (core reproduction ratio ~ 1.05) and compensation of reactivity at fuel burnout. The fuel type considered for the first core and the first partial fuel reloads of the BREST-OD-300 fast reactor is nitride of depleted uranium mixed with plutonium, whose composition corresponds to that of irradiated (spent) fuel from VVER’s following reprocessing and subsequent cooling for ~ 25 years. After completion of the initial stage the reactor operates in a closed fuel cycle. For the production of fuel, it uses own spent fuel reprocessed and purified from fission products.

10. Waste Management and Disposal Plan

Progressive approximation to radiation-equivalent (in relation to natural raw materials) RW disposal – at the operating stage after development of fuel with MA.

11. 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
2020	Expected start of construction
2026	First of a kind engineering demonstration plant

ARC-100 (ARC Nuclear Canada, Inc., Canada)

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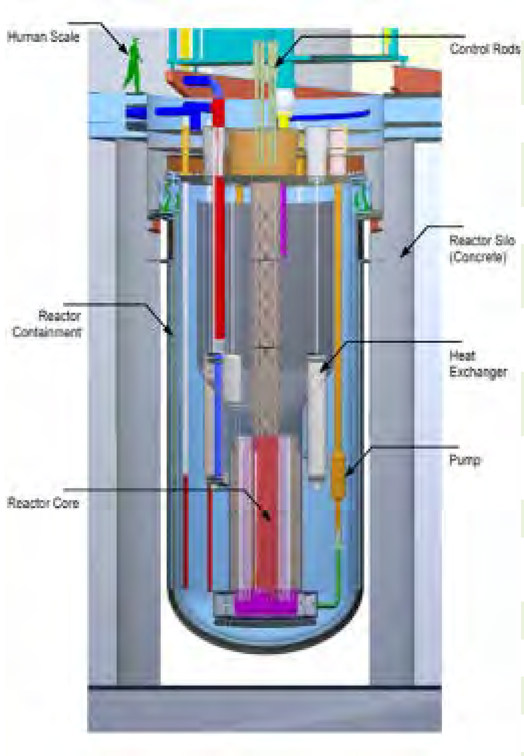

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ARC Nuclear Canada, Inc., Canada
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant/moderator	Sodium
Thermal/electrical capacity, MW(t)/MW(e)	286 / 100
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Non- pressurized
Core Inlet/Outlet Coolant Temperature (ºC)	355 / 510
Fuel type/assembly array	Metal fuel (U-Zr alloy) based on enriched uranium
Number of fuel assemblies in the core	99
Fuel enrichment (%)	Avg. 13.1
Core Discharge Burnup (GWd/ton)	77
Fuel Cycle (years)	20
Reactivity control mechanism	Control Rods
Approach to safety systems	Passive, diverse, redundant
Design life (years)	60
Plant footprint (m^2)	56 000
RPV height/diameter (m)	15.6 / 7.6
Distinguishing features	Inherent reactor safety with passive, diverse and redundant decay heat removal. Core lifetime of 20-years without refueling.
Design status	Conceptual design

1. Introduction

The ARC-100 is an advanced small modular reactor that utilizes proven prototype experience while integrating modern design improvements. It is a 100 MW(e) sodium-cooled, fast flux, pool-type reactor with metallic fuel that builds on the 30-year successful operation of the EBR-II reactor, built and operated by the Argonne National Laboratory in the U.S. The ARC-100 effectively addresses the four challenges which have limited the public acceptance and expansion of the nuclear industry. First, its 100 MW(e) electrical generation capacity is less than one-tenth the capacity of traditional nuclear power plants, and, consequently, its upfront cost will be affordable by a much broader range of customers in both the developed and developing worlds. Second, because its coolant is liquid sodium instead of water, its ‘fast’ neutrons have much more energy, giving it the capacity to be fuelled with and recycle its own used fuel. Third, the ARC-100 utilizes a metallic alloy of uranium instead of uranium oxide, which provides the foundation for its inherent, walk away safety. And fourth, the operator refuels this power plant only once every 20 years, rather than every 18-24 months which is typical of the light water reactors which dominate the current worldwide market. The long refueling cycle reduces operational costs and complexity, opening markets in the third world and many isolated off-grid applications. Replacement of the entire 20-year fuel cartridge and its removal by the vendor for recycling greatly reduces the risk of nuclear proliferation.

2. Target Application

The global energy industry is searching for an affordable, flexible, and mature utility-scale nuclear power solution to address the rapidly evolving energy market landscape and the geopolitics of environmental regulation. The ARC-100 offers has a unique solution to these challenges – a solution which can deliver:

- Breakthrough economics;
- Flexible operations and load following to complement intermittent renewable power sources;
- Technical maturity and demonstrated industrial reliability;
- Inherent safety performance;
- The ability to address the important issue of nuclear waste by recycling over and over its used fuel.

The ARC-100 is a low-risk, low-cost, clean energy solution that is ready for near-term development and deployment. It will initially target grid-scale electricity generation markets in the developed world. Also, its inherent safety and simplicity of operation make it ideally suited to satisfy electricity needs at remote locations like mine sites and the smaller grid markets in the developing world that can accommodate not much more than 100MW(e). It will also be targeted at industrial heat applications and water desalination.

3. Main Design Features

(a) Design Philosophy

The philosophy of the ARC-100 reactor is to rely on simple, passive safety features to achieve reactor safety under any normal operational occurrence or accident condition. The ARC-100 has adopted five traditional levels of safety for its defense in depth:

i. Minimize risk by the prevention of abnormal operation and failure by maximizing safety margins.

ii. Protection against abnormal operations and anticipated events via the large thermal inertia of the sodium pool.

iii. Protection against design basis accidents (DBA) through diverse and redundant systems.

iv. Control of severe plant conditions through designed passive and inherent safety characteristics of the facility.

v. Protection of the public health and safety in case of accidents by designing the inherent and passive safety characteristics such that operator intervention and external power are not required for plant survival. Additionally, the design goal of the plant is that the evacuation zone is limited to the site boundary.

(b) Nuclear Steam Supply System

The primary circuit of the ARC-100 is the coolant loop of the reactor core which is contained within the reactor vessel. Two Intermediate Heat Exchangers (IHX) serve as mean of transferring heat to the Intermediate Heat Transfer System (IHTS), both located within the reactor vessel. The IHX's penetrate the redan from the hot sodium pool and transfer heat to the cold sodium pool. Four submersible EM pumps provide sodium circulation within the reactor. The IHTS serves as the means of transferring the heat to the Steam Generator.

(c) Reactor Core

The ARC-100 core consists of driver assemblies successively surrounded by steel reflector assemblies and shield assemblies. The core is divided into inner, middle, and outer core zones to flatten the radial power distribution. The fuel assembly contains fuel pins, each of which provide a plenum to contain fission gases. The fuel is U-10%Zr binary metallic fuel with an average uranium enrichment of 13.1%. The maximum enrichment will be in compliance with IAEA requirements.

(d) Reactivity Control

The ARC-100 core employs two independent, safety grade, reactivity control systems. The Primary System consisting of six control rods is designed to have sufficient reactivity worth to bring the reactor from any operating condition to cold subcritical with the most reactive control assembly stuck at the full power operating position. Any operating condition includes an overpower condition together with a reactivity fault. The Primary System also serves to compensate for the reactivity effects of the fuel burnup and axial growth of the metal fuel. The reactivity associated with uncertainties in criticality and fissile loading is accommodated by the Primary Control System. The Secondary Control Rod System consisting of three control rods is designed to shut down the reactor from any operating condition to the cold shutdown, also with the most reactive assembly inoperative.

(e) Reactor Pressure Vessel and Internals

The Reactor Vessel contains the nuclear fuel and forms the coolant loop of the Primary Heat Transport System. The primary coolant boundary is completely enclosed within the Reactor Vessel shell and top plate which forms a pool of sodium. A cover gas of argon, held at pressure slightly higher than atmospheric pressure, resides above the sodium pool within the Reactor Vessel. The Primary System boundary consists of the Reactor Vessel, the reactor top plate, and the top plate mounted components with the principal component being the intermediate heat exchanger. The Guard Vessel surrounds the Reactor Vessel shell to serve as a leak jacket should the Reactor Vessel shell develop a leak. The core support structure uses a welded connection at the bottom head of the Reactor Vessel. Other than the core support connections and shipping restraints, the vessel has no attachments and no penetrations below the reactor top plate. Reduced number of penetrations below the reactor top plate and low operating pressure precludes any pipe ruptures. It is a key factor in the ability to keep the core continuously cooled for the entire spectrum of design basis events. The design lifetime of the reactor vessel, as well as the other components, is not less than 60 years.

(f) Reactor Coolant System

The primary circuit of the ARC-100 is the coolant loop of the reactor core which is contained within the reactor vessel (the reactor vessel is bounded by a guard vessel to serve as a leak jacket). The reactor employs sodium as the coolant and argon as a cover gas over the pool of sodium. Two Intermediate Heat Exchangers (IHX) serve as the method of transferring heat to the Intermediate Heat Transfer System (IHTS), both located within the reactor vessel. The IHX's penetrate the redan from the hot sodium pool and transfer heat to the cold pool. Four submersible EM pumps provide sodium circulation within the reactor.

(g) Secondary System

The secondary system is referred to as the Intermediate Heat Transport System (IHTS). The IHTS is the fluid system for transporting reactor heat between the IHX and the steam generator. It consists of two piping loops between the IHX, which resides in the Reactor Vessel, and the steam generator. Each piping loop includes an EM pump and permanent magnet flowmeters in the cold leg. The system also includes instrumentation for detecting steam generator tube leaks and a rupture disc driven pressure relief line for overpressure protection for the steam generator shell, intermediate piping and IHXs. The steam generator, the sodium dump valve, the Intermediate Sodium Processing System are located in the Steam Generator Building (SGB).

(h) Steam Generator

The steam generator is a helical coil, single wall tube, vertically oriented sodium-to-water counter-flow shell-and-tube exchanger. The steam generator provides the interface for where the sodium flowing in from the IHTS heats the water to generate superheated steam for the steam turbine plant. Sodium is distributed through the shell side of the steam generator while the water flows through the helical coil tube bundles. The steam generator includes a cover gas space in the upper head of the steam generator which accommodates sodium level changes due to intermediate sodium thermal expansion and pump transients.

(i) Pressurizer

The ARC-100 is a sodium cooled fast reactor that operates at atmospheric pressure and therefore does not need a pressurizer.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

In the design, the reactor's inherent reactivity feedbacks have been leveraged for safety and economics in two ways:

- The systems provide the basis for the reactor to passively self-regulate its power production to match the heat demand from the power conversion cycle without moving control rods (passive load following).
- The systems provide 'defense-in-depth' protection to arrest accident progressions before the reactor reaches unsafe conditions, even if the control and safety rods fail to scram.

(b) Decay Heat Removal System

The Balance of Plant steam turbine generator system is relied upon for normal shutdown heat removal. The ARC-100 emergency Heat Removal systems consist of the following:

- Direct Reactor Auxiliary Cooling System (DRACS);
- Reactor Vessel Auxiliary (Air) Cooling System (RVACS).

The RVACS always operates in a passive state, RVACS removes the reactor's decay heat through the Reactor Vessel and Guard Vessel walls by radiation and convection to naturally circulating air outside the Guard Vessel without exceeding structural temperature limits. The DRACS is composed of three units operating in a natural convection mode. Heat exchanger loops using NaK as an intermediate transfer heat from the cold sodium pool to air heat exchangers in which the hot air vents to the atmosphere. DRACS can be either passive or active using forced air convection

(c) Emergency Core Cooling System

Being a pool-type sodium cooled fast reactor which operates at atmospheric pressure, a loss of coolant accident is not a credible scenario for the ARC-100. In the unlikely event of a breach of the reactor vessel, the guard vessel would ensure adequate core coverage and support natural circulation. Therefore, there is no need to have light water reactor-like emergency core cooling systems.

(d) Containment System

The ARC-100 containment system is a Low-Leakage containment type, where the reactor vessel is designed to operate at near atmospheric pressure. Containment leak rate will be determined based on the dose consequences analysis results in future detailed analyses. Damage to the ARC-100 core does not directly relate to radioactive releases as the high chemical compatibility between the fission products and the sodium coolant trap radionuclides. The core itself is isolated from the secondary side using an intermediate heat transfer system, which itself limits if not prevents the propagation of nuclides via being pressurized above the primary sodium loop.

A Cover Gas System is used to remove radionuclides from the cover gas region. Radiation monitors are installed in the head access area and cover gas service vault to detect any gas leakage. The slightly pressurized argon cover gas is circulated with a filtering stage to eliminate sodium vapor, aerosols, and any other impurities. This monitoring system is also used to survey the reactor cover gas to check for elevated fission gas levels that could indicate fuel failures, and to detect fuel handling and criticality accidents. The pressurization is used to ensure that no sodium will be released into the environment if a leak occurs. Argon that is treated is reintroduced to reduce consumption.

5. Plant Safety and Operational Performances

The ARC-100 will require only minimal active involvement of the plant operators. The operator's role will be to monitor plant behavior and transient response to ensure that it is within the specific design parameters. The core design features include a low burn-up reactivity swing which reduces the need for frequent control rod motion and the inherent load following characteristics of the reactor support simplified load following operation.

6. Instrumentation and Control Systems

The fail-safe safety-related shutdown Distributed Control and Information System (DCIS) is a four-division control and monitoring system design, each with separate and independent power supply electrical systems. Divisions are used to support automatic shutdown of the reactor and decay heat removal via DRACS and RVACS. The system is designed to be able to operate with one division continuously out of service when a design basis event occurs, and the plant safety functions will still be achieved with an additional random failure. Additionally, the DCIS operates at low voltage as the fail-safe shutdown systems are designed to operate without electricity.

7. Plant Layout Arrangement

(a) Reactor Building

The Reactor Building is a box-shaped shear wall building made of reinforced concrete floors and walls. Floor slabs can also be composite structures. Roof trusses and their supporting columns are made of structural steel. Constructed integrally with the Reactor Building is the reinforced concrete silo that surrounds the Primary Reactor System. The Reactor Building Structure houses the Primary Reactor System, reactor support and safety systems, essential power supplies and equipment, and refueling area. The refueling floor of the Reactor Building Structure includes the refueling and fuel handling systems and the overhead crane. The design pressure and temperature of the Reactor Building will be established from maximum calculated pressures and temperatures resulting from postulated design basis events including sodium fires. The reactor building and the primary systems including the concrete silo will have a design life of at least 60 years.

(b) Balance of Plant

The main turbine is a reheat steam cycle that uses a single shaft two-casing steam turbine with a single flow high pressure turbine and combined intermediate and low-pressure turbine. A side exhaust has been chosen to minimize Turbine Island cost. The Condensate and Feedwater System collects water from the main turbine and auxiliaries after available thermal energy in the water has been extracted, conditions it, and returns it to the steam generator at design temperature and pressure.

The Turbine Auxiliary Steam System transports the steam produced by the steam generator to its point of use and extracts its available thermal energy via auxiliary equipment. Included within this system are the systems needed to support the operation of the main turbine generator. The turbine generator auxiliary systems provide supportive services to the turbine generator via cooling, sealing, lubricating, and control functions to sustain the operation and assure the maximum efficiency of the turbine generator.

8. Development Milestones

2020	Conceptual Design complete
2022	Detailed Design complete
2025	First Concrete for first unit
2028	First Unit in service

4S (Toshiba Corporation, Japan)

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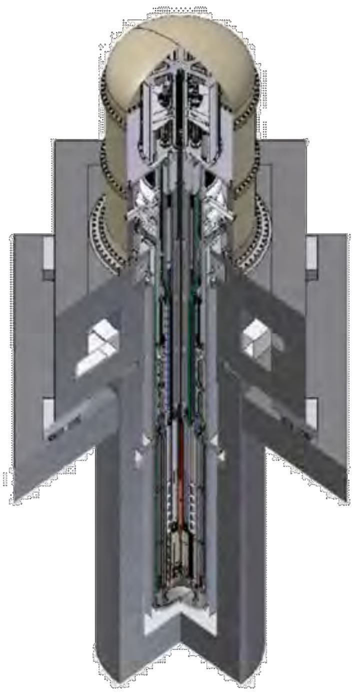


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Toshiba Energy Systems & Solutions Corporation, Japan
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant/moderator	Sodium
Thermal/electrical capacity, MW(t)/MW(e)	30 / 10
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Non- pressurized
Core Inlet/Outlet Coolant Temperature (°C)	355 / 510
Fuel type/assembly array	Metal fuel (U-Zr alloy) based on enriched uranium
Number of fuel assemblies in the core	18
Fuel enrichment (%)	< 20
Core Discharge Burnup (GWd/ton)	34
Refuelling Cycle (months)	N/A
Reactivity control mechanism	Axially movable reflectors / fixed absorber
Approach to safety systems	Hybrid (passive + active)
Design life (years)	60
Plant footprint (m^2)	157 000
RPV height/diameter (m)	24 / 3.5
RPV weight (metric ton)	-
Seismic Design (SSE)	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 a non-breeder fast reactor. Blanket fuel, usually consisting of depleted uranium located around the core to absorb leakage neutrons to achieve breeding of fissile materials, is not is not adopted for the 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 168 000 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 15 000 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. Main 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 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 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 abnormal operation or 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

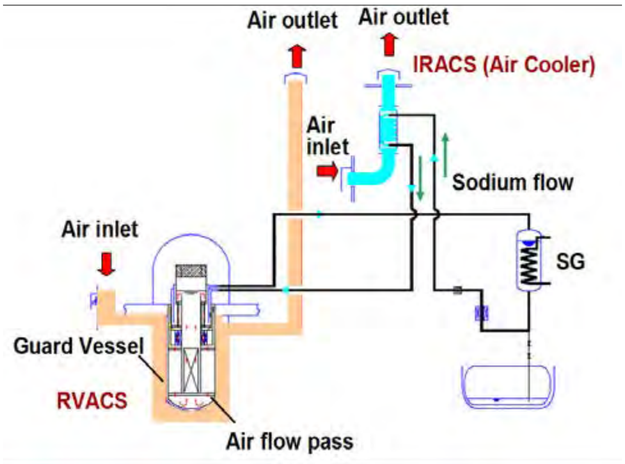
The 4S adopts a cylindrical/spherical containment system. The containment system consists of the GV and the top dome, which covers the upper region of the RV, a shielding plug and the equipment located on the shielding plug. The GV provides the second boundary for the primary sodium at the outer side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

5. Plant Safety and Operational Performances

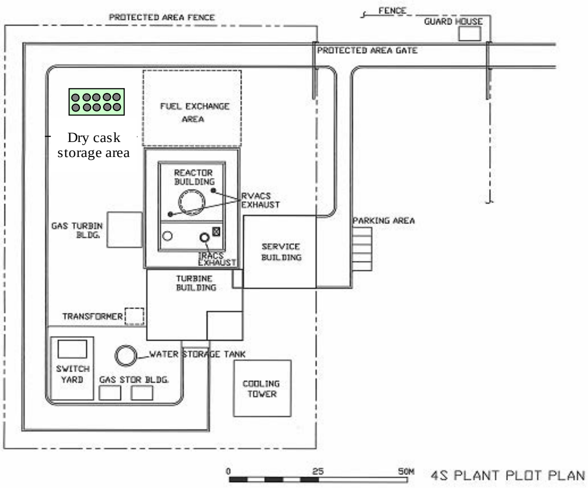
The 4S is designed to operate safely without active involvement of the plant operators. The design features to support such operation include: (1) burn-up reactivity swing automatically compensated by the fine motion reflectors, (2) no need in reloading and shuffling of fuel in the course of 30 years for the 10 MW(e)-4S, (3) reduction in maintenance requirements achieved by adopting static devices and (4) 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 systems. The safety-related systems include 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 Layout 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 Energy Systems & Solutions continues to look for 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

MicroURANUS (UNIST, Republic of Korea)

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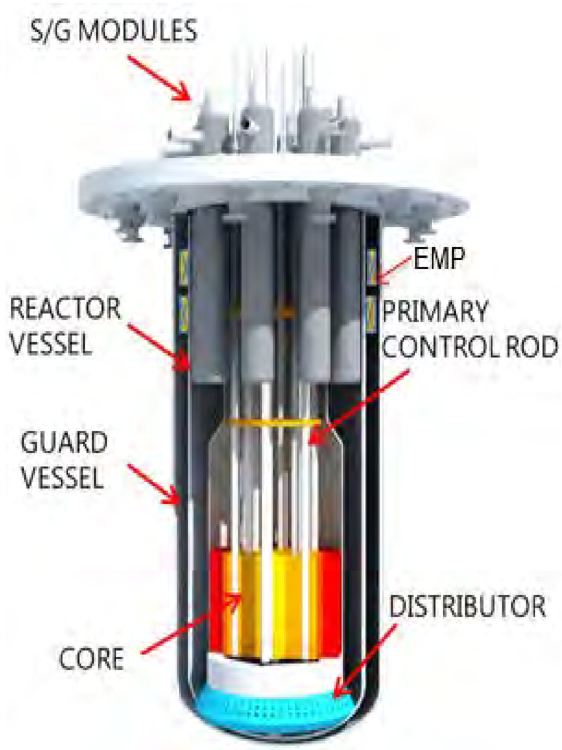


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Ulsan National Institute of Science and Technology (UNIST), Republic of Korea
Reactor type	Lead-bismuth cooled Reactor
Coolant/moderator	Pb-Bi (45-55% Wt.%) eutectic alloy
Thermal/electrical capacity, MW(t)/MW(e)	60 / 20
Primary circulation	Electromagnetic Pump (EMP)
NSSS Operating Pressure (primary/secondary), MPa	0.1 / 8~15
Core Inlet/Outlet Coolant Temperature (°C)	250 / 350
Fuel type/assembly array	UO_2 / Hexagonal
Number of fuel assemblies in the core	85
Fuel enrichment (%)	3 radial zones; 8, 10, 12
Core Discharge Burnup (GWd/ton)	60
Refuelling Cycle (full power years)	No refuelling (30)
Reactivity control mechanism	Control rods and shutdown rod
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	N/A
RPV height/diameter (m)	8.0 / 2.0
RPV weight (metric ton)	200
Seismic Design (SSE)	3D shock isolation system
Fuel cycle requirements / Approach	No HLW/PyroGreen Recycling
Distinguishing features	Ex-vessel emergency water cooling system
Design status	Pre-conceptual design

1. Introduction

Based on developments of Lead-Bismuth-Eutectic (LBE)-cooled transmutation reactors such as PEACER, PASCAR and URANUS, an LBE-cooled micro reactor using UO_2 fuel rods has been developed as MicroURANUS (Universal, Robust, Accident-forgiving, No proliferating and Ultra-lasting Sustainer) with a nominal power rating of 20 MW(e) from 60 MW(t). The MicroURANUS is designed to be compact and completely sealed after factory-fuelled and tested, prior to hot-transportation to installation sites. No refuelling is needed throughout 40-year design life and the encapsulated reactor module will be shipped to recycling facilities so that no spent nuclear fuels or high-level wastes will be left to the owners. It can operate either in dynamic environments of merchant ships or in terrestrial settings by utilizing both natural and forced circulation of chemically inert LBE liquid. For the reliability of system structure components for the design lifetime, necessary design margins are included.

2. Target Application

MicroURANUS will be deployed primarily for the propulsion of merchant ships and for the offshore electricity generation, with zero carbon footprint. MicroURANUS can also be used to replace land-based coal plants and to make steep load-follow operations in symbiosis with renewable energy sources.

3. Main Design Features

(a) Design Philosophy

Safety in marine environments is assured by eliminating chances of radioactivity release accidents and minimizing the emergency planning zone radius to within engineered power block boundaries. Even in the worst hypothetical accidents, the radioactive core materials will be contained by frozen LBE within the reactor vessel. Safeguards and physical protection are assured by weld-encapsulated reactor vessel without any provisions for fuel removal from engineered power block boundaries. It is only after decommissioning when the encapsulated reactor module containing spent nuclear fuels can be removed and transported to globally accepted recycling centres where all high-level waste will be decontaminated to intermediate level wastes for safe and secure geological disposal without uncertainty. Economy of a MicroURANUS power block will be far superior to current engines of merchant ships when manufacturing infrastructure is established.

(b) Nuclear Steam Supply System

The nuclear steam supply system consists of an integral reactor module that includes a reactor core, eight steam generators, control rod clusters, a reactor barrel, a lower plenum and a reactor head a reactor vessel. LBE coolant circulates upward inside the reactor barrel including the reactor core until it meets steam generator inlet where the flow direction changes to downward by entering stream generator tubes that lead to annular downcomer formed between the reactor barrel and reactor vessel. The flow direction changes to upward after passing through a circular flow distributor unit located at the reactor lower plenum.

(c) Reactor Core

MicroURANUS has evolved from URANUS-40 design with hexagonal-lattice using LEU oxide fuels. The total 85 fuel assemblies, 12 control assemblies and two ultimate shutdown assemblies. Three enrichment zones are used to reduce the radial power peaking and to increase internal breeding gain correspondingly small reactivity swing. The fissile enrichment adopted for the inner enrichment zone is lower than that of the outer fuel zone. The MicroURANUS core has an active height of 1.55 m and equivalent diameter of 1.6 m that generates 60 MW(t) with the average coolant temperature rise of 100ºC from the core inlet to the core exit.

(d) Reactivity Control

The MicroURANUS has a very small reactivity swing over thirty effective full power years, requiring almost no control rod movements if operated steadily at full power. Reactivity can be controlled by using 12 assemblies for versatile navigation of merchant ships. Rapid power increases can be tolerated by using PCI-resistant fuels. To meet regulatory requirements for the reactor module transportation before and after commercial operations, additional reactivity suppression is made by using the ultimate shutdown assembly.

(e) Reactor Pressure Vessel and Internals

The reactor vessel of MicroURANUS is approximately 8.0 m in height and 2.0 m in diameter. The reactor vessel is encased in the guard vessel with an annular space used for EMP installation, in-service inspections and maintenance. The reactor barrel is a monolithic shell that provides both structural supports for reactor core and internals while serving as LBE flow divider, neutron reflector and thermal insulator.

(f) Reactor Coolant System

LBE reactor coolant flows up through the reactor core and flows down through steam generator tubes by hydraulic head generated by both electromagnetic pump (EMP) and natural circulation. The reactor coolant operates at atmospheric pressure by taking advantage of its high boiling point (1670ºC). Dissolved oxygen concentration in LBE coolant is administered by injecting oxygen into the upper plenum in order to keep all structural materials stably passivated against corrosion.

(g) Secondary System

Currently a comparative study is in progress for the secondary systems, between traditional superheated steam cycle and supercritical CO2 cycle. Criteria for the down-selection process include technical maturity, transient responses, system footprint as well as maintainability.

(h) Steam Generator

Eight units of vertical once-through steam generators employ double-wall tubes. LBE flows downward on tube-side while the secondary working fluid flows upward on shell-side. Inter-tube space is filled with thermocouples, helium gas for leak detection and thermal conductors. By using ductile austenitic tube materials and compressive stresses due to higher shell-side pressure, leak-before-break (LBB) can be guaranteed.

4. Safety Features

The MicroURANUS design has been embedded with passive safety features by utilizing high capacity of natural circulation and superior heat conduction as well as high boiling point of LBE. In contrast with water, LBE coolant has excellent retention capability for safety-critical volatile radioactive species including iodine and caesium. LBE solidification by ex-vessel emergency water cooling can assure long-term isolation of radioactivity within the reactor coolant system, in the event of hypothetical accidents caused by both internal events and externalities, including flooding, collision, air crash, explosion, capsizing. Safety features of MicroURANUS is designed based on the defence-in-depth principle.

(a) Engineered Safety System Approach and Configuration

Reactor scram systems can be actively actuated with independent emergency power supplies. During normal shutdown and reactor trip conditions, decay heat is removed by the reactor vessel auxiliary cooling system (RVACS) that circulates coolant to transport heat from the reactor outer wall to external coolers. Under accident conditions, a dedicated ex-vessel emergency water cooling system is passively activated to remove any heat beyond the capacity of RVACS. If the reactor vessel is heated above limits, the annular space between reactor vessel and guard vessel is flooded with shielding water through fusible valves. Steam produced from the water-cooling of the external surface of reactor vessel goes to dedicated coolers where the steam is condensed into water before returning to the shield water tank by gravity.

(b) Decay Heat Removal System

During normal operation and transient conditions, decay heat is removed by the RVACS that circulates coolant between cooling coils on the reactor exterior to an external heat exchanger.

(c) Emergency Core Cooling System

Under accident conditions, a dedicated ex-vessel emergency water cooling system is passively activated to enhance cooling power beyond the capacity of RVACS. If the reactor vessel is heated above limits, the annular space between reactor vessel and guard vessel is flooded with shielding water through fusible valves. Steam produced from the water-cooling of the external surface of reactor vessel goes to dedicated coolers where the steam is condensed into water before returning to the shield water tank by gravity. The ex-vessel water cooling will gradually freeze LBE coolant inward so that radioactive materials can be trapped inside the reactor vessel.

(d) Containment System

The guard vessel and its gas-tight cover forms the primary containment system. The secondary gas-tight containment system encloses shield water tank, auxiliary systems for inspection and maintenance, shield structure as well as 3D shock isolation systems. This system is physically protected by external shield from external attacks including air strikes.

5. Plant Safety and Operational Performances

The plant safety and operational performance will be assessed by the end of 2022.

6. Instrumentation and Control Systems

Analogue/digital hybrid instrumentation and control systems will be designed by the end of 2022.

7. Plant Layout Arrangement

The reactor building and the balance of plant will be designed upon the down-selection of the secondary cycle.

8. Design and Licensing Status

A conceptual design of MicroURANUS is to be completed in 2022. Currently, the activities for Front End Engineering Design (FEED) are carried out for design, optimization, modelling and experimental validations. License application is expected in 2024.

9. Fuel Cycle Approach

At the end of life, LEU oxide fuel used for 40-year life will be remain in the reactor for about three years for decay heat cooling, prior to decommissioning. The encapsulated reactor module containing all spent nuclear fuels will be shipped to globally accepted recycling centres, either by land or water. Recycled TRU will be used to fabricate oxide fuels for the next generation MicroURANUS.

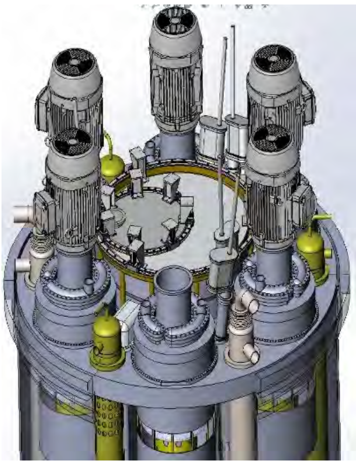
10. Design and Licensing Status

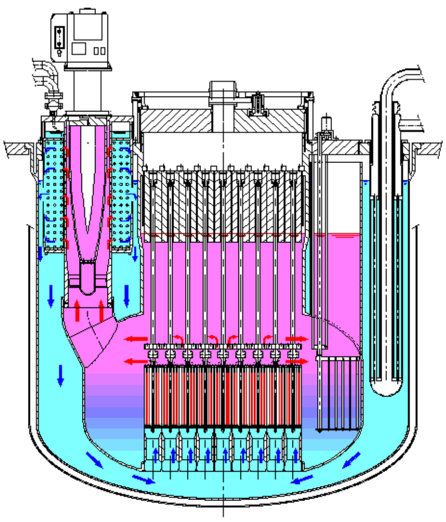
All high-level waste from recycling of MicroURANUS spent nuclear fuels will be further decontaminated using advanced partitioning technology including PyroGreen to intermediate level wastes for safe and secure geological disposal without long-term uncertainty. Limited quantities of low and intermediate level wastes produced during operation and from decommissioning can be readily disposed of using available repositories.

11. Development Milestones

2023-2024	Engineering design and license application
2025~2027	Nuclear pilot plant licensing and construction
2028~2030	Commissioning of merchant ship powered by MicroURANUS

LFR-AS-200 (HNE, Luxembourg)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Hydromine Nuclear Energy S.à r.l. (HNE), Luxembourg
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant/moderator	Lead / none
Thermal/electrical capacity, MW(t)/MW(e)	480 / 200 for prototype
Primary circulation	Forced convection (6 pumps)
NSSS Operating Pressure (primary/secondary), MPa	0.01 / 18
Core Inlet/Outlet Coolant Temperature (°C)	420 / 530
Fuel type/assembly array	MOX, hexagonal
Number of fuel assemblies in the core	61
Fuel enrichment (%)	19 max. / 23.2 in Pu
Core Discharge Burnup (GWd/ton)	100
Refuelling Cycle (months)	16
Reactivity control mechanism	Rods and B_4C solid boron absorber
Approach to safety systems	Hybrid active and passive
Design life (years)	60
Plant footprint (m^2)	1100
RPV height/diameter (m)	6.2 / 6
RPV weight (metric ton)	42
Seismic Design (SSE)	0.3g
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 is an innovative reactor cooled by molten lead; AS stands for Amphora-Shaped, referring to the shape of the inner vessel and 200 is the electrical power in MW. The embodied innovations exploit the lead properties and enhance the potential for future deployment, owing to plant simplification and compactness, while behaving passively safe; it is quite distinct from any previous LFR designs.

2. Target Application

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 cost per kWh. Market application is energy production with use of stockpiled Pu and perspective recycle of minor actinides without burden of long-lived transuranics in the waste. The breeding ratio is 0.9 without blanket assemblies and can be reduced with core design adaptation where required.

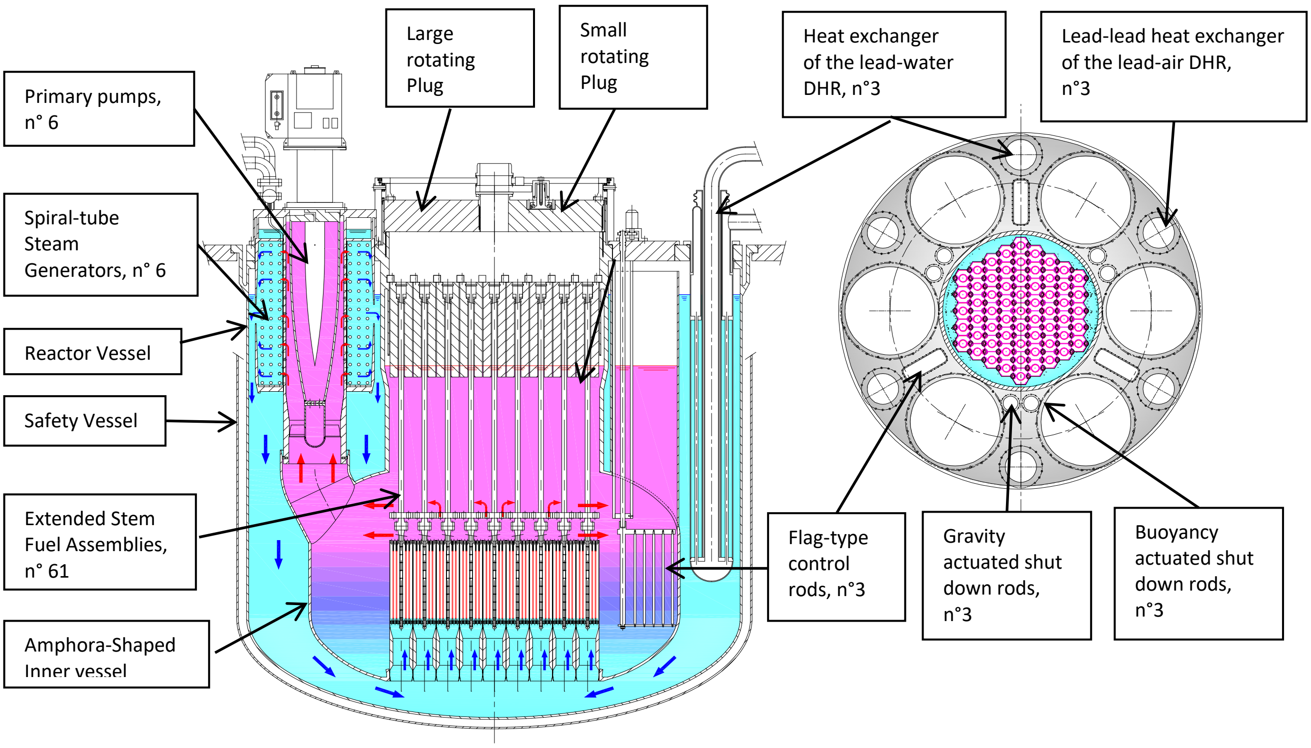
3. Main Design Features

(a) Design Philosophy

The LFR-AS-200 is a pool-type fast reactor. Main primary components are six innovative spiral-tube steam generators (STSG), six mechanical pumps, flag-type control rods and three + three dip coolers belonging to two diverse, redundant decay heat removal systems, fuel assemblies with stem extended above the lead free level and hung by their heads. There is no need of in-vessel refueling machine nor of intermediate loops. The risk of important primary system pressurization, in case of the steam generator tube rupture accident, is deterministically eliminated by several, special provisions: among them, water and steam collectors located outside the RV and short STSG partially raised above the lead-free level of the cold collector.

(b) Nuclear Steam Supply System

The LFR-AS-200 is a pool type reactor, which means that all the primary components are installed in the Reactor Vessel. Among the key-components are: Core, six Spiral-Tube Steam Generators (STSG), six Recirculation Pumps, three + three Dip Coolers of the Decay Heat Removal Systems (DHR), and the Amphora-Shaped Inner Vessel (ASIV).



Scheme of the LFR-AS-200.

(c) Reactor Core Design Approach

The core consists of 61 wrapped, hexagonal FAs, each containing fuel pins laid out on triangular pitch. The active region, 850 mm tall, is made by a stack of mixed uranium-plutonium oxide (MOX) fuel pellets with three different MOX blends:

- all the fuel pellets in the outermost 24 FAs have 23.2 wt.% plutonium content;
- the fuel pellets of the remaining 37 internal FAs are grouped in three axial sections: the central one contains 14.6 wt.% of plutonium, the two equal external sections have a plutonium content of 20.4 wt.%.

The aimed burnup is achieved with 2400 EFPD fuel residence time in the core, corresponding to about 80 months. In order to reduce the reactivity swing during burnup, the management of the fuel is based on a 5-batch strategy that is every 16 months, one fifth of the fuel is discharged and replaced with fresh fuel.

(d) Reactivity Control

In the LFR-AS-200, core reactivity is controlled by ex-core rods, installed in the lead pool between the core and the ASIV. The main conditions against which the LFR-AS-200 is challenged are those determined by an ULOOP accident. Such conditions may result from the complete loss of power to the plant and the contextual failure of the RPS to actuate SCRAM: in practice, the simulated accident is alike the one occurred at the Fukushima-Daiichi nuclear power plant, but further complicated by the unsuccessful shutdown of the core.

(e) Reactor Vessel and Internals

As an integrated reactor, all the primary components are installed in the Reactor Vessel, among them the key-components: the Amphora-Shaped Inner Vessel (ASIV), the core, the STSG, the Dip Coolers of the Decay Heat Removal Systems (DHR), and the Recirculation Pumps. The SS316LN stainless-steel is adopted as the reactor vessel material. For reactor internals and fuel cladding, new steels and/or protective coating are necessary.

(f) Reactor Coolant System

The LFR-AS-200 is a pool-type reactor that uses forced convection by 6 pumps for the lead coolant primary circulation. The vertical axial-flow pump is integrated inside the SG; 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.

(g) Secondary System

The secondary system of LFR-AS-200 is based on the Rankine cycle with superheated steam. The only peculiarity is related to the use of lead as the coolant in the primary system. To avoid lead freezing inside the SG, a prudential high SG inlet temperature is set to 340ºC; this temperature is higher than the melting point of Pb, at 327ºC. The condenser pressure is assumed at 6 kPa. The turbine is made of one high pressure stage and two low pressure stages, with a deaerator fed by steam from the outlet of the high-pressure stage. The turbo-generator set operates at 3000 rpm. A hot water storage is provided in order to reduce the amount of steam bled from the low-pressure turbine and allow a temporary operation at 110% Pn.

(h) Steam Generator

The Spiral-Tube Steam Generator (STSG) is an innovative key-component. The STSG can be considered a simplification of the well-known helical-tube steam generator.

4. Safety Features

The research and development program of the LFR-AS-200 is guided by a GIF IV Technology Roadmap document which identified three specific safety goals for Generation IV systems 'to be used to stimulate the search for innovative nuclear energy systems and to motivate and guide the R&D on Generation IV systems':

- Generation IV nuclear energy systems operations will excel in safety and reliability.
- Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
- Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

Lead has excellent cooling properties and its nuclear properties (i.e., its low tendency to absorb neutrons or to slow them down) enable it to readily sustain the high neutron energies needed in a fast reactor, while offering the reactor designer great flexibility. Lead has a very high boiling point, namely 1737°C. As a result, the problem of coolant boiling is for all practical purposes eliminated.

(a) Engineered Safety System Approach and Configuration

Combined active and passive safety systems are adopted for the LFR-AS-200 comprises reactor shutdown system, decay heat removal system, containment system, containment isolation system, DC power with 72-hour capacity and fileting for severe accident containment provision.

(b) Decay Heat Removal System

For conservativeness, only one of the two Decay Heat Removal (DHR) systems is assumed to work, but with only two out of the three loops in operation (single failure assumption). 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.

(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 an 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 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. 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

Signals from in-core instrumentation could be made available to inspectors to detect anomalies resulting from design modifications. Moreover in-core instrumentation will remain mostly operational even during refuelling.

7. Plant Layout 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.

(a) Reactor Building

The reference plant is constituted by one or more modules, each made of by two reactors with the pool for spent fuel installed in the same compact reactor building protected from aircraft impact.

(b) Balance of Plant

The two reactors are connected to a single turbine, are controlled by the common control room and share the same refuelling equipment. The two reactors are connected to a single turbine, are controlled by the common control room and share the same refuelling equipment.

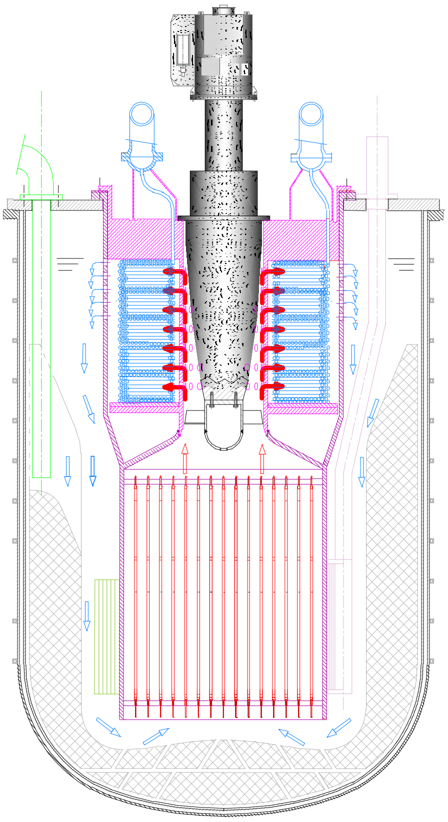
8. Design and Licensing Status

Although Hydromine has carried out economic evaluations of the LFR-AS-200 project with favourable results, does not intend to publicly provide economic data before receiving feedback from the licensing process, but only to provide some general considerations to resolve the dilemma between the cost-effectiveness of large plants and the SMRs to which the LFR-AS-200 belongs.

9. Development Milestones

2014	Completion of the conceptual design
2020	Preliminary design ongoing

LFR-TL-X (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. (HNE), 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
NSSS Operating Pressure (primary/secondary), MPa	Atmospheric, non-pressurized
Core Inlet/Outlet Coolant Temperature (°C)	360 / 420
Fuel type/assembly array	LEU, cylindrical cassette
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	19.75
Core Discharge Burnup (GWd/ton)	40
Refuelling Cycle (months)	≥100
Reactivity control mechanism	Ex-core, reversed-flag type, rotating staff moves absorbers closer to or away from the core
Approach to safety systems	Active + passive
Design life (years)	30
RPV height/diameter (m)	3.5 / 2
Seismic Design (SSE)	0.5g
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. Main 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 Design Approach

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.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

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.

(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 loop is adequate 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 Layout 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

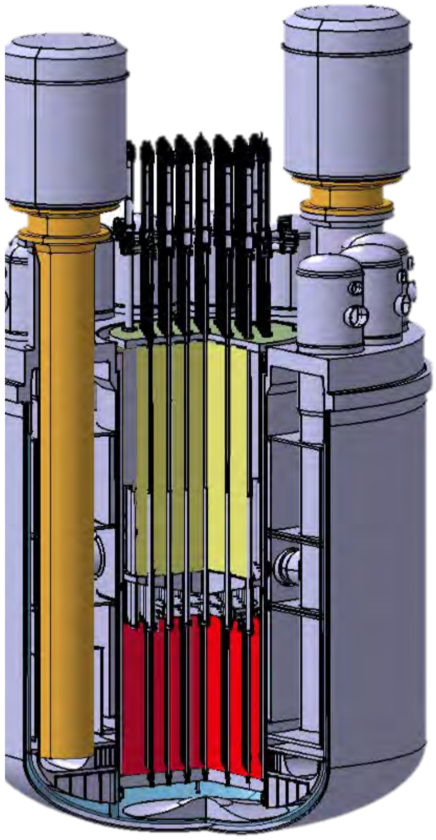
8. Design and Licensing Status

The LFR-TL-X is in the preliminary design stage, no licensing activities yet started.

9. Development Milestones

2017	Conceptual design work initiated
2020	Conceptual design in progress

SVBR (JSC AKME Engineering, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
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
NSSS Operating Pressure (primary/secondary), MPa	Low pressure
Core Inlet/Outlet Coolant Temperature (^oC)	340 / 485
Fuel type/assembly array	UO_2 / hexagonal
Number of fuel assemblies in the core	61
Fuel enrichment (%)	< 19.3
Core Discharge Burnup (GWd/ton)	60 (average)
Refuelling Cycle (years)	7 – 8
Reactivity control mechanism	Control rod drive mechanism
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	150 000
RPV height/diameter (m)	8.2 / 4.53
RPV weight (metric ton)	280, without core and coolant
Seismic Design (SSE)	0.5g
Fuel Cycle Approach/Requirements	On the first stage will use mastered UO_2 fuel with postponed reprocessing. In the more distant future transition to closed fuel cycle with self-supplying mode
Distinguishing features	Integral monoblock primary circuit where reactor, steam generators, pumps are installed in one vessel
Design status	Detailed design for potential construction in 2025

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 (NSs). 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 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 to satisfy 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 the NPPs with expired reactors' lifetime; and
- Nuclear desalination systems.

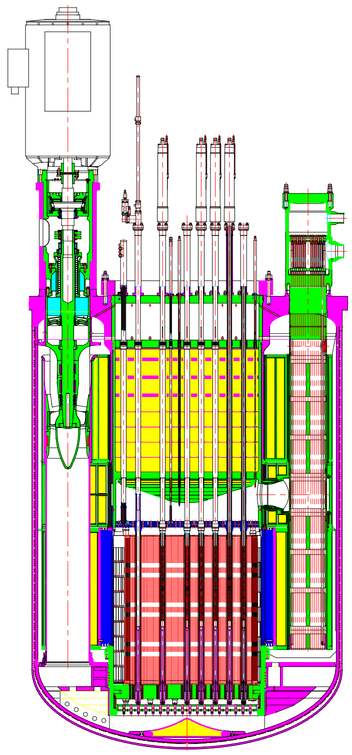
3. Main Design Features

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

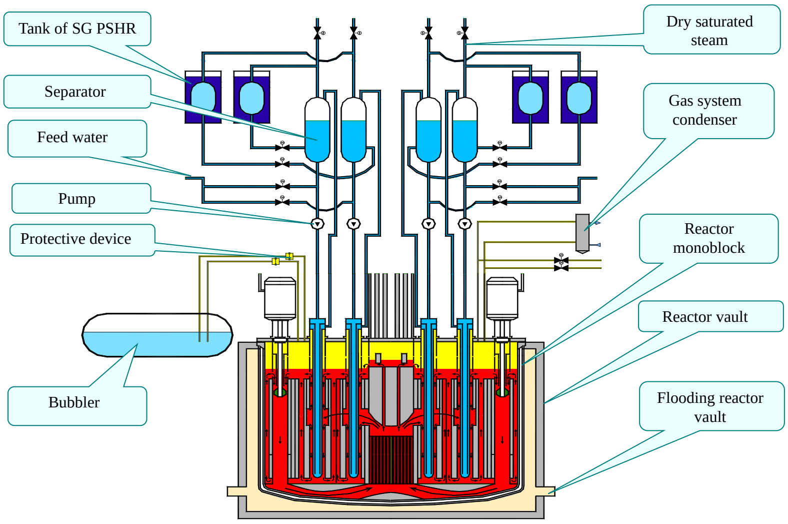
(b) Reactor Coolant System



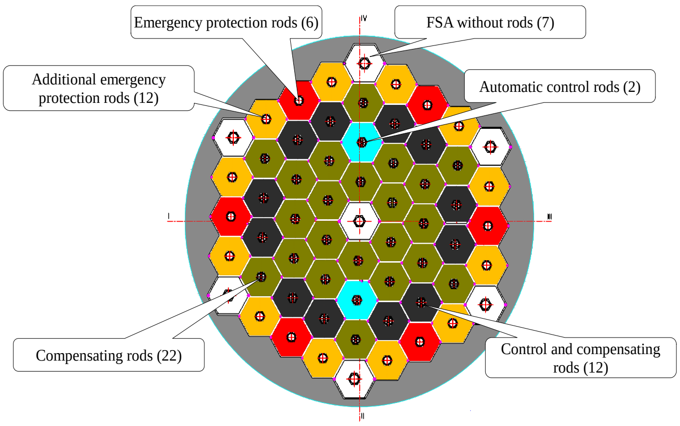
The entire primary equipment circuit of SVBR-100 is contained within a robust single reactor vessel. A protective casing surrounds the single-unit reactor vessel. The reactor transfers generated heat into a two-circuit heat-removal system and SG with forced multiple-circulation secondary coolant system. Natural circulation of coolant in the reactor heat removal circuits is sufficient to passively cool down the reactor and prevent superheating of the core. The coolant 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 entirely 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 the peripheral buffer chamber with a free surface level of the "cold" coolant. Another flow moves downwards and enters 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 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 be flooded by water in beyond the design accidents that involve failures of 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.



(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 the 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.

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 Monju NPP 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 level 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. Plant Layout Arrangement

General view of pilot plant is shown in Figure.

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. Fuel Cycle Approach

The reactor without changing the design can operate in various fuel cycles using different types of fuel. On mastered oxide fuel, core breeding ratio (CBR) will be less than one. On the MOX-fuel, the CBR will be slightly larger than one, and in the closed NFC, the SVBR-100 reactor will operate in the fuel self-supply mode with depleted uranium make-up.

9. Waste Management and Disposal Plan

Spent nuclear fuel (SNF) will be accumulated in the repository until the period when its reprocessing and nuclear fuel cycle (NFC) closure becomes economically viable. Today it is the cheapest fuel cycle.

10. Economic Characteristics

Parameter	**FOAK (pilot plant)**	**NOAK (4 RF)**
Cost of construction, $/kW(e)	~ 6000	~ 3000
LCOE, $/MWh	~ 100	~ 60

In accordance with Projected Costs of Generating Electricity, IEA, NEA, OECD, 2015 (discount rate 7%).

11. Development Milestones

2015	License for placement
2025	License for constructing (planned)
2031	License for operation and commissioning (planned)
2032	Serial production and supply of packaged equipment (planned)

SEALER (LeadCold, Sweden)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	LeadCold, Sweden
Reactor type	Small-sized lead cooled
Coolant/moderator	Lead
Thermal/electrical capacity, MW(t)/MW(e)	8 / 3
Primary circulation	Forced
NSSS Operating Pressure (primary/secondary), MPa	Non-pressurised
Core Inlet/Outlet Coolant Temperature (ºC)	390 / 432
Fuel type/assembly array	UO_2 / Hex-can
Number of fuel assemblies in the core	19
Fuel enrichment (%)	19.75
Core Discharge Burnup (GWd/ton)	33
Refuelling Cycle (full power years)	27
Reactivity control mechanism	Boron carbide, (W, Re)$^{10}B_2$
Approach to safety systems	Passive
Design life (years)	30
Plant footprint (m^2)	600 / 10 000 (buildings / fence)
RPV height/diameter (m)	6.0 / 2.748
Seismic Design (SSE)	None
Distinguishing features	SEALER is a nuclear battery
Design status	Conceptual design

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. Main 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 (considering 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$, 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 allows 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 a sufficient elevation of the steam generator to remove 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

SEALER requires compact steam generator. 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

Due to the above 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 820K, 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 the End of Life (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 Layout 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, United States of America)

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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), United States of America
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
NSSS Operating Pressure (primary/secondary), MPa	13.3 (Peak)
Core Inlet/Outlet Coolant Temperature (ºC)	550 / 850
Fuel type/assembly array	UC pellet / hexagon
Number of fuel assemblies in the core	85
Fuel enrichment (%)	~14.5 (LEU)
Core Discharge Burnup (GWd/ton)	~130 (average)
Refuelling Cycle (months)	~360
Reactivity control mechanism	Control rod drive mechanism
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	~90 000 (4 modules)
RPV height/diameter (m)	12.5 / 4.6
RPV weight (metric ton)	700
Seismic Design (SSE)	N/A
Fuel cycle requirements / Approach	Open fuel cycle
Distinguishing features	Silicon carbide composite cladding and fission gas collection system.
Design status	Conceptual design

1. Introduction

Energy Multiplier Module (EM^2) is a helium-cooled fast reactor with a core outlet temperature of 850°C. It is designed as a modular, grid-capable power source with a net unit output of 265 MW(e). 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™ (silicon carbide technology developed by GA).

2. Target Application

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

3. Main Design Features

The EM^2 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

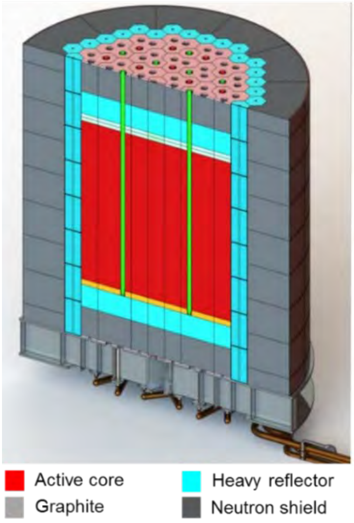
EM² design philosophy is to develop a new nuclear plant to address the following goals for enhancing the likelihood of commercial success:

- 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 EM² 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.

The basic building block of the EM² 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 EM² 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.



EM² core arrangement composed of fuel and reflector.

(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

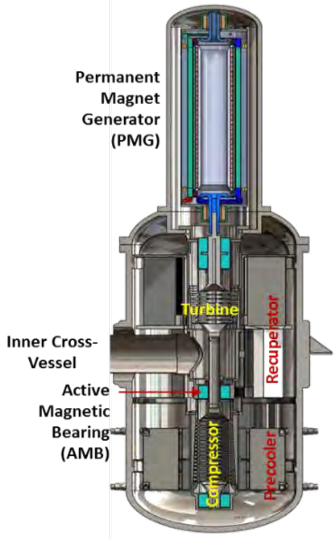
Reactivity control is provided by the 18 control rods and 12 shutdown rods. Both control rod system and shutdown rod system each have sufficient negative reactivity to render the core cold subcritical. The control and shutdown drives are located at the top of the reactor vessel. The control rod drives 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

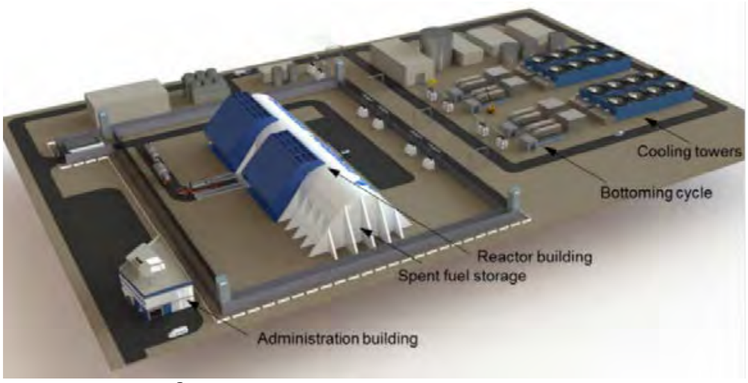
Unlike large power reactors, each EM^2 module utilizes a unique, non-synchronous, variable speed generator with a frequency inverter and generator load commutator to follow load demand. In the automatic load following mode, the generator speed is set by modulating the generator speed. A field-oriented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

The baseline EM² plant is composed of four 265 MW(e) modules for a combine net power of 1060 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.



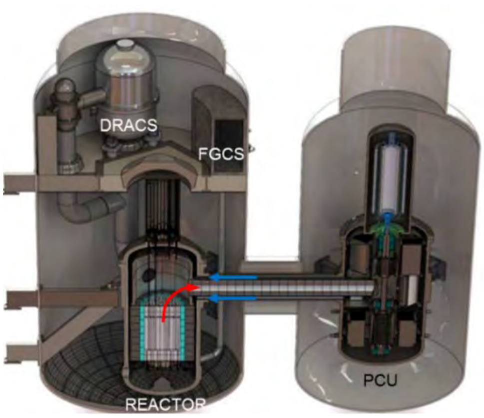
EM² plant layout on 9 hectares of land.

(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) Reactor Building

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



EM² primary heat transport system enclosed in two-chamber sealed containment

8. Design and Licensing Status

The EM² 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. Fuel Cycle Approach

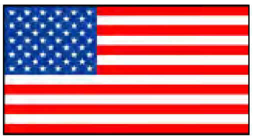
The EM² open fuel cycle with LEU/DU vented fuels exceeds 30 years without refuelling or shuffling, leading to a reduced cost of power, low proliferation risk, high fuel utilization, and low mass of waste streams. The core is capable of burning used light water reactor (LWR), plutonium, and thorium fuels. Fissile self-sufficient fuel cycle is feasible by removing fission products from the EM² spent fuel.

10. Waste Management and Disposal Plan

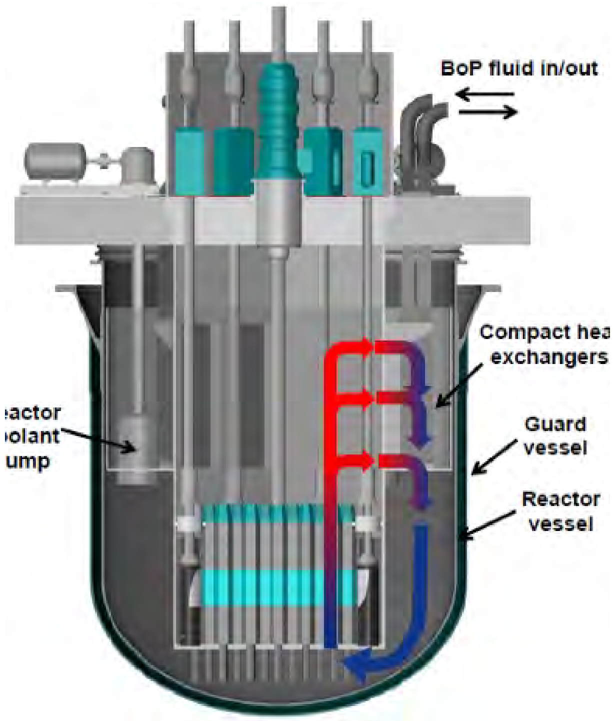
The radioactive waste handling system collects, transfers and stores radioactive materials from plant operating systems, including immobilized fission gases from the vented fuels. The spent fuels are stored in storage facility at site. After cooling, the used fuel will be directly disposed of or recycled.

11. 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.



Westinghouse Lead Fast Reactor (Westinghouse Electric Company, 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
NSSS Operating Pressure (primary/secondary), MPa	Nearly atmospheric
Core Inlet/Outlet Coolant Temperature (ºC)	420 / 600 (or higher, pending materials demonstration)
Fuel type/assembly array	Oxide, with provision for transition to UN
Fuel enrichment (%)	≤ 19.75%
Core Discharge Burnup (GWd/ton)	≥ 100
Refuelling Cycle (full power years)	≥ 24
Approach to safety systems	Passive: IAEA passive safety category B goal
Design life (years)	60 components, 100 structures
RPV height/diameter (m)	Approx. 8.0 / 7.5
Seismic Design (SSE)	Isolation of primary systems
Distinguishing features	High pressure containment not required; compact configuration with hybrid microchannel heat exchangers; 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 challenging 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 fulfill a range of non-electric applications such as process heat, desalination, and hydrogen production needs according to market demand. Its output is sufficiently small to integrate 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 lower 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 an expanded 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. Main 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) 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.

(c) 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 MW.D/kgHM 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 U_{235} fuel with enrichment <20% while maintaining the option to burn Pu-containing fuel.

(d) 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.

(e) Balance of Plant

While supercritical water remains a high-technology-readiness, high-performance option for the LFR, a significant advantage can be captured using sCO_2. The 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 significantly 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
- Favorable reactivity feedback typical of liquid metal fast reactors
- Robust, microchannel hybrid PHE reduces chance of secondary break and substantially reduces its severity; and
- Underground placement of components important to safety.

(b) Passive Heat Removal

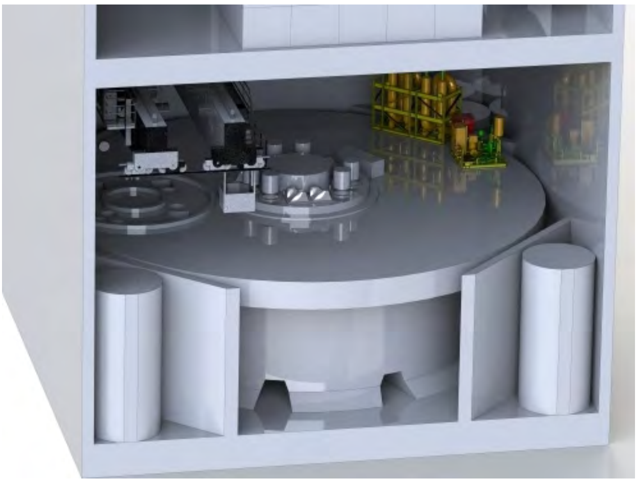
The LFR harnesses its high temperature capability in order to use radiation heat transfer between the RV and GV to remove reactor core decay heat during a safety event. The GV is submerged in a pool of water sufficient to remove heat from the reactor through boiling long enough to ultimately transition 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 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, most 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 Layout 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 Licensing Status (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 behaviors and properties of lead coolant, material compatibility, and new safety-related phenomena.

8. Development Milestones

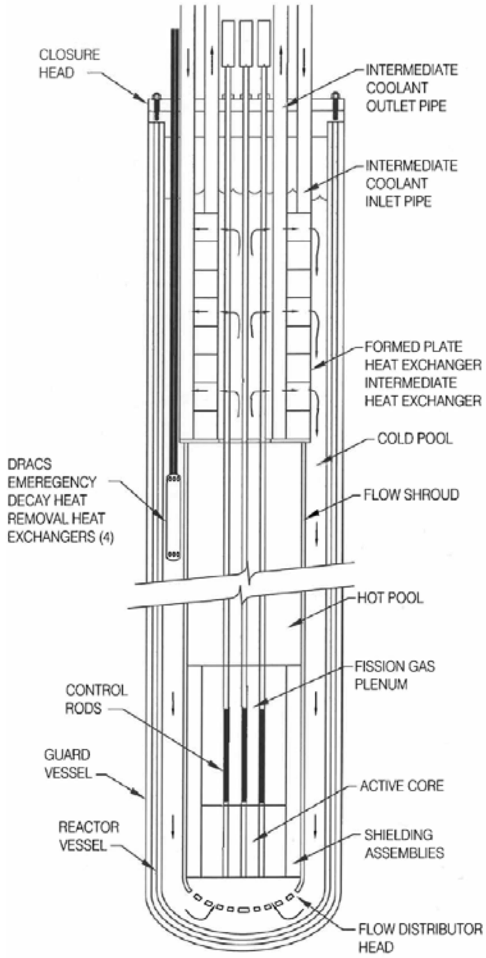
Achieving a plant with the number of innovative features as described herein is a considerable step forward for the commercial nuclear power industry. As a result, a staged approach is planned to demonstrate key aspects of the design while additional testing is performed to validate further enhancements 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.

By adopting 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.

SUPERSTAR (Argonne National Laboratory, United States of America)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Argonne National Laboratory, United States of America
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
NSSS Operating Pressure (primary/secondary), MPa	Recuperated supercritical CO_2 Brayton cycle, 25 MPa
Core Inlet/Outlet Coolant Temperature (°C)	400 / 480
Fuel type/assembly array	Particulate-based U-Pu-Zr metallic fuel with weapons Pu
Number of fuel assemblies in the core	188
Fuel enrichment (%)	< 12
Core Discharge Burnup (GWd/ton)	55 (mean) / 84 (peak)
Refuelling Cycle (full power years)	15
Reactivity control mechanism	W-Re control rods containing B_4C enhanced in B_{10}. Control rods are denser than lead coolant
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	To be determined
RPV height/diameter (m)	22.86 / 4.57
RPV weight (metric ton)	To be determined
Seismic Design (SSE)	Seismic isolation
Fuel cycle requirements / Approach	Use of weapons grade Pu. Used fuel sent to storage.
Distinguishing features	No refuelling equipment on site except during refuelling; heavy Led-cooled intermediate circuit; supercritical CO_2 Brayton cycle, autonomous load following
Design status	Conceptual design

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 or overland transporter. 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

For deployment demonstration in remote regions and on small grids. Later, it will be international deployment in developing nations with growing economies and infrastructures that cannot support economy-of-scale plants.

3. Main 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 the T91 with 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 (RV) 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 the RV. This feature enables the RV to be exposed to only the cold lead inside of the cold pool during normal operation. Consequently, the RV 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 heat up and cooldown transients and thermal stresses in the portion of the RV that would otherwise be exposed to hotter primary coolant during startup and shutdown operations. There is no fuel handling equipment located inside of the RV.

(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 Divison 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 RV. 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 RV 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 Po_{210}.

(h) Steam Generator

Compact diffusion-bonded HLMC-to-CO_2 heat exchangers are utilized with a recuperated supercritical carbon dioxide Brayton cycle power converter with heat rejection to the air atmosphere heat sink.

(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 RV. 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 natural circulation within the DRACS circuit intermediate coolant, the louvers has an orifice that induces a small 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. 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 RV to maintain a sufficiently high faulted level of lead inside of the RV to maintain natural circulation heat transport in the event of a RV's 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 Layout 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

Design suspended due to lack of funding. Pre-licensing interactions with regulator not started.

9. Fuel Cycle

Use of weapons grade Pu. Used fuel and waste sent to appropriate storage facilities.

10. Waste Disposition

Used fuel and waste sent to appropriate storage facilities.

11. Development Milestones

The SUPERSTAR concept was initially developed in 2010 and 2011.

MOLTEN SALT
SMALL MODULAR REACTORS

Integral Molten Salt Reactor (Terrestrial Energy Inc., Canada)

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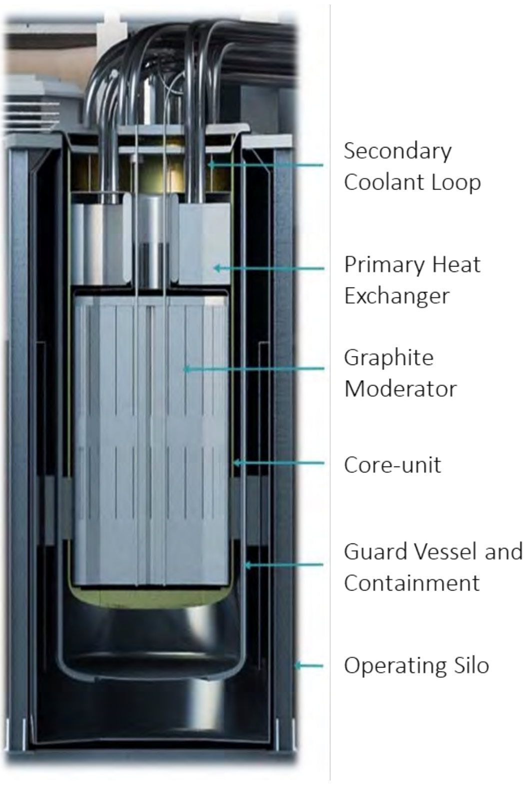


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Terrestrial Energy Inc., Canada
Reactor type	Molten salt reactor
Coolant/moderator	Fluoride fuel salt/graphite
Thermal/electrical capacity, MW(t)/MW(e)	440 / 195
Primary circulation	Forced circulation
Operating Pressure (primary/secondary), MPa	< 0.4 (hydrostatic)
Core Inlet/Outlet Coolant Temperature (°C)	620 / 700
Fuel type/assembly array	Molten salt fuel
Fuel enrichment (%)	<5%, low enriched uranium
Fuel cycle (months)	84; before core-unit replacement
Main reactivity control mechanism	Short term: negative temperature coefficient Long term: online liquid fuel additions
Approach to safety systems	Passive
Design life (years)	56
Plant footprint (m^2)	45 000
RPV height/diameter (m)	10.0 / 3.7
RPV Weight (metric ton)	154 000
Seismic design (SSE)	0.3g
Distinguishing features	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 440 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.

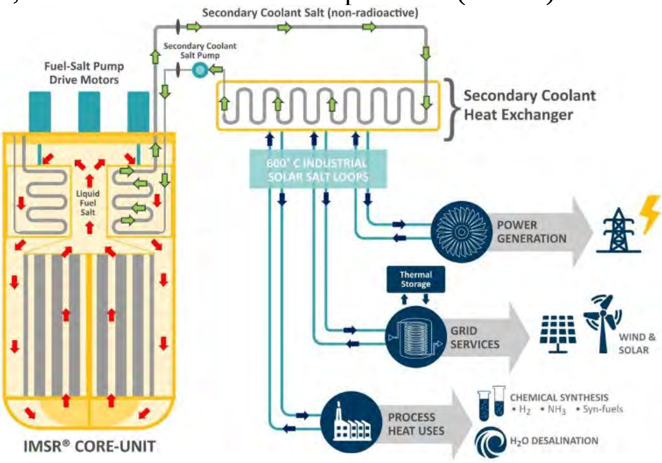
2. Target Application

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. Main 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 circulates 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, it 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 if 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 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 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, in the form of uranium tetrafluoride (UF_4), is dissolved in a eutectic mixture of low-cost fluoride salts without the addition of either lithium or beryllium. The benefit of this eutectic is that it minimizes the production of tritium which hampers any MSR design proposal that includes lithium fluoride (LiF) and/or beryllium difluoride (BeF_2) in its fuel salt mixture. The IMSR400 liquid Fuel Salt is an integral system – nuclear fuel, coolant and heat transfer medium – which provides the basis for a less complex reactor configuration with many safety attributes. 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

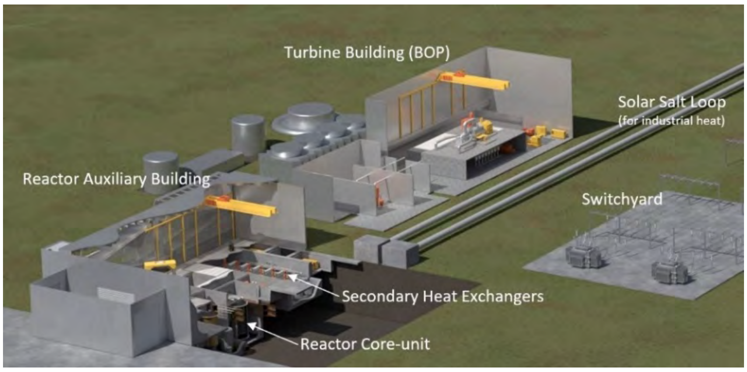
The IMSR® core-unit is designed to be simple and safe to replace. This supports a high operational utility factor for the IMSR® power plant and high capital efficiency. The replaceable core-unit ensures the materials' lifetime requirements of all other reactor core components are accommodated by the 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. 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 Layout 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

Terrestrial Energy completed Phase 1 of the Canadian Nuclear Safety Commission's (CNSC) vendor design review (VDR) in November 2017, successfully meeting CNSC's requirements for the IMSR® design. In October 2018, the IMSR® entered Phase 2 pre-licensing VDR with the CNSC. Phase 2 VDR involves a detailed follow-up on Phase 1 VDR activities, and an assessment of the IMSR® power plant design's ability to meet all 19 CNSC focus areas of a power plant's future license application.

9. Fuel Cycle Approach

The fuel may consist of low enriched uranium fluoride, plutonium fluoride, thorium fluoride, or any mixture of these. The first of a kind IMSR400 however, will utilize a once-through, low enriched uranium fuel cycle as this is the simplest option.

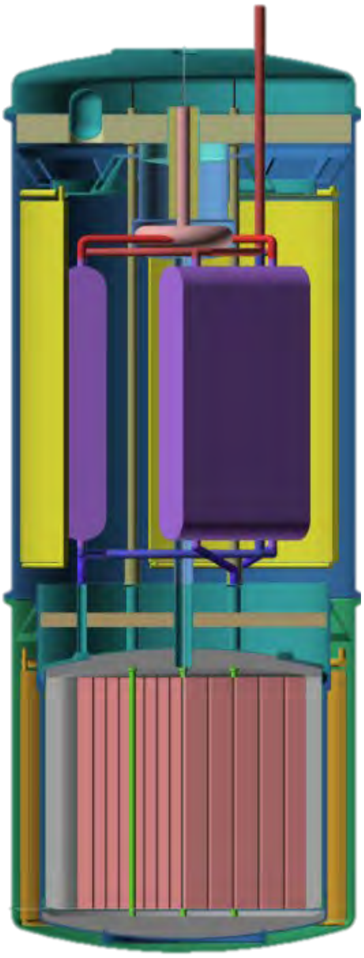
10. Waste Management and Disposal Plan

The spent liquid fuel inventory is much easier to recycle than solid fuel elements, making it more attractive to recycle the fuel. If fuel recycling is employed, Terrestrial Energy envisages this to occur in a central fuel recycling centre, servicing many IMSR power plants.

11. Development Milestones

2015	Conceptual design completed
2016	Start of basic engineering phase
2017	Completion of CNSC pre-licencing phase-1 vendor design review
2018	Commenced CNSC pre-licencing phase-2 vendor design review
Early 2020's	Secure necessary licenses
Early 2020's	Commence construction of a first full-scale IMSR® NPP in Canada

smTMSR-400 (SINAP, CAS, China)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	SINAP, CAS, China
Reactor type	Molten salt reactor
Coolant/moderator	$LiF\text{-}BeF_2\text{-}ZrF_4\text{-}ThF_4\text{-}UF_4$ fuel salt / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	400 / 168
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Near ambient
Core Inlet/Outlet Coolant Temperature (ºC)	650 / 700
Coolant type (Secondary loop)	Molten salt fuel
Number of prismatic graphite blocks (defining the fuel channels)	~240
Fuel enrichment (%)	19.75
Core Discharge Burnup (GWd/ton)	~300
Refuelling Cycle (months)	120 (batch-reprocessing off-line)
Reactivity control mechanism	Control rods, Negative reactivity feedback, Online fuel addition, Draining off fuel salt
Approach to safety systems	Passive
Design life (years)	60
Reactor module height/diameter (m)	~10 / 3.8
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Th+U loading initial, LEU addition online, FP gas removal online, Batch-reprocessing offline
Distinguishing features	Replaceable Reactor module (~10 years), Passive safety, Near 40% power contributed by thorium
Design status	Pre-conceptual Design

1. Introduction

A three-step development route has been proposed by Shanghai Institute of Applied Physics, Chinese Academy of Science, (SINAP, CAS), to realize thorium-uranium breeding in molten salt reactor by the middle of this century. smTMSR-400 is a 400MW(t) / 168 MW(e) small modular Thorium Molten Salt Demonstration Reactor that form part of the third step. It will demonstrate large-scale power produced based on the thorium fuel cycle and verify the conversion properties in smTMSR-400. Most off-line pyro-processing techniques, such as, fluoride volatility, vacuum distillation and electrochemical reduction, will be used in fuel cycle of smTMSR-400.

Commercialization at this stage focus the R&D efforts and can reduce the risk and cost. The option also enable the gradual establishment of the thorium fuel and MSR supply chain system. smTMSR-400 will demonstrate the highly intrinsic safety and engineering reliability of molten salt reactor, such as shutdown by negative temperature feedback and draining off fuel salt, passive residual heat removal systems. The aim of safety design is to avoid any off-site emergencies and therefore increase the siting flexibility. smTMSR-400 uses small and modular designs to reduce and simplify the R&D challenge and difficulty of MSR. Modular design will make MSR more flexible to deploy at different places with various demands. Modular manufacture and assembly can improve the quality of equipment and enhance safety and economy. Modular construction can also reduce the financing cost and risk.

2. Target Application

smTMSR-400 is designed as a thorium convertor and in situ burner driven by low enriched uranium. The fuel cost is expected to be cheaper than pure uranium fuel mode. After offline batch reprocessing, thorium and uranium can be recycled into a new reactor in order to minimum the spent fuel mass as well as enhance the neutron economy.

It will be applied as a high temperature heat source, which not only can be used for electricity generation, but also can satisfy the energy diversified demands, such as, seawater desalination, heat supply, supercritical steam supply for industry demands and hydrogen production, etc.

3. Main Design Features

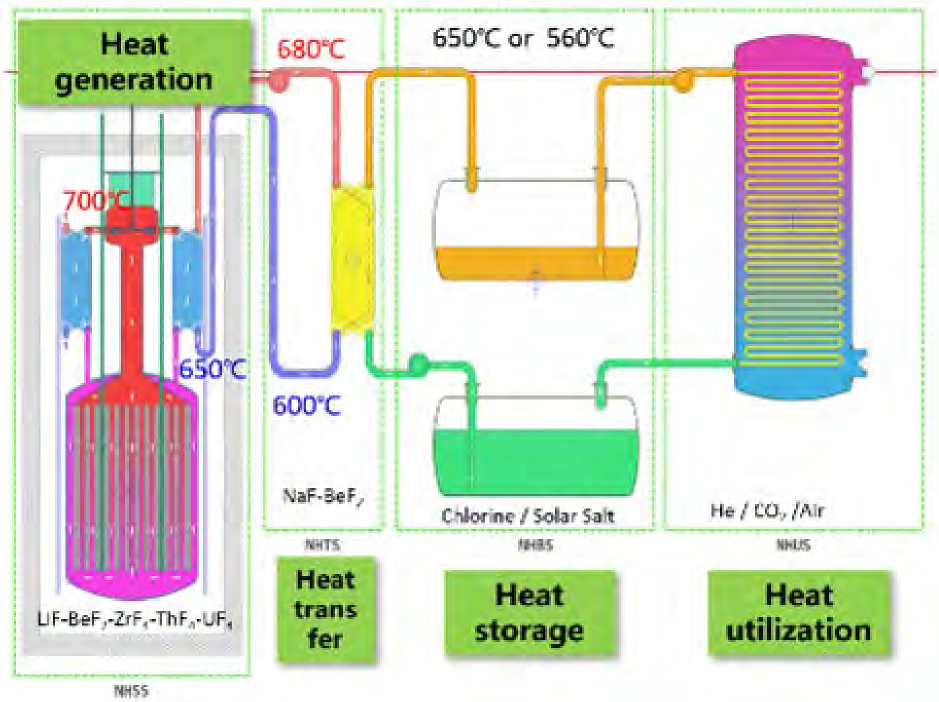
(a) Design Philosophy

The design philosophy of smTMSR-400 includes:

- Using mature technologies and experiences accumulated from TMSR-LF1 project and potential technologies to be developed in the next years.
- The primary loop is designed as a compact reactor module and can be replaced every 8-10 years to solve the long-term irradiation problems of materials.
- Online fuel addition and offline batch reprocessing mode is used. The power contribution from thorium should be more than 40%. The spent fuel can be used for offline pyro-processing and recycled for three times.
- Passive safety design, consist of negative temperature feedback, passive fuel salt discharge, passive residual heat removal systems, and radioactive nuclides retention by salt coagulation.
- Heat storage system is applied for comprehensive energy utilization and peak-shift of electrical demand.

(b) Nuclear Heat Supply System

The nuclear heat supply system consists of reactor module, heat transfer system, heat storage system and heat utilization system. These subsystems will be detailed introduced in the following sections.



(c) Reactor Module

Reactor module is designed as a compact loop structure with reactor core, three salt/salt heat exchangers, one centrifugal pump and connecting pipelines. Reactor core is filled with LiF-BeF_2- ZrF_4-ThF_4 -UF_4 fuel salt and hundreds of prismatic graphite blocks. U_{235} is 19.75% enriched. Three interval angles of the graphite blocks are cut off, and three adjacent graphite blocks can form a fuel salt channel. Three salt/salt plate heat exchangers are arranged upon the reactor core side by side for compact layout. The size of reactor module is about ~3.8 m in diameter and ~10 m in height, which is suitable for railway transportation. Reactor module is packed in a safety vessel for additional radioactive confinement barrier. The reactor vessel and metal internals in touch with fuel salt are made by nickel-based alloy.

The fuel salt is heated up to 700°C by fission energy in the active core (fuel salt channels in graphite moderated core), then flows through the upper plenum, upper pipeline, driven by the pump, and then to the three plate heat exchangers. After heat transfer, the fuel salt is cooled down to 650°C, then flow through the down-comer, lower plenum and back to the fuel salt channel.

(d) Reactivity Control

Six control rods are used to control the reactivity change covering from subcritical status, power lift condition, xenon poison, operation disturbance, and excess reactivity. Most of control rods are used to adjust the power lift condition due to the negative temperature feedback.

(e) Heat Transfer System

Heat transfer system is designed to isolate the radioactive release from reactor module and avoid chemical pollution from heat storage system to reactor module. The coolant is chosen as NaF-BeF_2 due to its good thermal properties (low melt point, high boil point), good chemical stability and compatibility. The operation temperature of NaF-BeF_2 is from 600°C to 680°C.

(f) Heat Storage System

Heat storage system is a standby option and is designed into a double tank structure for peak-shift of electrical demand and heat recovery. Chlorine salt or solar salt will be used as coolant and storage media. The temperature of cold tank and hot tank is 290°C and 650°C (650°C for solar salt).

(g) Heat Utilization System

The heat can be applied for electricity generation by helium/air/CO_2 Brayton power system based on their technological maturity and application scenarios. Also, it can be applied for seawater desalination, heat supply and steam supply.

(h) Fuel Management System

Fuel management system is used for initial fuel loading, online fuel addition and fuel salt discharge under normal or accident conditions. The fuel salt in all those systems is driven by gas pressure and gravity.

(i) Online Fission Gas and Tritium Removal System

Fission gas and some noble metals are removed from the reactor module by entrained cover gas. Then they will flow into fission gas removal system for decay and separation. Tritium control technology is developed by TMSR, including tritium resistance coating, tritium absorption alloy, etc.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

smTMSR-400 contains two sets of control rod system with different driving mechanisms, which will automatically go down in accidents. Low excess reactivity will also make it possible to shut down with the negative temperature feedback alone. After that, fuel salt will be drained off for long-term shutdown.

(b) Decay Heat Removal System

smTMSR-400 is designed with two passive decay heat removal systems (DHRS). Natural circulation will form in the reactor module in case of force circulation failure. When the nuclear heat supply system fails, two kinds of DHRSs will ensure long-term cooling of fuel salt. One is located around the safety vessel in the silo (DHRS1), another is in the fuel salt drain tank directly installed below the reactor vessel (DHRS2). DHRS1 is operated under normal accidents to cool the safety vessel and reactor vessel. DHRS2 is triggered after the freeze-valve opens passively by increased accident temperatures or by active heating.

(c) Reactor Cooling Philosophy

There are four containment barriers in smTMSR-400. The first one is the reactor module made by high-temperature-resistant and corrosion-resistant nickel-based alloy, the freeze valve function prevents the first barrier from exceeding its temperature limits and thus to keep its integrity under accidents. The second one is the safety vessel, which will ensure gas tightness and contain the leaked salt under beyond design basic accidents. The third one is the fuel salt itself, which has a large degree of retention on some important radioactive elements. The solidification of fuel salt will prevent further leakage. The final one is the underground construction, which can effectively prevent the spread of radioactive materials in an accident and can resist natural disasters and terrorist attacks.

(d) Chemical Control System

Online chemical control system is designed to control the oxidation-reduction potential of fuel salt.

5. Plant Safety and Operational Performances

Inherent safety and using air as the ultimate heat sink make the site selection of smTMSR-400 more flexible, such as remote area, dry region and suburbs. The heat storage and high temperature output ability broaden the energy applications. Modular manufacture, transportation and construction will enhance economic competitiveness.

6. Plant Layout Arrangement

The reactor module, heat transfer system, fuel management and gas removal system are under the ground. The heat storage system and heat utilization system are above the ground. Power expansion can be realized by employing multiple reactor units. The power plant can adopt single unit, two units and different units at one site, and realizes different power output to meet the energy needs of different application sites.

7. Design and Licensing Status

The design is in a pre-conceptual stage of development and licensing activities have not yet been undertaken.

8. Fuel Cycle Approach

A pilot stage pyro-processing facility will be established in 2030s.The spent fuel in smTMSR-400 will be sent to the facility for fuel salt recovery and to minimize the radioactive inventory for disposal.

9. Development Milestones

2018-2021	Pre-conceptual design phase and technology research.
2022-2024	Conceptual design phase and technology validation.
2025-2027	Engineering design phase.
2028 -	Projected deployment (start of construction) time.

Copenhagen Atomics Waste Burner 0.2.5 (Copenhagen Atomics, Denmark)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Copenhagen Atomics, Denmark
Reactor type	Molten salt reactor
Coolant/moderator	Fuel salt/Heavy water
Thermal/electrical capacity, MW(t)/MW(e)	100 / (not defined)
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.05-0.25 / 0.1-0.25
Core Inlet/Outlet Coolant Temperature (ºC)	600 / 650 - 700
Initial fissile inventory	Transuranic
Fuel type/assembly array	LiF-ThF_4 / none
Online reprocessing	Vacuum spraying
Power conversion process	heat source, interchangeable with power conversion system or industrial process
Number of fuel assemblies	Single fluid - molten-salt
Fuel enrichment (%)	Inventory of spent nuclear fuel
Core Discharge Burnup (GWd/ton)	900-1000
Refuelling Cycle (months)	Continuous operation / fuel salt as needed
Reactivity control mechanism	Heavy water level adjustment
Approach to safety systems	Passive
Design life (years)	3-5
Plant footprint (m^2)	2500
RPV height/diameter (m)	12 / 2.4
Fuel cycle requirements / Approach	spent fuel initiated / conversion to Th-U cycle
Distinguishing features	Liquid moderator, Low fissile inventory, and Potential for breeding
Operation	Firmware, no human operators
Design status	Conceptual design

1. Introduction

The Copenhagen Atomics Waste Burner version 0.2.5 is a small modular 50MW(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 and liquid moderator, fuel salt loop and coolant salt loop, fission product extraction and separation systems, dump tank, heat exchanger, pumps, valves, and compressors are all contained in a leak tight stainless steel containment, the size of a 40 feet shipping container.

2. Target Application

The design target is to design the main components to fit inside a leak tight stainless-steel containment, the size of a 40 feet shipping container. The he following applications are foreseen:

- Addon units at existing nuclear sites, possibly coupled with a spent nuclear fuel reprocessing unit.
- A heat source of 500-600ºC for existing industrial sites.
- Biofuel production and desalination plants.
- Ship or barge-based power systems.
- Baseload power in Asia and Africa.

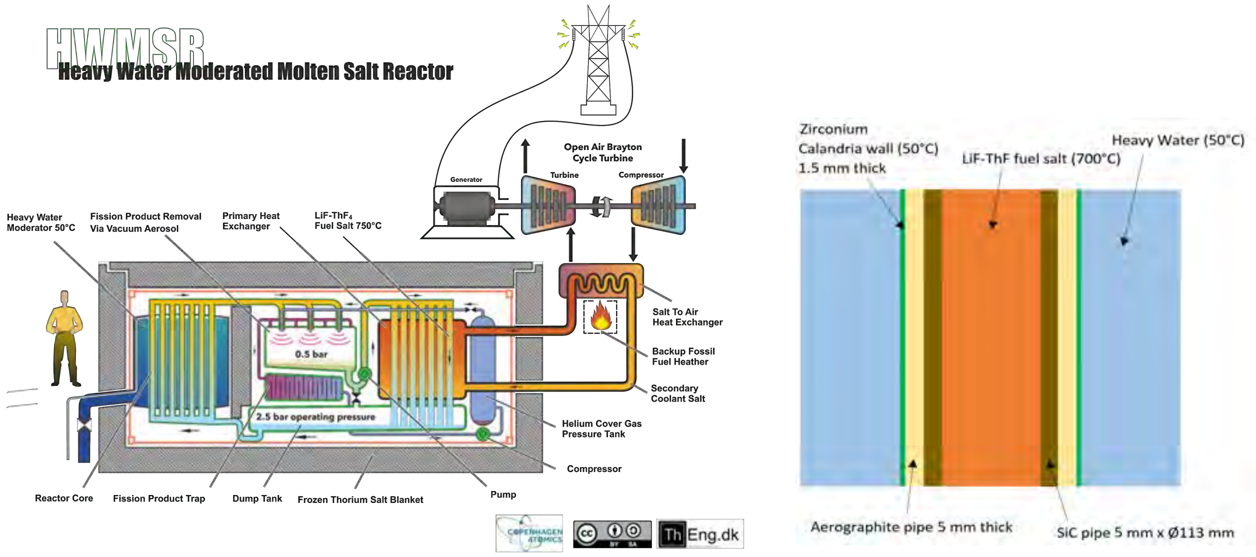
3. Main Design Features

(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 does not focus their design development on the first generation of molten salt reactors that last for 60 years or more. Rather, the organization instead focused 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 makes 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 0.2.5 is made to fit inside of a leak tight stainless-steel containment, the size of a 40-foot shipping container.

iii. Walkaway Safe and Prime Minister Safe:

Each Copenhagen Atomics Waste Burner 0.2.5 unit is expected to operate fully autonomously, with experience and learnings being synchronized through firmware updates to the whole reactor fleet. This, along with clever 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

Copenhagen Atomics Waste Burner 0.2.5 is intended as a heat source, interchangeable with any power conversion system or industrial process. The reactor will be owned, operated, and decommissioned by Copenhagen Atomics and the heat sold by the kWh to the customer.

(c) Reactor Core and Fuel Characteristics

The Copenhagen Atomics Waste Burner 0.2.5 is a heavy water moderated, single fluid, fluoride salt based, thermal spectrum, molten salt reactor. The fuel salt is LiF-ThF_4 with an initial fissile inventory of spent nuclear fuel transuranic. The fuel salt is the heat transport medium, and the fuel salt flow rate determines the core thermal power. The fuel salt flows through thermally insulated pipes in the core, surrounded by the heavy water moderator. The figure shows a possible configuration leading to manageable heat loss (excluding of course the heating from thermalization of neutrons and other particle interactions). The small reactor core tank also operates at low pressures and has a short design lifetime.

While most of the transuranic suffers from poor reproduction factor in the thermal spectrum, the bred U_{233}

does not. It is expected that the Copenhagen Atomics Waste Burner 0.2.5 will start off with a breeding ratio below one but increase in future versions to well above one as the initial fissile inventory of transuranic is converted to a fissile inventory of U_{233}. Future versions might employ a dual fluid design if protactinium separation is deemed too cumbersome or to improve neutron economy.

(d) Heavy Water Moderator

Unpressurized heavy water is chosen as the moderator due to its superior neutron economy and its ability for increased power to fissile inventory over graphite moderation and ease of reactivity control. The heavy water is thermally insulated from the molten fuel salt, continuously circulated and cooled to below 50°C. Besides its radiolysis, which will be managed with a passive recombiner, heavy water is favoured since it suffers no long-term degradation (like swelling, cracking), or ingress of fuel salt and fission products, unlike graphite, whose integrity will probably be lifetime limiting.

(e) Online Fission Product and Tritium Separation

The Copenhagen Atomics Waste Burner 0.2.5 does not employ any online wet chemistry but rather aims to achieve online removal 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, and allows for a much larger fraction of the fission products to be extracted from the fuel salt.

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 gases 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 entails spraying the entire salt stream just after it leaves the core and before passing through the heat exchanger.

The enhanced performance is achieved by increasing the surface to volume ratio and hence increasing the evaporation rate of noble gases and volatile compounds, and decreasing the ambient pressure to a fraction of atmospheric pressure hereby increasing the volatility of the majority of the fission products. It is 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. At the end of each unit's expected 3-5 years operational lifetime the remaining fission products can be removed by a wet chemistry batch reprocessing at reprocessing plants. The 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.

(f) Lithium Breeding

Since it is 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 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

During normal operation 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 start-up temperature defect.

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.

(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. The fuel salt is always draining into the dump tank while continuously being pumped back out. On power loss all the fuel-salt will drain to the dump tank where passive decay heat removal is achieved by means of heat transfer into the surrounding earth, possibly via long oil-filled pipes. 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 fuel salt boundary, the 40 feet sized stainless steel container, and a tertiary steel containment structure. In addition, fission products and tritium are 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°C, there is 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 would be contained by the secondary containment alone.

6. Plant Layout Arrangement

The reactor and containment structures can be installed underground or inside existing building structure, functioning as an adjacent heat source for existing or new industrial plants.

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 may merge in the future. Notably a molten lithium-7 deuteroxide (^{7}LiOD) liquid moderator variant is researched.

Copenhagen Atomics' approach towards achieving a fast technology cycle and rapid prototyping of molten salt reactors is to focus on the chemistry, measurement technology, simulation and control software, and components that make 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. Copenhagen Atomics is developing, manufacturing, and testing leak tight active electromagnetic bearing canned rotor pumps, leak tight valves, plate heat exchangers, purifications systems, sensor and measurement equipment, and ceramics and composite core materials in molten salt loop scale experiments.

Copenhagen Atomics is planning to build and test a scaled down 1MW(t) demonstration molten salt reactor in the mid 2020's, before starting the approval process for the Copenhagen Atomics Waste Burner.

8. Fuel Cycle Approach

The Copenhagen Atomics Waste Burner 0.2.5 has an initial fuel composition of LiF-ThF_4-PuF_4, where the U_{233} production benefits from the high number of excess neutrons from the fissioning of plutonium. As the thorium fuel cycle converges towards equilibrium the breeding process benefits from U_{233}'s superior neutron economy. Therefore, the later generations of Copenhagen Atomics Waste Burners, version 2.0 and above, are expected to be breeder reactors.

9. Waste Management and Disposal Plan

The chosen fuel-salt and moderator mean we're now in the paradigm of infinite moderator and fuel lifetime with the core vessel being the only expendable part. Copenhagen Atomics is pursuing the possibility of automated resmelting of whole 40 feet reactor units. This will have the result that the salt, heavy water, and metals are all reused, vastly decreasing the amount of material that has to be decommissioned.

10. 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.
2019	Award two Danish research grants and start of collaboration with Alfa Laval.
2020	First public funding round and first fertile salt test.
2022	Completion of non-fission prototype of a 1MW(t) demonstration reactor.
mid 2020s	First test of 1 MW(t) demonstration reactor.
late 2020s	First test of a 100 MW(t) commercial reactor.

ThorCon (ThorCon International, United States of America and Indonesia)

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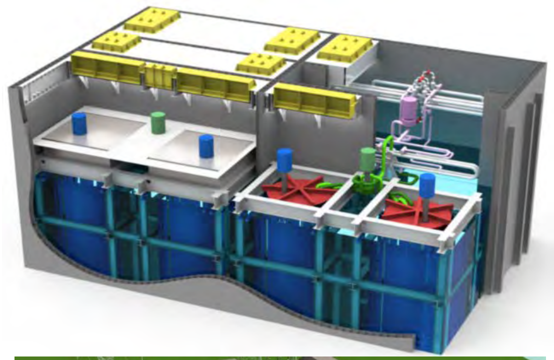
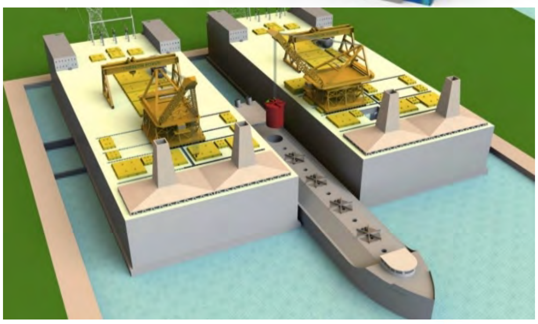
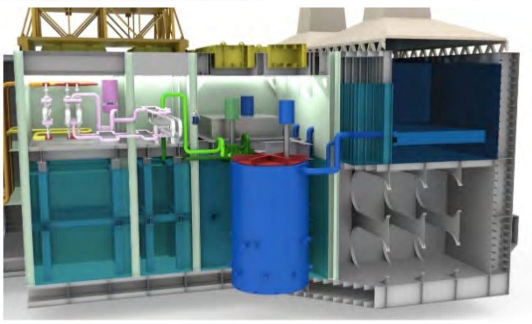






MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ThorCon International, first deployment in Indonesia
Reactor type	Thermal molten salt reactor
Coolant/moderator	Molten salts / graphite
Thermal/electrical capacity, MW(t)/MW(e)	557 / 250 (per module)
Primary circulation	Forced (4 pumps), 1 per loop
System operating pressures (4 loops: fuel salt, clean salt, solar salt, steam), MPa	0.6 / 2.9 / 0.86 /25.7
Core Inlet/Outlet fuel-salt temperature (^oC)	565 / 704
Fuel type/assembly array	UF_4, ThF_4 / n/a.
Number of fuel assemblies	1 (one mixture of fuel salt)
Fuel enrichment (%)	5.0 minimum / 19.7 maximum
Discharge burnup (GWd/ton)	12.4
Refuelling cycle (months)	48
Reactivity control mechanism	Negative temp coeff, salt flow rate; fissile/fertile additions
Approach to safety systems	Intrinsic, passive, using natural circulation, water
Design life (years)	80
Plant footprint (m^2)	67x174
Can height/diameter (m)	10.3 / 7.8
Can weight (metric ton)	343
Seismic design (SSE)	1.0 (under review)
Fuel cycle requirements / approach	Thorium converter. Fissile: LEU05, LEU19, or plutonium
Distinguishing features	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. Main 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.

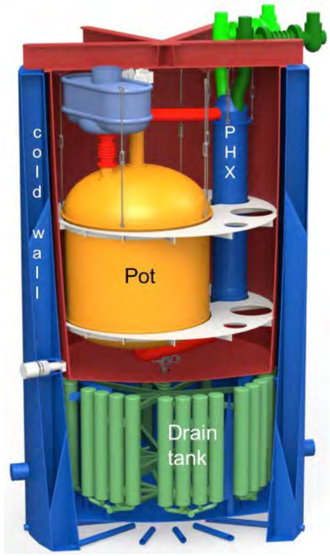
ii. ThorCon is Ready to Go - The ThorCon design needs no new technology. ThorCon is a scale-up of the successful Molten Salt Reactor Experiment (MSRE). A full-scale dual 250 MW(e) ThorConIsle prototype can be operating under test within four years and subjected to the failures and problems that the designers claim the plant can handle before commercial production can begin.

iii. ThorCon is Rapidly Deployable - The entire ThorCon 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 single large reactor yard can turn out thirty 500 MW(e) ThorCons per year. 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. 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 the used Can starts its 4-year cool down.

The fuel salt is a mixture of sodium, beryllium, uranium and thorium fluorides at 704ºC. 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 565ºC 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.

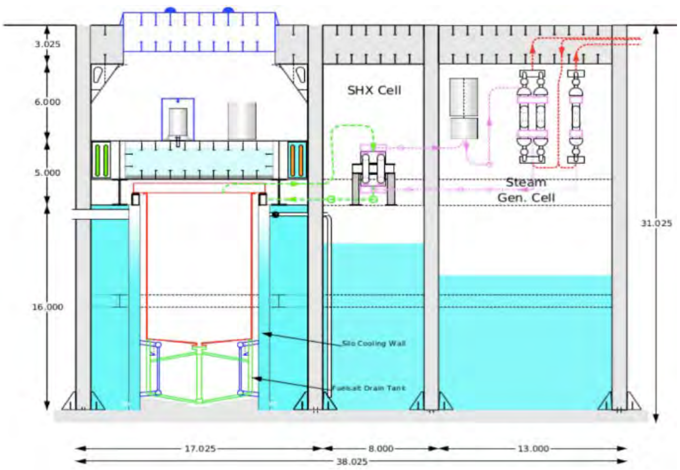
The Pot pressure is 3 bar gauge at the maximum stress point. The outlet temperature of 704ºC results in an overall plant efficiency of 46% with net electric power output of 250 MW per Can. Consumption of fissile uranium is 135 kg per year. The Can has only one major moving part, the pump impeller.

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 Fuel salt Drain Tank (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) Engineered Safety System Approach and Configuration

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 a year 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 holding tank in its own silo alongside the Can, within the power module, within the hull. The silo is passively cooled by basement water. The cooling fuel salt is as well protected as the fuel salt being burned. After 4 years the cooled fuel salt is transferred to transferred to a fuel cask in the vault module for storage. It may later be transferred to a shipping cask and 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 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 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. 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. 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 Layout Arrangement

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. Fuel Cycle Approach

ThorCon is a thorium converter. Each ThorCon will require 5.3 kg of 19.7% enriched uranium and 9.0 kg of thorium per day, on average. During the 8-year fuel cycle, a portion of the fertile thorium is converted to fissile U_{233} which then becomes part of the fuel.

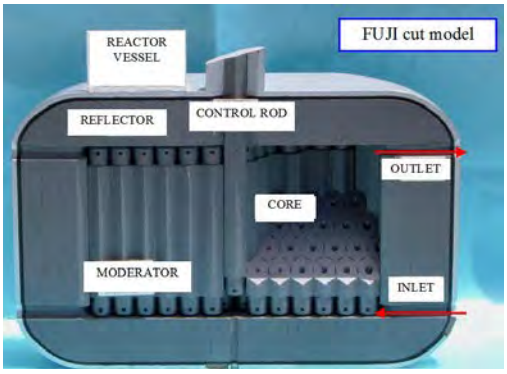
10. Waste Management and Disposal Plan

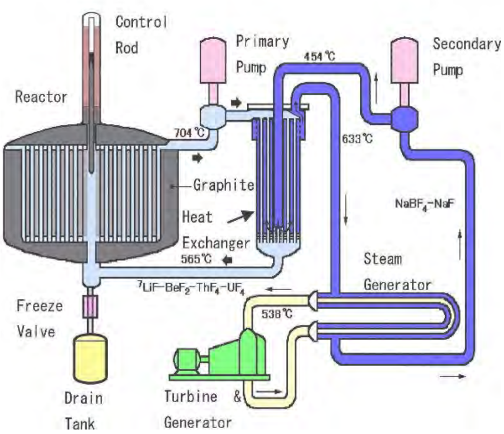
ThorCon will send one cask to storage every four years. After 8 full-power years of operation the fuelsalt is spent and transferred to a fuel cask in the vault module for storage. It can be transferred to a special fuel cask and removed by crane and loaded to a CanShip to transfer it to a facility for conditioning and disposal. The cost of Can recycling or disposal, and ultimate decommissioning (by re-floating the platform and scrapping it in a shipyard) are included in calculated LCOE.

11. Development Milestones

2015	Concept design completed
2019	Pre-licensing vendor design review in Indonesia; Basic engineering design complete
2021/22	Start construction of Pre-fission Test Platform; Testing of the Pre-fission Test Platform
2023/24	Construction of the demonstration power plant; Begin testing of the demonstration power plant
2025	Complete testing of the demonstration power plant and obtain type license
2026/28	Begin commercial construction of multiple power plants; Start of commercial operation

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
NSSS Operating Pressure (primary/secondary), MPa	0.5 / 0.5
Core Inlet/Outlet Coolant Temperature (°C)	565 / 704
Fuel type/assembly array	Molten salt with Th and U
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	2.0 (0.24%U233 + 12.0%Th). Pu or LEU can be used
Core Discharge Burnup (GWd/ton)	No mechanical limit for burnup
Refuelling Cycle (months)	Continuous operation possible
Reactivity control mechanism	Control rod, or pump speed, or fuel concentration
Approach to safety systems	Passive
Design life (years)	30
Plant footprint (m^2)	<5000 (R.B. + SG.B. + TG.B.)
RPV height/diameter (m)	5.40 / 5.34 (inner)
RPV weight (metric ton)	60 (made of Hastelloy N)
Seismic Design (SSE)	Same as LWRs
Fuel cycle requirements / Approach	Self-sustaining at FUJI-U3. No online reprocessing, except removal of gaseous FPs. Spent fuel salt is reprocessed
Distinguishing features	Self-sustaining at FUJI-U3. No online reprocessing, except removal of gaseous FPs. Spent fuel salt is reprocessed.
Design status	3 experimental MSRs were built. Detail design 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 (International Thorium Molten-Salt Forum: ITMSF), based on the ORNL's results to deploy it in the world.

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 1000 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 U_{233} 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, 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. Main 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 of graphite moderator within a reactor vessel. 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). The above figure shows only one loop, 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. Fuel assembly fabrication is not required for molten fuel salt, 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 HX to the SG. 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 (SG) and Turbine Generator (T/G)

A 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 T/G to produces 200 MW(e) electricity.

4. Safety Features

(a) Engineered Safety System Approach Configuration

In case of pipe break, leaked molten salt is drained to an emergency drain tank without passing through a freeze valve. A pressurization accident is very unlikely 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

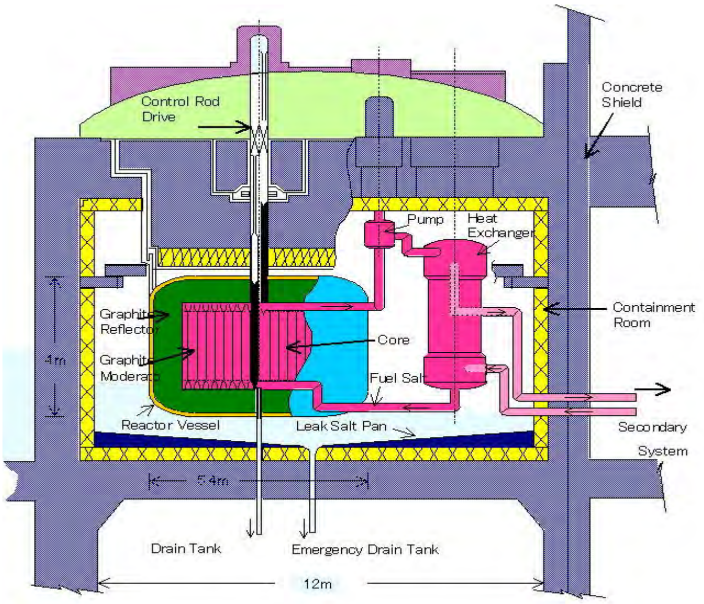
In normal shutdown condition, decay heat is transferred to a secondary loop and a steam-line loop, and disposed to the ultimate heat sink (seawater etc.). If all pumps in a primary or secondary loop stop, fuel salt is drained to a drain tank through a freeze valve. Decay heat at the drain tank is cooled 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 (ECCS)

As is explained above, redundant and diverse 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 very unlikely, 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 3 levels of containment. The 1st is the reactor vessel and pipes made of Hastelloy N. The 2nd 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 3rd level is a reactor building composed of two layers. As explained above, a pressurization accident is very unlikely due the low vapor pressure. Therefore, a containment cooling system and makeup water pools are not required.



5. Plant Safety and Operational Performance

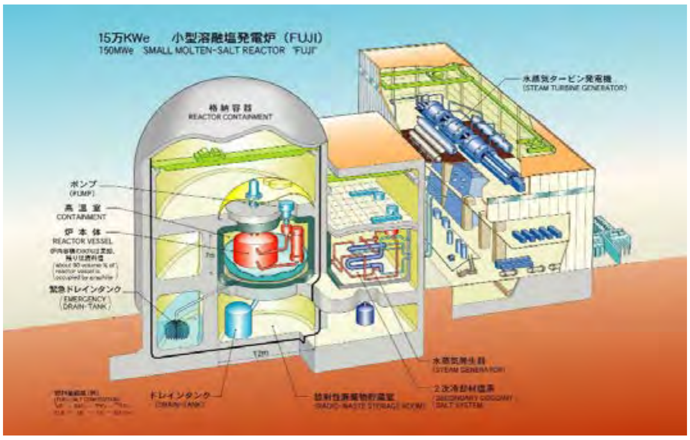
Overall safety is described above. In case of a station blackout (SBO: Loss of all AC electricity) the MSR-FUJI can be shut down and cooled without electricity. Core meltdown or steam/hydrogen explosion is physically excluded by design, and no ECCS is needed. Long-time and daily operations are described in Section-3(d). Based on those features, load following is performed without control rods.

6. Instrumentation and Control Systems

Instrumentation and control (I&C) systems in the MSR-FUJI design are the same as 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 Layout Arrangement

Major buildings of MSR-FUJI are a reactor building, a SG building with a main control room, and a T/G 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 SG 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 Plant

i. Turbine Generator Building (T/G Building):
The T/G building contains the supercritical turbine and generator, which produce electricity. Also, it contains condensers for disposed steam. The condensers use outside water (sea water etc.) 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. Fuel Cycle Approach

As explained in Section-1 and 3, FUJI-U3 design can attain a self-sustaining fuel cycle.
MSR-FUJI is simplified by removing the online reprocessing facility. Only gaseous fission products (FP) are always removed from molten fuel salt.
Spent fuel salt is discharged, and reprocessed at the off-site pyro-reprocessing facility. Some usable actinides can be sent back to the reactor.

10. Waste Management and Disposal Plan

Actinides such as U/Pu/Th/MA separated at the off-site reprocessing facility are recycled to MSR. FPs and salt are stored at disposal facility. These facilities are to be developed in parallel with MSR deployment.

11. 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	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 and Canada)

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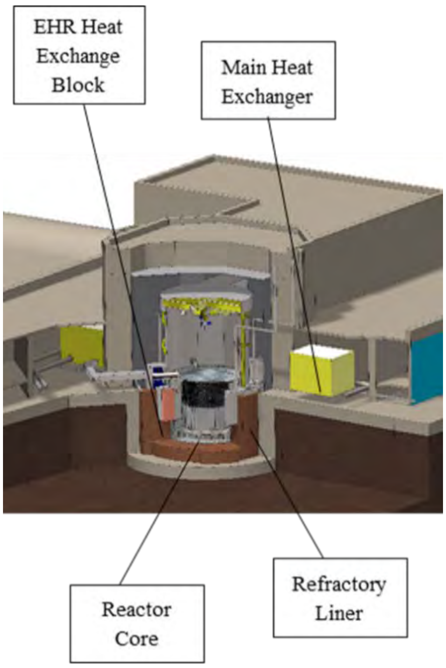


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Moltex Energy, United Kingdom and Canada
Reactor type	Static fuelled molten salt fast reactor
Coolant/moderator	No moderator, coolant is molten salt ZrF_4-KF
Thermal/electrical capacity, MW(t)/MW(e)	750 / 300 continuous as baseload, 750 / 900 as 8-hour peaking plant
Primary circulation	Forced circulation
NSSS Operating Pressure (MPa)	~0.1
Core Inlet/Outlet Coolant Temperature (°C)	525 / 590
Fuel type/assembly array	Molten salt fuel within vented fuel tubes in a conventional hexagonal array fuel assembly
Number of fuel assemblies in the core	451
Fuel enrichment (%)	Reactor grade plutonium
Core Discharge Burnup (GWd/ton)	120 - 200
Refuelling Cycle (months)	Refuelling at power
Reactivity control mechanism	Regulation through reactor temperature coefficients; refuelling mechanism; shutdown with boron carbide assemblies
Approach to safety systems	Eliminate hazards, inherent safety features, dedicated safety systems
Design life (years)	60
Plant footprint (m^2)	22 500
Tank height/length/width (m)	6 m diameter, 10 m height
Reactor tank weight (metric ton)	~500
Seismic design (SSE)	At least 0.3g
Fuel cycle requirements / approach	Fuelled from higher actinides from spent oxide fuel; spent SSR-W300 fuel processed in the similar way
Distinguishing features	Molten salt fuel in conventional fuel assemblies; burning of nuclear waste; thermal energy storage to allow operation as peaking plant; very low cost.
Design status	In transition from conceptual to engineering; Canadian vendor design review in progress

1. Introduction

The Stable Salt Reactor Wasteburner (SSR-W300) 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. The steam cycle is identical to that in traditional fossil fired steam plant and it can be operated completely independently of the nuclear plant.

2. Target Application

The SSR-W300 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 low carbon flexible generation to complement to intermittent renewable energy sources.

3. Main 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 radically reduced to a level where they can be considered practically to have been eliminated.

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-W300 on smaller sites with smaller emergency planning zones because of the large reduction in potential for off-site radioactive releases.
- 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 equivalent to 300 to 450 MW(e) continuous generation.
- 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 at typical subcritical conditions where there is already a wealth of experience in flexible operation of generating plant. The design can be based upon a single larger turbine of up to 900 MW(e) or multiple smaller turbines, as best fits utility need for the local electricity grid needs.

(c) Reactor Core

The core is made up of hexagonal fuel assemblies made up from 271 fuel pins which sit into a diagrid structure located at the bottom. Hexagonal plates at the top the assemblies sit together and form part of the reactor 'lid'. The core made from 12 hexagonal 'rings' (counting the centre assembly as the first 'ring') and a partially filled 13^{th} ring to give a nominally cylindrical shape.

The assemblies are constructed from Alloy-91 steel, which has been investigates for use in traditional LMFR designs and for which there is a wealth of thermal creep data from the conventional power industry. The tubes are partially filled with fuel salt but retain a 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 under normal operating conditions, eliminating the potential for control failures that can lead to an increase in the core reactivity.

The excess reactivity that exists at start up, prior to the fuel salt temperature rising to design levels, is controlled using the shutdown rods. Initial criticality is achieved with these partially withdrawn and they are then fully withdrawn during the remaining power ascension.

Shutdown is achieved with boron control mechanisms which are lowered into the core for a routine shutdown but will drop into the core on demand to achieve a rapid shutdown. 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 followed by the injection of neutron absorber into dedicated channels that is driven also by the temperature rise.

(e) Fuel System

The fuel is 45% KCl/~25% reactor grade Pu and associated minor actinide trichlorides /~30% UCl_3 and lanthanide trichlorides. It is redox stabilised to render it noncorrosive 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 Coolant System

The reactor core sits in a tank that is supported on a refractory lining that sits within the foundations for the plant. The reactor is a loop type reactor, although provision is made for any coolant leakage from the coolant loops to be inherently drained back to the reactor tank. The coolant salt is 42% ZrF_4 /58%KF melting point ~420°C

The heat is transferred in the Main Heat Exchangers to the thermal storage coolant which 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.

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-W300 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 failure of forced circulation. Copper emergency heat removal (EHR) heat exchange blocks are located in a band around the reactor tank at the level of the upper part of the core. The EHR heat exchange blocks provide a route for passive heat removal of decay heat to the ambient air, as an ultimate heat sink. Decay heat removal can be maintained indefinitely with this approach, even in the event of a degraded core.

(c) Emergency Core Cooling System

The unpressurised nature of the reactor coolant system, and the inherent provisions to ensure that leakage is returned to the reactor tank, and the spaces available for coolant from any reactor tank leaks to take up is limited, means that the traditional make up function of an emergency core cooling system is not required. Decay heat removal is carried out as described in (b) above.

(d) Electrical Supplies

There is no requirement for motive power electrical supplies to ensure that safety functions are provided. The only safety grade electrical supplies are required to support the instrumentation required to monitor the plant during accident sequences. These can be supported with battery backed supplies to cover an initial mission time and then off-site grid supplies are then restored or alternative standby generation can be brought to site.

(e) Reactor Containment System

There are no pressurised systems or components within the reactor building so there is a limited formal requirement for the containment system to contain pressure. The reactor tank and the reactor coolant loops are entirely contained within the envelope of a containment structure that sits between them and the concrete biological shield. The biological shield also serves to protect missile hazards. The thermal storage coolant lines and the air ducts for the EHR heat exchange blocks are arranged such that the containment boundary is effectively at the heat exchange surfaces where these systems take heat from the primary coolant. The use of matrix style heat exchangers ensures that even at these points multiple containment barriers remain in place.

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 Layout Arrangement

The functional separation of the nuclear island from the steam generation and turbine system by the large 'solar salt' stores minimised claims on nuclear safety from the steam system. In safety terms, the availability of the thermal storage salt system is equivalent to claims made on availability of off-site power or cooling from an adjacent body of water.

The principal safety claim for decay heat removal is on the passive air cooling provided by the emergency heat removal system.

The licensed nuclear site is therefore rather small, with the large solar salt tanks and steam turbine systems outside the licensed site.

8. Design and Licensing Status

A conceptual design has been completed and is currently progressing through the Canadian Nuclear Safety Commission Vendor Design Review.

9. Fuel Cycle Approach

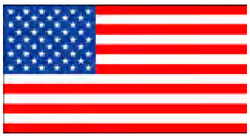
New fuel for the initial core is derived from spent oxide fuel (as discharged from LWRs, CANDUs and AGRs) that is processed in the WATSS facility. Fuel that is discharged from the SSR-W300 can also be processed in the WATSS facility so an operational reactor will only require ongoing additional fissile material from the spent oxide fuel source to provide a make-up because its conversion ratio will be less than 1. Aside from the final core used for an SSR-W300 at the end of life, the net effect is to convert the plutonium and minor actinides in the original spent oxide fuel to fission products.

10. Waste Management and Disposal Plan

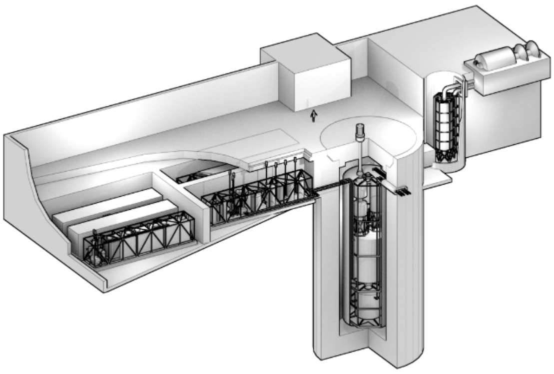
The SSR-W300 burns most of the common nuclear spent fuel wastes. The residual fuel waste will be transported to geological repositories when the plant is in the decommissioning stage.

11. Development Milestones

2017	Conceptual design completed and Canadian Nuclear Safety Commission Vendor Design Review commenced
2018	Partnership with New Brunswick Power established to construct the power plant and the fuel facility at the Point Lepreau site in New Brunswick, Canada
2020	Partnership with Canadian Nuclear Laboratory established to support the fuel development; reactor design improved to provide even better capabilities to burn nuclear waste



Liquid Fluoride Thorium Reactor (Flibe Energy, United States of America)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Flibe Energy, Inc., United States of America
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
NSSS Operating Pressure (primary/secondary), MPa	Ambient
Core Inlet/Outlet Coolant Temperature (ºC)	500 / 650
Fuel type/assembly array	LiF-BeF$_2$-UF$_4$
Fuel enrichment (%)	Not applicable, uses U$_{233}$ derived from Thorium
Refuelling Cycle (months)	Continuous refuelling from U$_{233}$ produced in blanket
Reactivity control mechanism	Negative temperature coefficient; control rod insertion
Approach to safety systems	Passive
Distinguishing features	Complete consumption of thorium resource for energy generation
Design status	Conceptual Design

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 Th$_{232}$ in the blanket region is ultimately converted to U$_{233}$ 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 fluorides fuel-salt streams respectively. Utilizing thorium fuel in a thermal neutron spectrum, the reactor can extract almost all the energy content thus assuring practically unlimited thorium resources and the associated insignificant basic fuel costs.

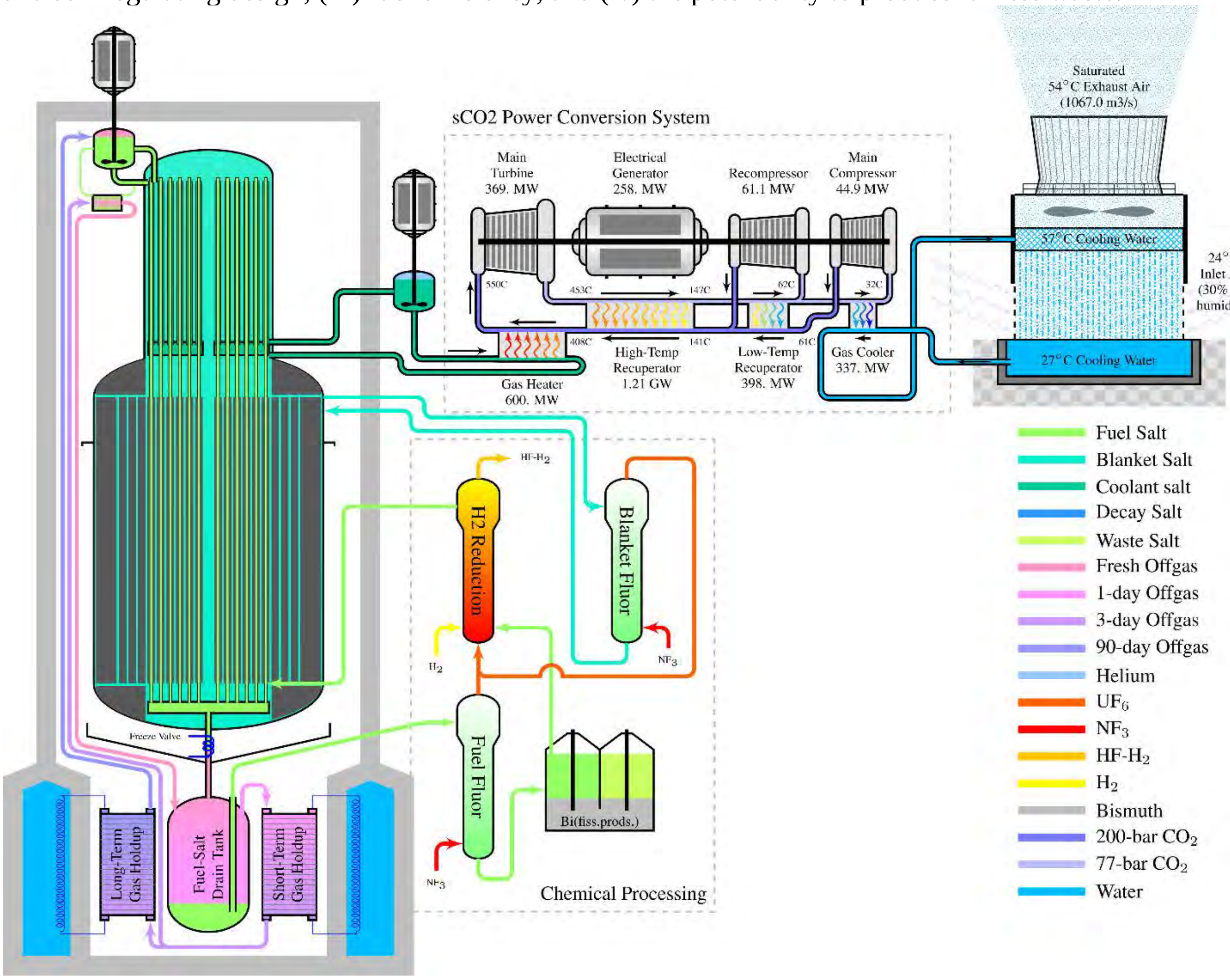
2. Target Application

Develop a power-generating nuclear reactor that will produce electrical energy at low cost by efficiently consuming thorium.

3. Main 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) inherent safety, with a no meltdown and non-pressurized core; (ii) simplicity, to have an intrinsically stable and self-regulating design; (iii) fuel efficiency, and (iv) the potentiality 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 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 forms 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 that a subcritical configuration can be established in the drain tank.

The internal graphite structures need to be replaceable since they are subject 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 Th_{232} via nuclear beta decay, first to protactinium-233 (Pa_{233}) and later to U_{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, 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 Li_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 Layout 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. Fuel Cycle Approach

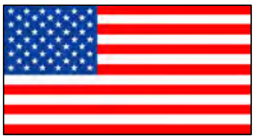
The LFTR two-region core facilitates feed and breed and it utilizes a closed fuel cycle based on thorium.

8. Waste Management and Disposal Plan

LFTRs have the potential to produce far less waste than LWRs along the entire fuel cycle and process chain, from ore extraction to nuclear waste storage. LFTR technology can also be used to consume the remaining fissile material available in spent nuclear fuel stockpiles around the world and to extract and resell many of the other valuable fission byproducts that are currently deemed hazardous waste.

9. Development Milestones

October 2015	EPRI-funded study of LFTR design published
July 2018	DOE award announcement for advanced fluorination development work



KP-FHR (Kairos Power, United States of America)

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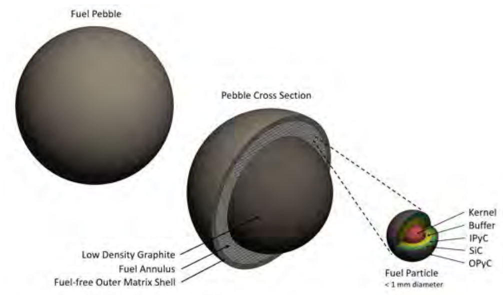
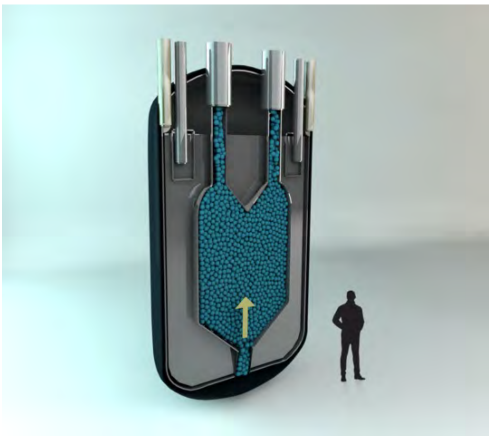




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Kairos Power, LLC, United States of America
Reactor type	Modular, pebble bed, high temperature, salt-cooled reactor
Coolant/moderator	Li_2BeF_4 / graphite
Thermal/electrical capacity, MW(t)/MW(e)	320 / 140
Primary circulation	Forced Circulation
NSSS Operating Pressure (primary/secondary), MPa	< 0.2
Core Inlet/Outlet Coolant Temperature (ºC)	550 / 650
Fuel type/assembly array	TRISO particles in graphite pebble matrix / pebble bed
Fuel enrichment (%)	19.75
Fuel cycle (months)	Online refueling
Main reactivity control mechanism	Control elements, boron
Approach to safety systems	Passive
Design life (years)	20 (vessel), 80 (plant)
Seismic design (SSE)	Target: contiguous USA
Fuel cycle requirements / Approach	Once-through Uranium
Distinguishing features	Longer than 72-hour coping time for core cooling without AC or DC power, or operator action
Design status	Conceptual design in progress

1. Introduction

Kairos Power is a mission-driven engineering company focused on the delivery of a clean, affordable and safe energy solution through the integrated design, licensing and demonstration of Generation IV advanced reactor technology.

The Kairos Power fluoride salt-cooled high temperature reactor (KP-FHR) is a novel advanced reactor technology that leverages TRISO fuel in pebble form combined with a low-pressure fluoride salt coolant. The technology uses an efficient and flexible steam cycle to convert heat from fission into electricity and to complement renewable energy sources.

2. Target Application

The KP-FHR aims to be cost competitive with natural gas and to provide a long-term reduction in cost. This advanced reactor technology is designed for high availability and performance with low maintenance and lifecycle costs, providing dispatchable power that improves grid resiliency and security. In combination with variable renewables, this technology can create a path to a truly clean energy system.

3. Main Design Features

(a) Design Philosophy

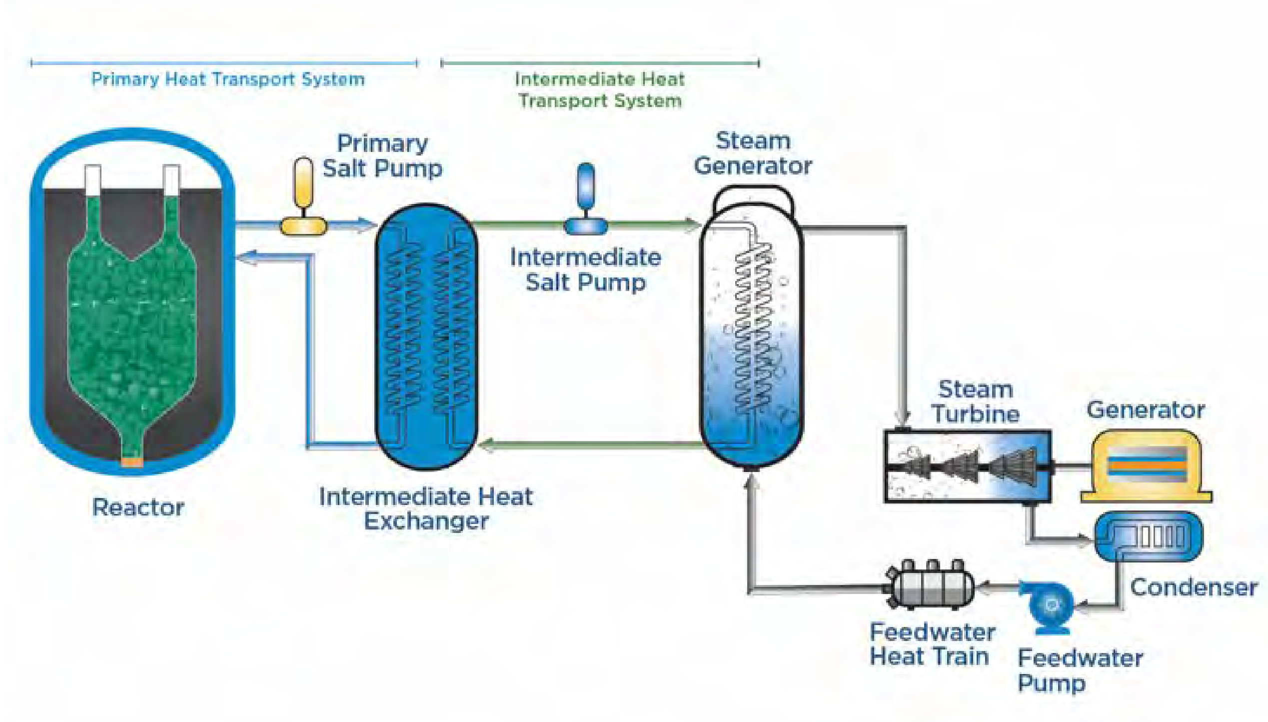
The fundamental design concept is the combination of Tri-structural Isotropic (TRISO) particle fuel coupled with molten fluoride salt coolant (2LiF:BeF_2, Flibe). This combination results in a high temperature, low-pressure reactor with robust, passive safety systems. In addition to robust, inherent safety, the design also reduces reliance on high-cost, nuclear grade components and structures and leverages conventional technologies to lower capital costs.

(b) Reactor and Core-unit

The KP-FHR reactor is a pebble-bed type core with spherical fuel and moderator pebbles forming the active region of the core. The core design utilizes a cylindrical geometry with a graphite side-reflector and bottom and top graphite structures. The core internal structures enable reactivity control and shutdown elements.

(c) Power Conversion System

The power conversion system will leverage prior technology demonstration of solar nitrate salt steam generators and conventional power conversion technologies for the balance of plant.



(d) Reactivity Control

Reactivity control for reactor maneuvering and non-accident events is provided by control elements that insert into the graphite reflector surrounding the pebble bed core. Reactivity control during accident events is provided by shutdown elements that insert directly into the pebble bed. The shutdown elements are gravity driven and are released by the reactor protection system. Both the shutdown and control elements consist of a composite structure of neutron absorber material made of natural B_4C in an inert gas with SS316H cladding. The shutdown and control elements fail safe (insert) on a loss of power.

(e) Fission Product Retention

The fundamental safety strategy for the KP-FHR is rooted in the retention of fission products within the TRISO layers of the fuel particle design with additional retention in the flibe salt coolant. The TRISO layers are credited for providing a ‘functional containment’ for meeting design basis accident dose limits.
The silicon carbide (SiC) coating on the TRISO particles is the primary fission product barrier, while the pyrolytic carbon layers and matrix act as secondary barriers for trapping or impeding the transport of fission products and protecting the integrity of the SiC layer.

(f) Fuel Handling Approach

During operation, fuel and moderator pebbles circulate through the core in a pebble handling system that supports online refueling, recirculation, and discharge of pebbles to spent fuel storage, as needed.

(g) Cooling System

Normally, the Flibe primary coolant transfers heat to an intermediate heat transfer loop which uses a nitrate salt coolant. The intermediate loop transfers heat to a third loop which uses a steam Rankine cycle to produce electricity. The KP-FHR also includes a passive system called the Reactor Vessel Auxiliary Cooling System (RVACS) which provides shutdown heat removal for licensing basis events.

(h) Fuel Characteristics

Kairos Power’s reactor uses fully ceramic fuel, which maintains structural integrity even at extremely high temperatures. This fuel will be undamaged to well above the melting temperatures of conventional metallic reactor fuels.

4. Safety Features

The KP-FHR relies on the combination of fuel and coolant as the only safety-related barriers necessary to provide functional containment for all design basis accidents. Additional engineered safety features include the reserve shutdown system and a passive decay heat removal system. The primary safety function of these systems is to limit temperature excursions in the reactor vessel to maintain geometry of the fuel and coolant in the vessel.

(a) Engineered Safety System Approach and Configuration

The reactor system includes a pebble bed core, surrounded by a graphite reflector, contained within a cylindrical 316H stainless steel reactor vessel.

(b) Decay Heat Removal System

Decay heat removal during normal operations and non-accident events is provided by a normal shutdown cooling system that connects directly to the primary heat transport system. Decay heat removal during accident events is provided by a passive reactor vessel auxiliary cooling system (RVACS) located external to the reactor vessel. No core injection is required for inventory makeup nor decay heat removal functions under accident conditions. The RVACS relies on decay heat through thermal radiation and natural convective heat transfer utilizing a thermosyphon concept. The system is a closed system that does not require inventory makeup and relies on rejection to an ultimate heat sink. System valves are fail-safe and do not rely on safety related electrical power or operator action.

(c) Emergency Core Cooling System

Passive safety means that Kairos Power reactors do not require electricity to remove heat from the core after shutting down. Kairos Power reactors have uniquely large safety margins based on the selected combination of fuel and coolant, which allows emergency cooling to be driven by fundamental physics rather than engineered systems.

(d) Containment System

Functional containment in the KP-FHR is provided by the robust intrinsic safety characteristics of the TRISO fuel and flibe coolant to ensure that the health and safety of the public and workers are protected. Multiple additional barriers in the KP-FHR provide defense-in-depth.

5. Plant Safety and Operational Performances

The KP-FHR leverages intrinsic safety characteristics of the fuel and coolant to achieve uniquely large safety margins while lowering capital costs and improving operating economics. The fuel in the KP-FHR is the TRISO particle fuel, which can withstand fuel particle temperatures up to 1600°C. The Flibe is chemically stable and is at low-pressure, with a boiling point of 1430°C, notably lower than 1600°C and yet functionally very high. The combination of extremely high-temperature-tolerant fuel and low-pressure, single-phase, chemically stable reactor coolant removes entire classes of potential fuel-damage scenarios, greatly simplifying the design and reducing the number of safety systems. The intrinsic low pressure of the reactor and associated piping, along with the functional containment provided by the TRISO fuel, enhances safety and eliminates the need for high-pressure containment structures.

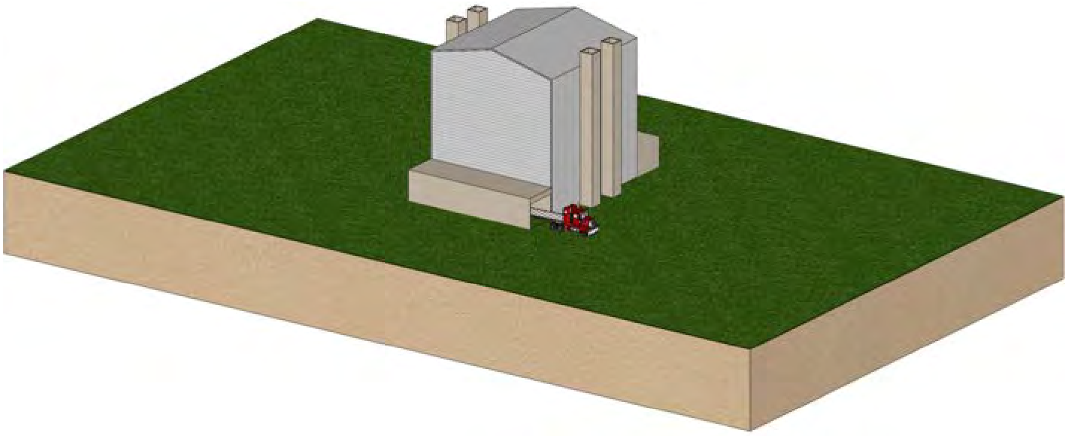
6. Instrumentation and Control Systems

The Kairos Power Instrumentation and Control Systems takes advantage of the inherent safety features of the KP-FHR technology to simplify the reactor protection system configuration and automated actions. The system is sufficiently simplified to allow for deterministic analysis of all design basis events.

The plant control system operates within the protection system established operating envelope as an industrial control system, maximizing automation and continuous health monitoring of the plant.

The I&C Reactor Protection System (RPS) provides protection during steady state and transient power operation and includes the capability to manually or automatically trip the reactor and activate RVACS. The RPS is fully independent of the Plant Control System, which provides overall control during normal operation, startup or shutdown.

7. Plant Layout Arrangement



8. Design and Licensing Status

The KP-FHR is in the pre-conceptual design phase. Kairos Power has initiated pre-application engagement with the U.S. Nuclear Regulatory Commission. Kairos Power has submitted and received safety evaluations for multiple topical reports, two of which the Advisory Committee on Reactor Safeguards has also reviewed. Kairos Power plans deployment of the first KP-FHR before 2030.

9. Fuel Cycle Approach

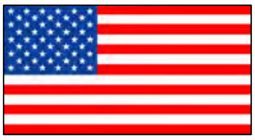
The KP-FHR employs a once-through fuel cycle.

10. Waste Management and Disposal Plan

The used TRISO pebble fuel and a small inventory of 'greater than Class C' waste is packaged in multi-purpose canisters for dry interim storage and subsequent off-site transportation for direct geologic disposal or recycling depending upon national policies. All remaining waste streams from operation and decommissioning qualify for low-level waste disposal.

11. Development Milestones

2018	Pre-Conceptual Design completed
2018	Commissioned R-Lab
2018	Initiation of Pre-Application Review with NRC



Mk1 PB-FHR (UC Berkeley, United States of America)

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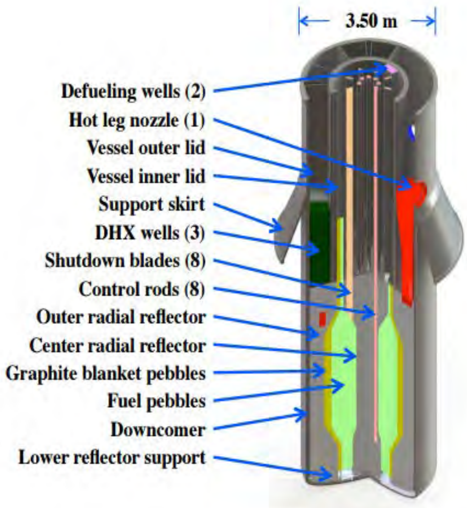
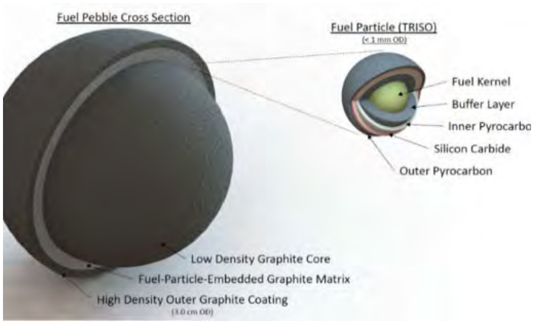




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	University of California, Berkeley, United States of America
Reactor type	Fluoride-salt-cooled high temperature reactor (FHR)
Coolant/moderator	Li$_2$BeF$_4$ / graphite
Thermal/electrical capacity, MW(t)/MW(e)	236 / 100
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.3 / 1.85 (compressor outlet)
Core Inlet/Outlet Coolant Temperature (°C)	600 / 700
Fuel type/ assembly array	TRISO particles in graphite pebble matrix / pebble bed / 470 000
Fuel enrichment (%)	19.9
Fuel burnup (GWd/ton)	180
Fuel cycle (months)	On-line refuelling
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 safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	45 000
RPV height/diameter (m)	12 / 3.5
Seismic design (SSE)	~0.3g
Distinguishing features	Large fuel and coolant thermal margin, high temperature operation
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 (^{7}Li$_2$BeF$_4$). 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. Main 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 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 ^7Li), and good neutron moderation capability.

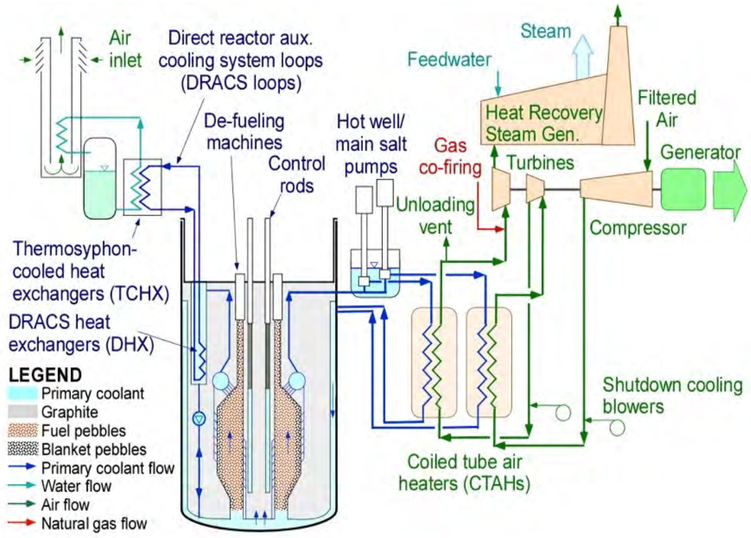
(b) Reactor and Core-unit

The core design incorporates an annular pebble-bed core geometry composed of a homogeneous mix of fuel pebbles adjacent to the center graphite reflector, with a layer of inert graphite reflector pebbles on the outside that reduces the fast-neutron fluence to the outer fixed radial graphite reflector sufficiently for it to last the life of the plant.

The center reflector provides 8 channels for insertion of buoyant control rods, and it also provides flow channels for radial injection of coolant into the pebble core, to provide a combined radial and axial flow distribution that increases the effectiveness of heat transfer from the fuel and results in lower average fuel temperature. The center graphite reflector internals need to be replaced periodically due to radiation damage.

(c) Power Conversion System

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



(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) Fission Product Retention

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 4730 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. The coated particles have the main fission retention function, but the molten salt coolant, primary circuit and building also serve as barriers to release.

(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) Heat Recovery Steam Generator System

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

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.

(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

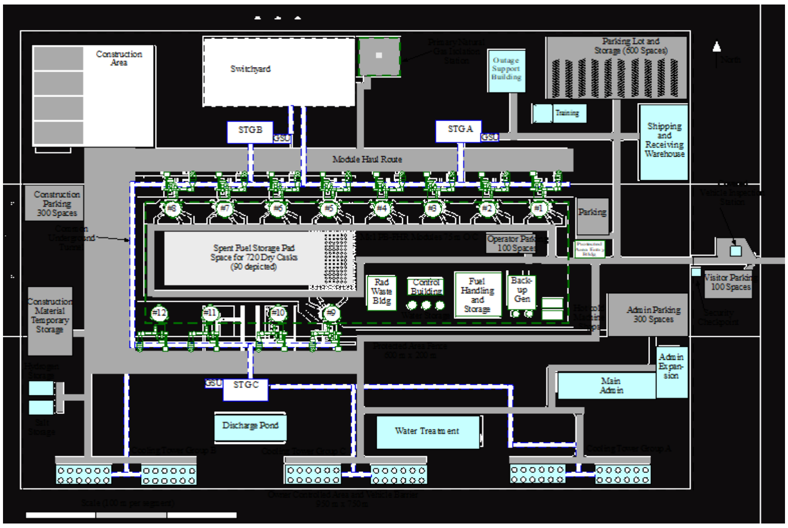
Due to the high thermal efficiency of the NACC system, the steam-bottoming condenser requires only 40% of the cooling water supply that is required for a conventional LWR, for each MWh of base-load generation. As with conventional natural-gas combined cycle (NGCC) plants, this makes the efficiency penalty 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.

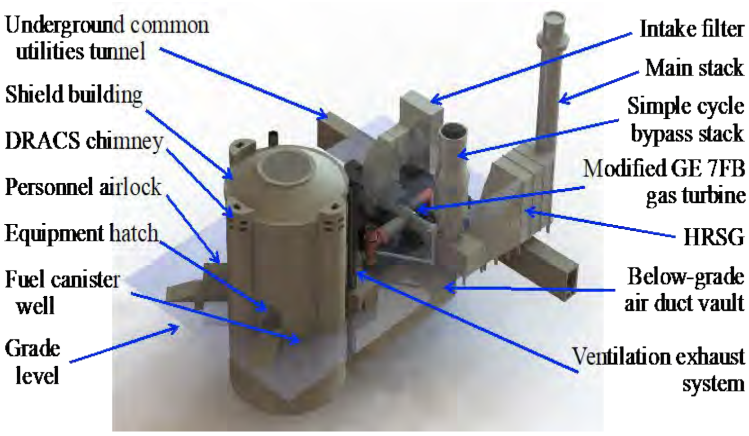
Except during startup and low-power conditions, the PB-FHR operates with constant core inlet and outlet temperatures. Load-following capability is made possible by air bypass flow to respond to rapid load-change transients and turbine inlet temperature control (by bypassing air around the CTAHs) for 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 Layout 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 1200 MW(e) base load and 2900 MW(e) peak power output. Due to the much smaller cooling requirements they do not need to be sited near bodies of water. Population centers tend to be located near bodies of water which means that FHRs can be sited in areas where fewer people want to live. So, rather than attempt to minimize the site footprint, the more important goal is likely to facilitate construction of modules adjacent to operating modules, and to optimize the degree to which some services are shared.

(a) Reactor Building



The Mk1 reactor building and NACC system arrangements supports a multi-module plant configuration 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. Fuel Cycle Approach

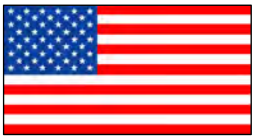
The initial design is based on once through LEU cycle but other thermal spectrum based fuel cycles (U-Th; Pu-Th, Pu burner) as illustrated by HTR coated particle fuel, are in principle possible.

10. Waste Management and Disposal Plan

The fuel pebbles are based on HTR coated particle fuel with its excellent radioactivity containment characteristics. Upon removal from the primary coolant, the spent fuel will require cleaning to remove residual salt and cooling to maintain acceptable fuel temperatures in the gas environment.

11. 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 MCSFR (Elysium Industries, USA)

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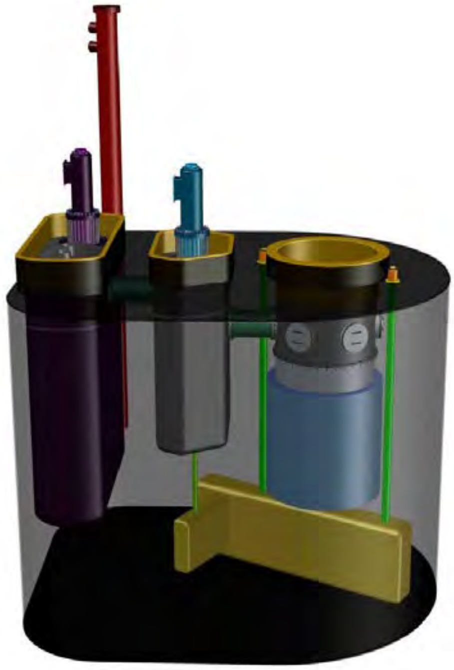


Elysium's modular reactor

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Elysium Industries, United States of America
Reactor type	MSR – Fast Chloride
Coolant/moderator	$NaCl\text{-}XCl_v\text{-}YCl_z\text{-}UCl_{3/4}\text{-}PuCl_3\text{-}FPCl_y$ fuel salt / None
Thermal/electrical capacity, MW(t)/MW(e)	(125 / 50), (500 / 200), (1000 / 400) , (3000 / 1200)
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.1+pump head+hydrostatic / slightly higher
Core Inlet/Outlet Coolant Temperature (ºC)	650 / 750 (Goal: 950)
Fuel type/assembly array	Molten Chloride Salt
Number of fuel assemblies in core	n/a, Full liquid fuel core
Fuel enrichment (%)	10% Pu fissile/(Pu+U total) or ~15% enriched HALEU
Core Discharge Burnup (GWd/tonne)	300-500 (40-60 years), fertile added, fuel never damaged
Purification cycle (months)	480-720
Main reactivity control mechanism	Fuel expansion in/out of core; Fertile fuel addition; Passive fuel draining
Approach to safety systems	Passive to Air Cooling
Design life (years)	Unlimited core, 15-40 for components, 100+ for plant
Plant footprint (m^2)	1/3 size of LWR
RV height/diameter (m)	9/4
Seismic design (SSE)	Tension skirt Lateral snubbers
Fuel cycle requirements / Approach	U/Pu Closed Fuel Cycle SNF/DU/NU (1t/GWe-yr)
Distinguishing features	Fast spectrum, no in-core structure, 60 years fuel life
Design status	Conceptual design

1. Introduction

The MCSFR is a modular configuration and construction reactor. The MCSFR enables the closing of the fuel cycle, while providing reliable, passively-safe, proliferation-resistant, and environmentally-friendly energy (heat/electricity) generation. The fuel is part of the liquid heat transport eutectic fluid with heat directly deposited in the liquid fuel/coolant.

2. Target Application

The MCSFR is designed for mass production for domestic use and export to address global markets for cost-competitive, low-emission electricity, and high temperature process heat (e.g.: H_2, syn-fuel, syn-fertilizer, desalination, district heating, cement, steel, etc). The MCSFR uses Spent Nuclear Fuel (SNF), plutonium (Pu), or depleted uranium (DU) 'waste' with fuel production denaturing and complete consumption (e.g.: US, Canada, UK, Japan, South Korea, etc.), as fuel sources for internal or export applications and developing countries with fuel take-back. Utilities can 'start small', then add heat exchangers (Hx's) to increase capacity without purchasing and licensing a new reactor.

3. Main Design Features

(a) Design Philosophy

Elysium's team designed the MCSFR using attributes from: 1) water reactors, common, low cost coolant (table salt), fuels (nuclear waste), and qualified materials, 2) liquid metal reactors, the low pressure and materials without corrosion concerns, 3) gas reactors, the flow pattern for low containment temperatures, and very high peak temperatures, 4) heat-pipe reactors, passive temperature dependent on/off heat-pipe decay heat removal.

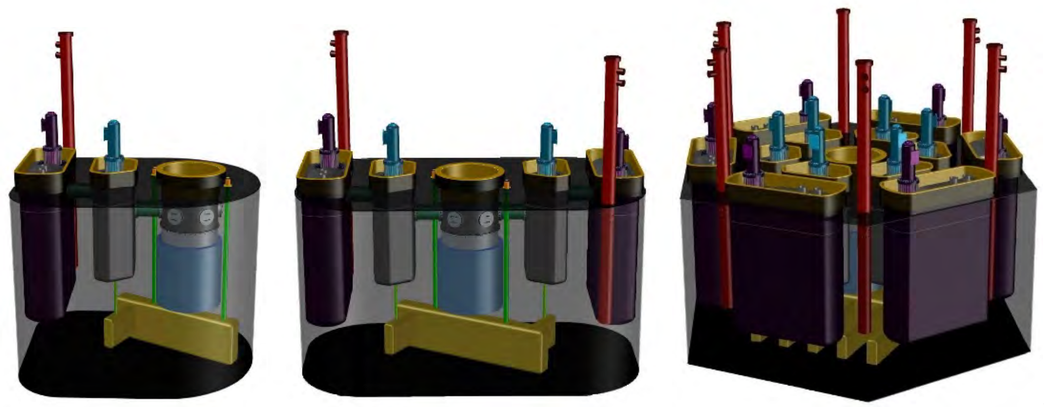
Elysium philosophy includes mitigating public concerns with passive safety, high thermal and fuel efficiency, SNF/Pu waste denaturing and consumption, dramatically increased fuel supply, elimination of coolant reactions with water/air/structure, significantly reduced construction, fuel, and operational costs, in addition reduced proliferation concerns.

(b) Nuclear Heat (Steam) Supply System (NHSS)

The MCSFR has a fuel/coolant loop, clean intermediate salt loop same as fuel salt, without the fuel/fission products (FPs), etc., the steam super-heater (SH) loop all inside containment, with a Loeffler steam boiler, outside of containment. The reactor has 1 to 6 Hx/pump loops for each fuel and clean salt loop. Each of the loops can be used for 6 different heat applications/customers. Loops can operate at variable power up to maximum with fuel cost not a concern at part-load.

(c) Reactor and Core-unit

The core size is minimized to barely maintain criticality with no in-core structures and is near spherical. The RV and all fuel and intermediate component shells are cooled by cold coolant inside, so includes pipe-in-pipe nozzles. The core is pure salt, except for a downcomer shroud near the edge of the core, and the lower RV is the core edge with an ex-RV radial reflector and above core reflector/shield.



40-50% efficiency

10/100 - 500 MW(t) 1000 MW(t) 3000 MW(t)

(d) Power Conversion System and Cogeneration

The intermediate clean salt heats power conversion unit (PCU) saturated steam to SH steam in the SH, with 35% of SH steam to the steam rankine turbine-generator (40-50% efficiency) and 65% of SH steam to the Loeffler boiler. The use of a salt to SH, i.e. a gas Hx allows use of other process heat gasses, especially at 950ºC, on a per Hx basis to allow flexibility of products. With the high outlet temperatures, process heat applications include H2, synthetic fuel and fertilizer, oil/gas recovery and refining, industrial process heat, and cement manufacturing. Other applications include district heating/cooling, thermal storage in chloride salt, and desalination. Use of salt to SH Hx's, dramatically reduces water and potential for transient pressures in containment.

(e) Reactivity Control

Reactivity control is via the negative temperature and void coefficients. As the fuel-salt temperature increases the fuel-salt expands and fissile/fuel salt is squeezed out of the core, reducing power and vice versa. The long-term reactivity adjustments are made by on-line fertile fuel additions. The reactor can also be shut down by tripping pumps to allow draining to criticality safe/passively cooled drain/expansion tanks.

(f) Fuel Characteristics

The fuel-salt is $NaCl\text{-}XCl_v\text{-}YCl_z\text{-}UCl_{3/4}\text{-}PuCl_3\text{-}FPCl_y$, allowing it to contain ~30% total actinide Chlorides with a 10-20% fissile fraction and 99.9% actinide consumption.

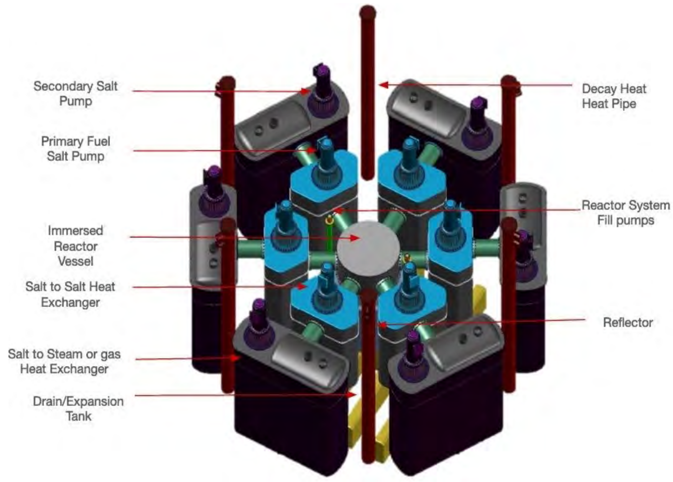
Fissile options include: 1) Preferred-Reactor Grade or Weapons Grade Plutonium (RGPu and WGPu) due to revenue and is denatured with SNF, if needed, 2) high assay low enriched uranium (HALEU), 3) high enriched uranium (HEU) denatured at fuel production site.

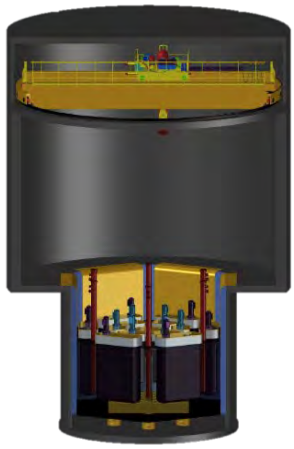
Fertile options include: a) Prefered-LWR/CANDU SNF, b) Depleted Uranium (DU), c) Natural Uranium (NU),

d) residual U from other mining, coal ash, or seawater extraction, e) thorium (Th) combined with >88% U_{238} for denaturing. Pu fissile is added for startup only. HALEU requires continued feed-in declining in enrichment over 5-10 years. Fissile is iso-bred plus enough to counteract fission product poison buildup to prevent needing purification for many decades.

(g) Reactor Vessel (RV)

The RV is low pressure, thin walled stainless steel, with up to 6 nozzles, cooled by cold fuel salt everywhere inside, and submerged in a cooling/shield tank of clean salt for corrosion prevention/cooling outside. The RV near the core is the core outer diameter and is replaceable separately from the upper RV nozzle region. A reflector, if needed, is outside the RV.





(h) Primary Loop

The primary loop is chloride fuel salt. The arrangement is modular, like an HTGR to allow for low temperature containment, yet high temperature for efficiency and process heat applications. Six pipe-in-a-pipes connect in a modular arrangement to up to 6 fuel salt to clean intermediate salt Hx's, with top mounted pumps after the Hx to prevent motor immersion and heating. Radial dimension is ≤ 4m for road shipping.

(i) Intermediate Loop

The intermediate loop is the same salt as the fuel salt without the actinides or FPs, and is at a higher pressure than the fuel salt to ensure any leakage is not of fuel salt outward, inward for dilution/shutdown of the core. The intermediate loop transports heat to the modularly mounted SH Hx, with the intermediate salt pump on top of one end of the SH. Radial dimension is ≤ 4m for road shipping.

(j) Fuel Cycle and Length Approach

On-line fertile fuelling is used. Iso-breeding ratio is ~1.014/yr to offset fission product buildup with a doubling time of 50-60 years with a fuel life of 40-60 years when it is sent to a central facility for partial purification and recycling of all actinides, carrier salt components, including all chlorine. The fuel consumption rate is ~1 ton U / GWe-yr. Fuel is added every ~8 hrs-7 days depending on the power level by dropping a fertile pebble into a perforated basket in the drain tank flow path.

(k) Cooling System

Four different cooling loops are included: the fuel salt, intermediate salt, the power/process heat loop, and the heat sink loop. With the high temperature and very low fuel cost, dry cooling is an economic alternative.

(l) Proliferation Considerations

The MCSFR never contains weapons grade materials. Fuel is not removed for 40-60 years. WGPu or HEU is denatured at the single fuel production facility. Purification at 60 years never separates Pu from U, Cs and Sr. This low separation of FPs reduces recycling cost. Only relatively short-lived, ~100 years to below U background, FPs are removed as waste.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

Passive features include the large mass and heat capacity of the fuel, intermediate, and tank salt to slow heat up. Pumps constantly fill the primary system from the bottom and can be tripped passively by high or low salt temperatures, or by the reactor operator. When pumps stop, fuel salt immediately starts draining from the RV and Hx's to a passively shut down high neutron and heat leakage drain tank with external tank salt cooling. The RV and Hx's are also tank salt cooled.

(b) Decay Heat Removal System

Passive on/off heat pipe decay heat HX's remove decay heat to air from the tank salt heat mass, when temperatures exceed the vaporization point, while also shutting down preventing freezing of the fuel salt in the drain tanks when decay heat decreases to allow faster plant recovery.

(c) Containment Function

The MCSFR has 3 intrinsic and 3 physical containment barriers to leakage: chemical binding, inward leakage, salt freezing, fuel salt RV and HX shells, shield/cooling tank shell, underground low pressure containment cylinder. Containment is below grade with the top being an airplane shield.

5. Plant Safety and Operational Performances

The MCSFR is designed for passive, fast load/source following. With a closed fuel cycle, no solid fuel manufacturing costs, and extremely low fertile fuel costs or revenue from SNF, fuel cost is very low. Turbine bypass is economically viable for fast load following. Consuming more fuel via bypass operation, i.e., more SNF consumption, is a revenue offset.

6. Instrumentation and Control Systems

The MCSFR operates on load following the by changing the temperature difference. Temperatures decrease as fertile is consumed and FPs build up, which is a signal to add more fertile to increase fissile breeding. Temperature, pressure, level, and flow rate are the primary control systems, with significant peak, or minimum temperature changes passively initiating draining, and average temperature controlling fuel expansion in/out of the core. Elemental, isotopic, and molecular composition are monitored.

7. Plant Layout Arrangement

The plant is a modular configuration, like an HTGR. The RV is in the center, surrounded by short pipe mounted, i.e., modular, salt to salt, and salt to process heat HX's radially around the core in a large salt tank. The reactor is sized to just achieve criticality. Low initial power can use one HX pair, and allow uprating by adding Hx pairs as needed. The reactor is underground for aircraft and security cost reduction.

8. Design and Licensing Status

The conceptual design of pilot fuel production and reactor are in progress.

9. Fuel Cycle Approach

High fissile startup fuel is made in a single fuel production facility and shipped to the new plant. The MCSFR is a closed fuel cycle requiring only 1 tonne SNF, DU, NU fuel/GW(e)/yr for each plant. Feed-in fuel is added daily/weekly, depending on power/burnup rate, for 40-60 years. Every 40-60 years the fuel is removed/sent to one or two recycling facilities. Short-lived FPs are removed. All actinides and long lived Cs and Sr are left together, split in two, and sent out to plants as new/replacement fuel. Chlorine is recycled. Breeding ratio (BR) is ~1.014/yr, with a ~50 yr doubling time. Elimination of solid fuel manufacture dramatically reduces fuel and recycling cost, consumes Pu and SNF waste.

10. Waste Management and Disposal Plan

The waste forms removed are fission gasses (which are allowed to decay), noble metal solid precipitates, 100-year FPs and carrier salt anions. These short-lived FPs can be mined for valuable radio- and stable-isotopes, then are allowed to cool as high temperature salts in surface stored cans cooled by air for ~100 years until stable.

11. Development Milestones

2018-2021	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
2020-2027	Design and building a pilot fuel facility and 10-30 MW(t) pilot plant
2027-2030	Licensing/building Commercial unit

MICRO-SIZED
SMALL MODULAR REACTORS

Energy Well™ (Centrum výzkumu Řež s.r.o., Czech Republic)

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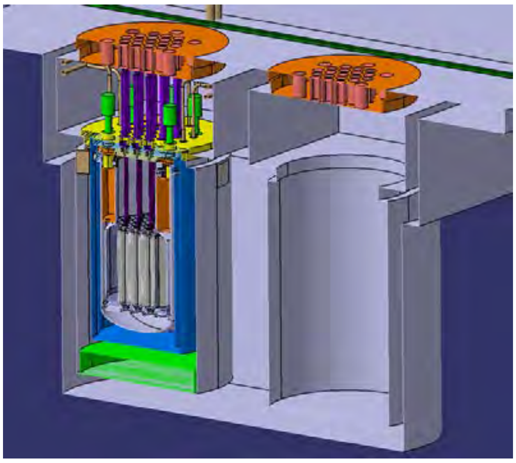
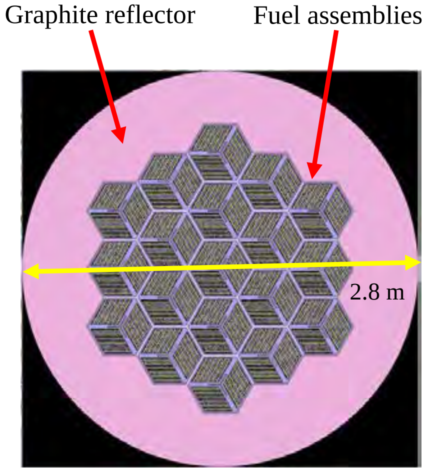




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Centrum výzkumu Řež, Czech Republic
Reactor type	Fluoride High Temperature reactor, pool type
Coolant/moderator	Molten Salt FLiBe
Thermal/electrical capacity, MW(t)/MW(e)	20 / 8
Primary circulation	Forced (mechanical pumps)
NSSS Operating Pressure (primary/secondary), MPa	Atmospheric pressure
Core Inlet/Outlet Coolant Temperature (°C)	650 / 700
Fuel type/assembly array	TRISO
Number of fuel assemblies in the core	19
Fuel enrichment (%)	15
Core Discharge Burnup (GWd/ton)	70
Refuelling Cycle (months)	84
Reactivity control mechanism	Control rods
Approach to safety systems	Active / Passive
Design life (years)	Not defined
Plant footprint (m^2)	< 4000
RPV height/diameter (m)	6 / 3
RPV weight (metric ton)	< 100
Seismic Design (SSE)	Yes
Fuel cycle requirements / Approach	Once through; no onsite refuelling; replace reactor approach
Distinguishing features	Passive decay heat removal, transportable reactor,
Design status	Pre-conceptual design

1. Introduction

Energy Well is a Fluoride High temperature micro Reactor of 20 MW(t) under development using unique knowledge in the Czech Republic on molten salt technologies. The project is under grants from the Ministry of Industry and Trade. The purpose of the project is to develop an advanced, inherently safe low-power high-temperature reactor. The design is mainly intended for remote areas as a long-term source of electrical energy and heat for island networks. Therefore, the reactor and associated power plant and/or a heat plant must meet the following requirements, among others: 25 MW(t) maximum power; transportable; long fuel cycle; no onsite refuelling; fuel enrichment < 20%; and thermal efficiency > 40%.

2. Target Application

Energy Well is focusing on operation both in remote and in populated areas, focusing on production of electricity, heat and hydrogen as a means of energy storage. The purpose is to provide a clean stable energy source in synergy the large-scale nuclear power reactors, heating plants and the renewable sources of energy.

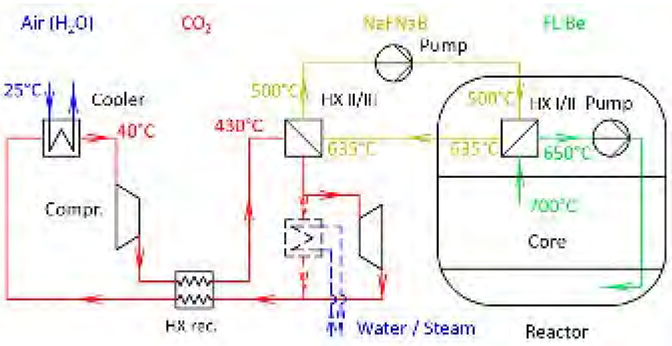
3. Main Design Features

(a) Design Philosophy

The 20 MW(t) Energy Well adopts a 7-year fuel cycle, low power density, high use of passive safety and simplicity. It is designed to be transportable with fresh or spent fuel so that no refuelling on site is required.

(b) Power plant concept

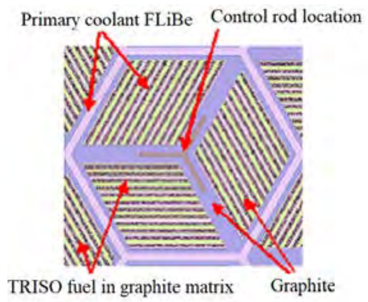
The power plant includes three cooling circuits. Liquid fluoride salts are used as a heat transfer medium (FLiBe, $NaBF_4$) in the primary and secondary circuits. Carbon dioxide in a supercritical state (sCO_2) is used in the tertiary circuit. The tertiary circuit considers an Ericsson–Brayton-based cycle optimized configuration for transformation of the heat to electric power. The primary circuit removes the heat generated in the core of the reactor, while the secondary circuit separates the active primary and the high-pressure tertiary circuit while ensuring the heat transfer.



Process Flow Diagram of Energy Well reactor with simple heat recovery system

(c) Reactor core and fuel

The reactor core design contains 19 hexagonal fuel assemblies with TRISO fuel. The current fuel assembly design (figure to the right) and core arrangement (see previous page) are shown. Alternative fuel assembly designs are also studied.



Top view of one fuel assembly

(d) Primary cooling circuit

Energy Well is a pool type reactor with molten salt FLiBe as the primary coolant. The primary cooling circuit includes the following main parts: the reactor core with top mounted control rods, reactor vessel, graphite reflector, core supporting plate, flow skirting, six primary heat exchangers molten salt/molten salt, and two reactor vessel top flanges.

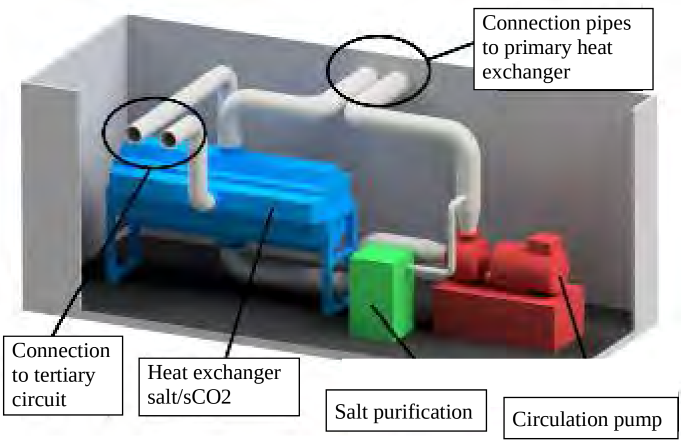
The core support plate and the graphite top reflector have holes to allow the molten FLiBe to flow in each fuel assembly. The molten salt flows upward through the core and then enters the primary heat exchangers located on the periphery of the reactor vessel above the core. The whole primary circuit is placed inside a transport container that allows the reactor to be shipped on road or rails with fresh fuel or spent fuel. Helium is also foreseen as a cover gas above the FLiBe molten salt level in the reactor vessel.

The control of the reactor power during normal operations is ensured by control rods, with a Y-shape, placed in the middle of the fuel assemblies. The fall of all control rods and a second independent passive system are foreseen during emergency shut down of the reactor. The reactor will operate at either 100% (nominal power) or 0% (shut down). The removal of decay heat shall be entirely passive through the wall of the reactor vessel, through the wall of the container to the air in the reactor pit and then to the environment.

(e) Secondary cooling circuit

The secondary circuit physically separates the primary circuit from the tertiary circuit and creates a pressure barrier in case of leaks in the salt/salt exchanger between the primary and the secondary circuit. The secondary circuit comprises 3 main components: (i) salt/salt heat exchanger; (ii) salt/sCO2 heat exchanger; and the circulation pump.

The heat exchangers create the interface of circuits, and the pump ensures the required mass flow of secondary salt NaBF4 to recover the thermal power out of the primary circuit. The secondary circuit is equipped with auxiliary systems that include molten salt refilling, salt purification system, expansion volume to cope with molten salt volume change.



The solidification temperature of the salt is a of key parameter. Therefore, the inlet temperature of sCO_2 includes a margin to avoid solidification of the salt. The '$NaBF_4$' salt is used for the secondary circuit with 384°C solidification temperature. Alternatively, LiF-BeF_2 with a solidification temperature of 455°C, FLiNaK with a solidification temperature of 454°C, KCl-$MgCl_2$ with a solidification temperature of 426°C or KF-ZrF_4 with a solidification temperature of 390°C (FluZirK) could be used. The secondary circuit is designed to be independent, with the components placed in a 20-inch transport container.

(f) Tertiary circuit

The function of tertiary circuit is for the conversion of heat to electric energy with the sCO_2 as working fluid. The sCO_2 technology was identified as the most compatible with this type of SMR. The main benefit is a higher thermodynamic efficiency due to high temperature through compression close to the critical point (7.38 MPa, 30.98°C), or even in the liquid phase. As a result, the requirements of the compressor are lower, and the intercooling is not required. On the other hand, the limitations of the sCO_2 circuits lie in their technical 'immaturity'. The change of media properties significantly complicates the design of components. A range of the experimental facilities with a power between 100 kW and 10 MW are being built around the world in order to verify this technology.

There are dozens of possible sCO2 cycles layouts. Based on preliminary studies, the recompression Brayton cycle with heat regeneration was selected for the Energy Well system as reasonable compromise between the complexity and efficiency of the cycle. Relatively high thermodynamic efficiency of the cycle of 42.71% was reached for the nominal operational conditions.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

The design of Energy Well reactor has a high focus on passive safety and simplicity. The main safety features of the reactor include: Atmospheric pressure in primary and secondary circuits; Primary circuit is underground; and the use of natural circulation and passive safety systems.

(b) Decay Heat Removal System

Passive residual heat removal from the primary circuit through the container is adopted in case of loss of flow. In the initial phase of event, the high heat capacity of molten salt will allow the heat output to accumulate in the primary coolant until the heat losses exceed the residual power. Maximum fuel temperature reaches 730°C.

(c) Containment System

The TRISO fuel envelope is the first barrier to prevent the spread of fission products. Depending on the design, the fuel assembly could be considered as a barrier since the TRISO fuel is enclosed in a thick graphite matrix. The reactor vessel, the transport container, the pit (together with the maintenance room shielding ceiling) and the reactor building are additional barriers of the containment system.

(d) Reactivity control

The primary safety system ensuring reactivity control are the control rods with a set scram signal. Scram is activated when the neutron flux or other technological parameters in the core are increased. During normal operation, control rods are kept in operational position by magnets that are power fed. In case of deviation from normal operation parameters, the Limiting System (LS) regulates the reactor power. The capsule melts after reaching a threshold temperature and the absorber ($NaBF_4$) is released in the active zone. This is a passive safety system. Additional B_4C control rods can be deployed for emergency shut down.

5. Plant Safety and Operational Performances

Air or water are foreseen as heat sink for the facility. The facility is foreseen to be operated at constant power and the refuelling shall be perform off-site. This shall minimize the number of operators on site.

6. Instrumentation and Control Systems

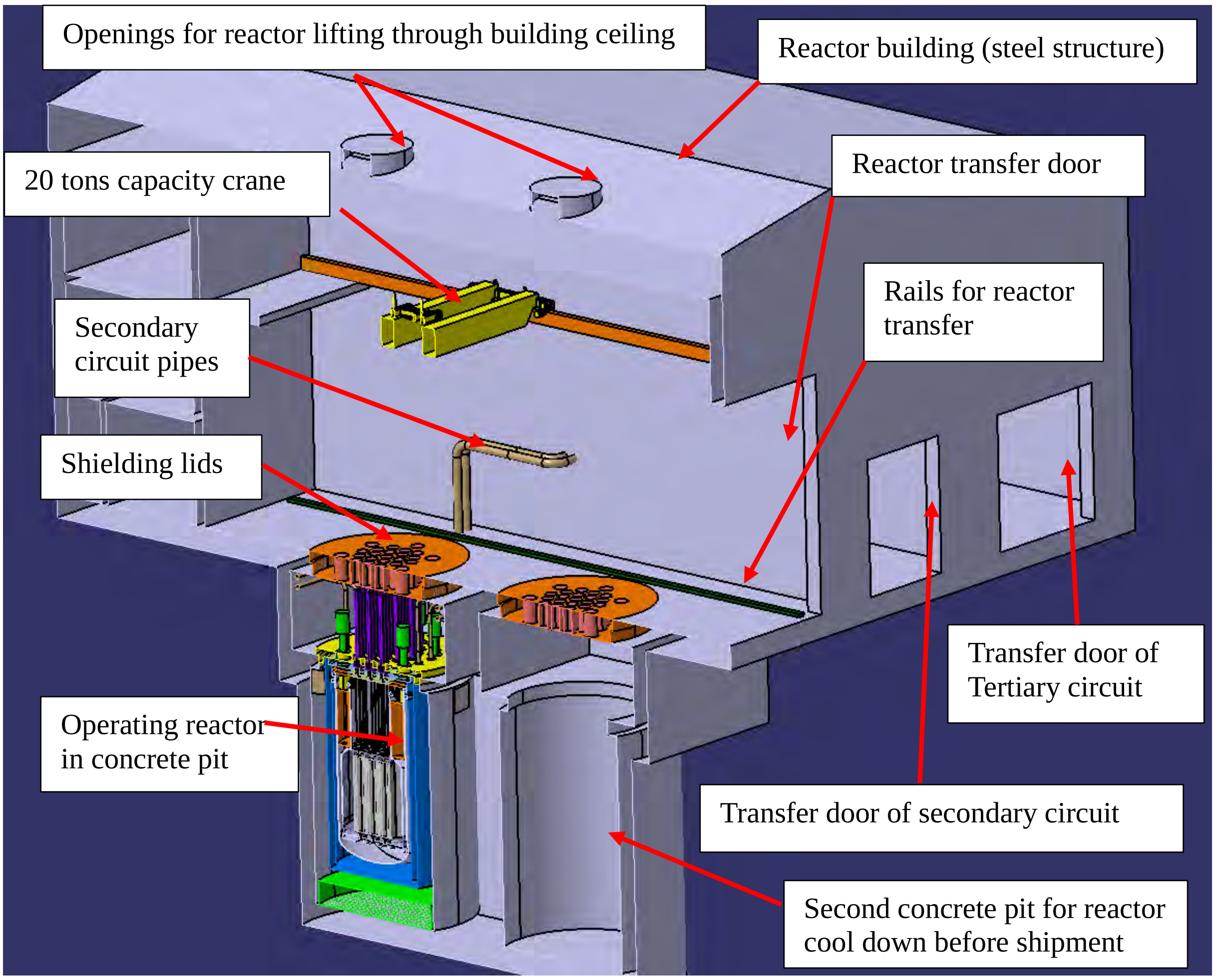
The I&C system is currently under development and being tested on the experimental loops currently operating in the Centrum výzkumu Řež s.r.o.

7. Logistic and transport

One of the key advantages of the Energy Well concept is the transportability to places, where electricity and/or heat production is needed. The secondary and tertiary circuit are designed to fit in ISO containers. At the end of 7-year fuel cycle, the first reactor module remains in its pit for 1 to 2 year(s) to cool down and reach the specifications needed for shipping back to the manufacturing site. Another reactor module with fresh fuel is installed in a second pit and connected to the secondary cooling circuit. The dose rates limits for transport container with spent fuel are < 2 mSv/h on the container surface and < 0.1 mSv/h at a 2 m distance from container surface.

8. Plant Layout Arrangement

The Energy Well reactor building layout is shown in the figure below. The primary, secondary and tertiary circuit are located in a common steel building. The three circuits are located in separate room with separate transfer doors. In the primary circuit room are located two concrete pits. In the first pit is located the operating reactor while the second pit is used to cool the reactor with spent fuel before its shipment for refuelling. A trolley on rails is used to transfer the reactor in its container between the reactor room and outside the building. A mobile crane, temporary placed outside the building is used for lifting operations through openings in the building ceiling. In the reactor room, an overhead crane is used to handle shielding lids and the upper components of the reactor (pump motors, control rods actuator).



3D view of Energy Well reactor building

9. Design and Licensing Status

As of 2020 the Centrum výzkumu Řež is preparing a basic design of the Energy Well reactor including experimental tests. Build an integral testing facility is planned. The Centrum výzkumu Řež is in close coordination with the State office for Nuclear Safety.

10. Fuel Cycle Approach

Energy Well is using the TRISO fuel embedded within a graphite matrix formed in plates. The campaign length is 7 years, after which the fuel will be transported in the reactor container to the refuelling plant.

11. Waste Management and Disposal Plan

The waste management and disposal plan are still under development.

12. Development Milestones

2010 - 2017	Basic research in the neutronics, thermohydraulics and material compatibility in regards to the FLiBe salt.
2017-2020	Pre-conceptual design phase and technology validation
2020 - 2025	Basic design
2025 - 2030	Experimental verification using an integral test facility
2030 - 2040	Finalization for fabrication

MoveluX (Toshiba Corporation, Japan)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Toshiba Energy Systems & Solutions Corporation, Japan
Reactor type	Heat-Pipe cooled and calcium-hydride moderated reactor
Coolant/moderator	None (Sodium heat-pipe cooled) / Calcium hydride (CaH_2)
Thermal/electrical capacity, MW(t)/MW(e)	10 / 3 - 4
Primary circulation	Natural
NSSS Operating Pressure (primary/secondary), MPa	0.1 / 0.3
Core Inlet/Outlet Coolant Temperature (ºC)	Heat pipe: 680 / 685 Heat exchanger: 450 / 680
Fuel type/assembly array	Silicide (U_3Si_2) / Hexagonal
Number of fuel assemblies in the core	177
Fuel enrichment (%)	4.8 - 5.0
Core Discharge Burnup (GWd/ton)	1.0
Refuelling Cycle (months)	Continuous
Reactivity control mechanism	Lithium Expansion Module
Approach to safety systems	Active / Passive
Design life (years)	10 - 15
Plant footprint (m^2)	100
RPV height/diameter (m)	2.5 / 6.0
RPV weight (metric ton)	TBE
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Either once-through or closed fuel cycle scheme depending on country's fuel cycle policy
Distinguishing features	Use inherent passive natural principles for reactor shut down by moderator material property and decay heat removal from the surface of the reactor vessel by natural circulation of air
Design status	Conceptual design

1. Introduction

MoveluX, Mobile-Very-small reactor for Local Utility in X-mark, is a 10 MW(t) class multi-purpose micro reactor. A heat-pipe is used as a primary core cooling that provides passive safety as well as system simplification. MoveluX uses low enriched uranium fuel of less than 4.99 wt% that improves nuclear security and non-proliferation. Moderator material is required to reduce the core size. In addition, high temperature operation is essential for the multi-purpose micro reactor. Therefore, calcium-hydride (CaH_2) capable of operating at up to 800ºC is adopted for the moderator material.

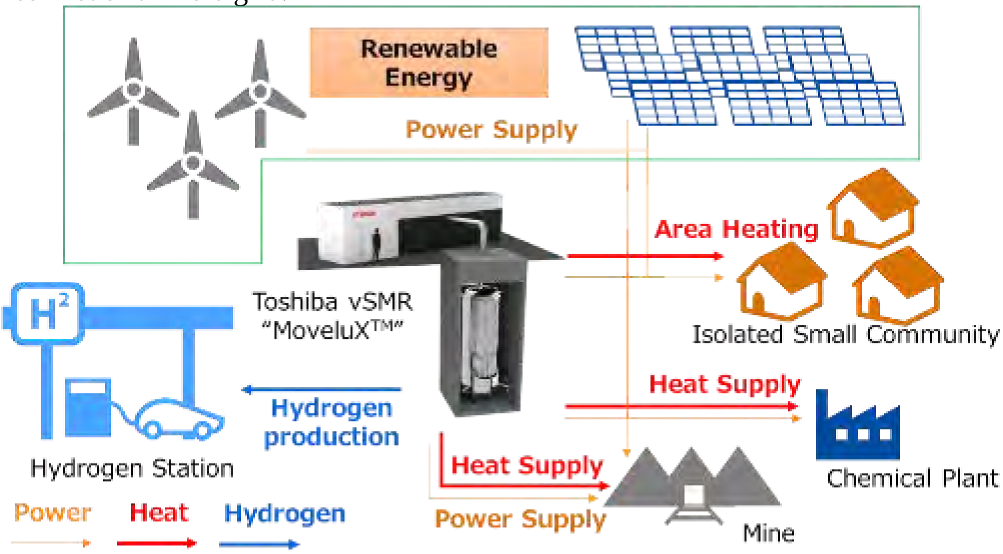
2. Target Application

The MoveluX reactor system is a multipurpose energy source that can be used to produce electricity, hydrogen and high temperature heat. Since the reactor system can be installed in remote sites, the heat can be provided for chemical plants and steel mills. When it is used as a power plant it can be used as a base load power source on small grids, possible in combination with renewable energy sources. Since MoveluX generates around 3-4 MW(e) and it can also be used for off-grid applications in remote places.

3. Main Design Features

(a) Design Philosophy

The MoveluX reactor system is designed as a multi-purpose energy source that includes off-grid and micro-grid electricity, high temperature heat source, hydrogen production and so forth. Designed to produce 10 MW(t), it can be used also for electricity production, possibly varying the output. Therefore, when the MoveluX reactor system is connected to a small/micro grid with load following as required and achieved by the passive reactivity control system. The figure below shows the MoveluX reactor system as a multi-purpose energy source in such a micro grid.

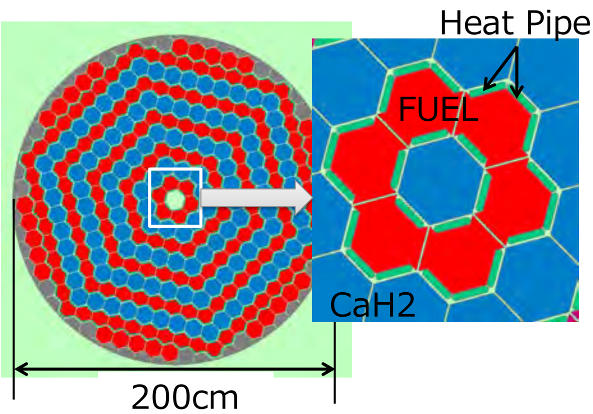


The major provisions of the MoveluX reactor for resource use optimization are as follows:

- Simplified plant design contributes to wastes reduction during operation and decommissioning.
- Low maintenance requirement using no-moving parts component contributes to low maintenance costs/labours and low waste amount.
- Reduced emergency planning zone contributes to accident management burden and/or cost such as for evacuation.

(b) Reactor Core and Fuel

The MoveluX core consist of the fuel, moderator, heat-pipe and control devices. In this core, uranium silicide and calcium-hydride were used as a fuel and moderator materials. The maximum fuel enrichment is set as 4.99 wt%, from the viewpoint of economics and non-proliferation. Fuel components are installed to the core in the fabrication phase and loaded as a lifetime core, i.e. this fuel will not be extracted from the core and therefore fuel handling is not required during operation. The fuel and moderator has 10 cm wide-hexagonal shape except for the sides where provision is made (cut off is made) for the heat-pipe installation.



(c) Primary Circuit

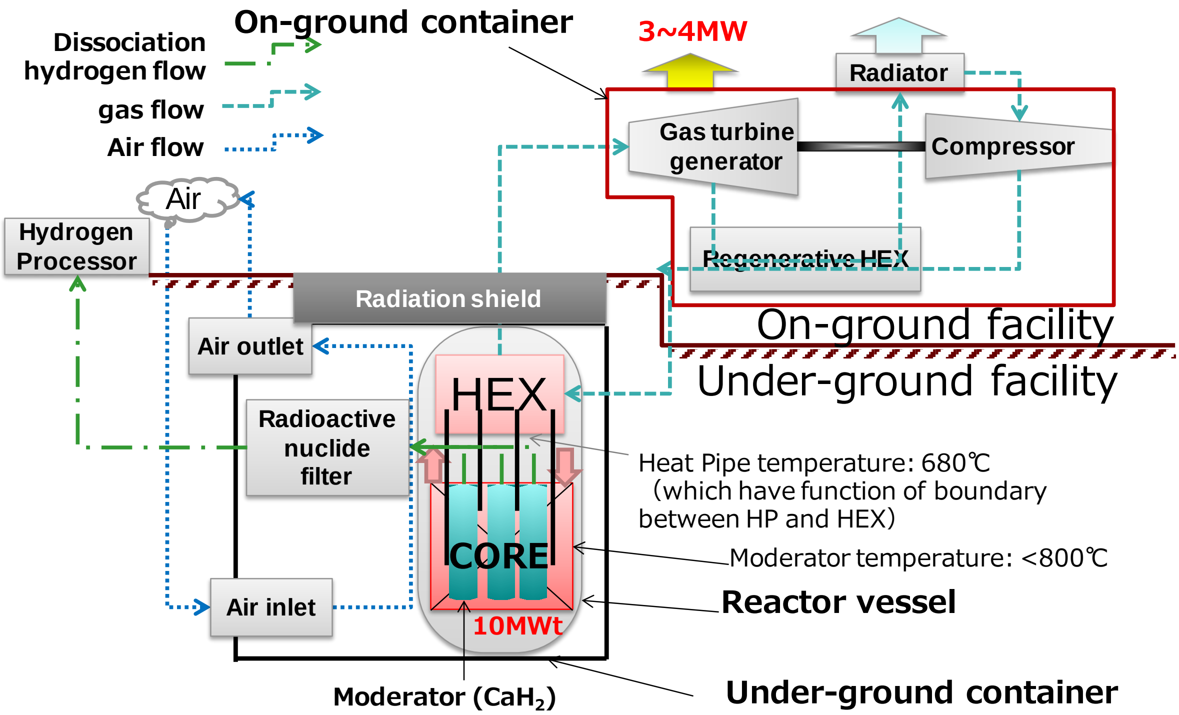
The primary circuit of the MoveluX reactor system makes use of heat-pipes which is one of the passive cooling devices, therefore, the primary circuit does not have pumps or other forced circulation devices. In the current design, sodium is selected as a working fluid of the heat-pipe from the viewpoint of usable temperature and heat transportability. The pressure in the primary system can be set close to atmospheric pressure since the proposed system does not utilize a pump for primary fluid circulation. Therefore, the risk of large-scale radioisotopes release can be reduced.

(d) Secondary Circuit

The secondary side of the MoveluX reactor system is currently a helium gas system. This gas system can provide high temperature around 700ºC and therefore usable not only for electric power generation, but also for heat supply, hydrogen production and so on. For electricity generation a Brayton cycle can be used as the power generation system.

(e) Heat-exchanger System

The heat-exchanger is very important component of the MoveluX reactor system, because a solution should be found that significantly improve the heat exchange density. To realize this goal, 2-phase micro channel heat-exchangers will be developed. The liquid/vapour sodium flow exists in the primary side while helium gas is flowing in the secondary side. In the primary side can use the condensation of sodium to improve the heat exchanger efficiency and heat transfer density. The complete system is illustrated below.



(f) Reactivity Control

A safety-rod is placed at the centre of the core for criticality safety assurance and reactor start-up. The core reactivity is controlled by the passive reactivity control device, such as Lithium Expansion Module (LEM).

(g) Reactor Coolant System

For the MoveluX reactor system operation, water as a coolant is not required in current design, because, final heat sink is assumed to be atmosphere. The reactor operation will be automated as possible by passive control devices based on the natural principle. Thus, manpower for the reactor operation is minimized.

4. Safety Features

The MoveluX reactor system is designed with inherent passive safety based on the natural principle, i.e. reactor shut down by moderator material property, decay heat removal from the surface of the reactor vessel by natural circulation of air.

(a) Engineered Safety System Approach and Configuration

The reactor vessel is place below ground to enhance reactor protection and radiation shielding. The radiation shield on top of the reactor is protect the reactor from physical attacks. For the natural hazards, the MoveluX reactor protection is by the core inherent safety characteristics based on the natural principles.

(b) Reactivity Control

In the case of emergency such as loss of cooling ability and/or RIA (with a postulated failure of the safety rod insertion) the core will be shut down autonomously by the material property of calcium-hydride. Specifically, calcium-hydride ability to moderate neutrons (neutron moderation power) decreases with increasing temperature since hydrogen dissociates in high temperature environments above 800°C. The proposed core design therefore has inherent safety for criticality in emergencies. If the hydrogen pressure is increased by its dissociation, hydrogen will be processed at a hydrogen processor (details still to be examined).

(c) Decay Heat Removal System

The decay heat after reactor shut down is removed passively. If heat-pipe and secondary circuit are keeping their function, decay heat removal are the same as for heat removal during nominal operation. In the case of loss of cooling ability of the heat-pipe, the decay heat is removed from the surface of reactor vessel by natural circulation of air. Furthermore, in the reactor vessel, the heat is transported from the centre to the periphery of the core by natural circulation and thermal conduction of liquid Pb-Sn, present in the gap between the fuel, heat-pipe and moderator. This decay heat removal system does not require a power source and can therefore realize a long (infinite) grace period. Consequently, the risk of core meltdown is expected to be very small.

(d) Containment System

The fuel material is contained in the reactor vessel. The fuel material is separated from primary cooling system, because, heat-pipe is closed heat transportation device. Additionally, the heat-exchanger between heat-pipe and secondary circuit is functioning as one of the boundaries. Therefore, radionuclide will be confined in the reactor vessel unless reactor vessel breaking.

5. Plant Safety and Operational Performances

For the MoveluX reactor system operation, water as a coolant is not required in current design, because the final heat sink is assumed to be the atmosphere. The reactor operation will be automated as possible by passive control devices based on natural principles. Thus, manpower for the reactor operation is minimized. Because remote monitoring and operation would become an option for operation cost reducing.

6. Instrumentation and Control Systems

Few I&C devices are installed to the MoveluX reactor system for the reactor start-up, monitoring and active control. Technically, the manned operation is not required during nominal operation in current design concept.

7. Plant Layout Arrangement

A footprint of MoveluX reactor system is designed to be two-container size. Additionally, atmosphere is used as a final heat-sink, therefore, restrictions of site locations are relaxed from conventional reactors. The requirement for the site during construction is not so much, because, the MoveluX reactor system assumed to be installed at remote places. Therefore, accessibility by trailer is one of the few requirements for the construction in the current design.

The MoveluX reactor system consist of the on-ground and the under-ground facility. The on-ground facility includes the power generator and/or other applications (heat utilization, hydrogen production and so on). The under-ground facility contains the reactor primary system (the core, heat-pipe and heat exchanger).

8. Design and Licensing Status

MoveluX is at the conceptual design stage.

9. Fuel Cycle Approach

The MoveluX reactor can be applied to either once-through fuel cycle scheme or closed fuel cycle scheme. It mainly depends on user country's fuel cycle policy.

10. Waste Management and Disposal Plan

The reactor vessel contains the spent fuel as it is, additionally, this spent fuel is carried out to temporary storage site with reactor vessel. After that, treatment of the spent fuel depends on the country's fuel cycle policy. One the one hand, in the once-through scheme, spent fuel is extracted from the core at the facility, then, spent fuel is stored to the cask for disposal. On the other hand, in the closed fuel cycle scheme, spent fuel is re-processed and re-fabricated as fresh MOX fuel for recycling use in LWR or FR

11. Development Milestones

2015	Start fundamental study based on the space reactor design
2017	Complete reactor type decision
2019	Start concept design
2025	Complete concept design and component demonstration
2028	Complete system demonstration without nuclear fuel
2030s	FOAK demonstration

U-Battery (Urenco, United Kingdom)

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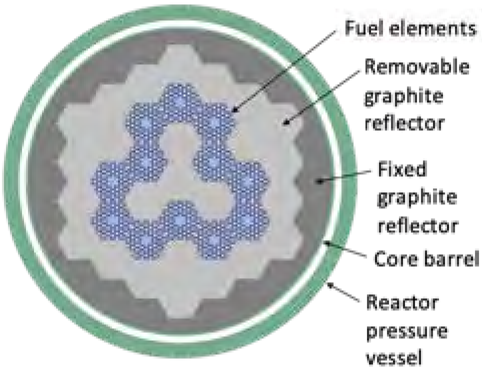


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Urenco, United Kingdom
Reactor type	High-temperature gas-cooled micro nuclear reactor
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 / 4
Primary circulation	Forced circulation helium
Secondary cycle	Nitrogen secondary circuit (no water)
Fuel type/assembly array	TRISO / Hexagonal
Number of fuel assemblies in the core	12 x 5
Fuel enrichment (%)	< 20%
Core Discharge Burnup (GWd/tonne)	~80 (average)
Reactivity control mechanism	Control rods, fixed burnable poisons, secondary shut-down absorber spheres
Design life (years)	5 EFPY core life, 30 year design life
RPV height/diameter (m)	Approx. 5.5 / 2.2
Distinguishing features	Simplicity, established technology basis, demonstrated fuel. Multiple modules can be installed at a single site.
Design status	Conceptual design

1. Introduction

U-Battery is an advanced/small modular High Temperature Reactor (HTR), capable of providing a low-carbon, cost-effective, locally embedded and reliable source of power and heat for energy-intensive industries and remote locations. The U-Battery concept is the result of a challenge initially set by Urenco to the University of Manchester and the Technical University of Delft to use existing, developed technology to bring nuclear to a market that, to date, has been serviced by diesel or other smaller sized fossil fuel or renewable technologies. The universities were asked to consider all nuclear technologies that were available but to then discount those that would require long research programmes. The study confirmed that there were opportunities to design a reactor that would be competitive when deployed at industrial sites and remote locations, and later developed for further applications.

U-Battery is being developed by Urenco in collaboration with a range of delivery partners, including: Jacobs, Cavendish Nuclear, and Kinectrics on reactor systems and safety, BWXT and NNL on fuel and fuel cycle, Rolls-Royce and Howden on key components, Costain on construction and civil works, and Mammoet and Daher on transport. The project has achieved some significant milestones, notably winning a funding under the UK Government's Advanced Modular Reactor competition, announced in June 2020, and engagement with the Canadian Regulator (CNSC) as part of their Vendor Design Review process. Development work to be undertaken in the frame of the AMR and VDR activities include design and development of the main nuclear island components (reactor pressure vessel and cross-duct, intermediate heat exchanger, helium circulator, control and instrumentation, and auxiliary systems for reactor cavity cooling, secondary shut-down, helium purification) and conventional equipment including an aero-derivative turbine and generator set, process heat exchanger, and building/civil works.

2. Target Application

U-Battery is a multipurpose reactor consisting of a standardised reactor block (comprising the primary helium and secondary nitrogen circuits, as well as a spent fuel store, and representing all of the nuclear-specific systems and components), coupled to a user-specific interface that utilises the energy contained within the nitrogen coolant. Three broad user-specific applications are envisaged: a unit focussed on providing process heat to a range of industrial applications; a unit focussed on electricity production via a gas-turbine; and a co-generation unit capable of providing a variable mix of heat and power. Potential applications include: heat and power supply to remote regions, localised heat and/or power supply for a range of existing industrial applications, and potential new applications such as hydrogen production. In common with many nuclear systems, rejected heat from the U-Battery primary applications could also be utilised for applications such as district heating and desalination.

3. Main Design Features

(a) Design Philosophy

The U-Battery design aims to (i) employ only established and nuclear-validated technology, (ii) maximise the use of modular manufacturing and off-site factory fabrication, (iii) be a cost-competitive and flexible energy supplier, (iv) serve different regions and diverse markets utilising a common reactor island, and (v) support alternative applications that may generate additional revenue or value. U-Battery exchanges the economies of large-scale for economies associated with localised power delivery (no transmission costs), simple safety systems (made possible by the highly robust fuel, very small thermal output, high thermal capacity of the moderator, and efficient natural heat transport), and factory-based manufacture and modular construction (again made possible by its simplicity and small size).

(b) Reactor Core

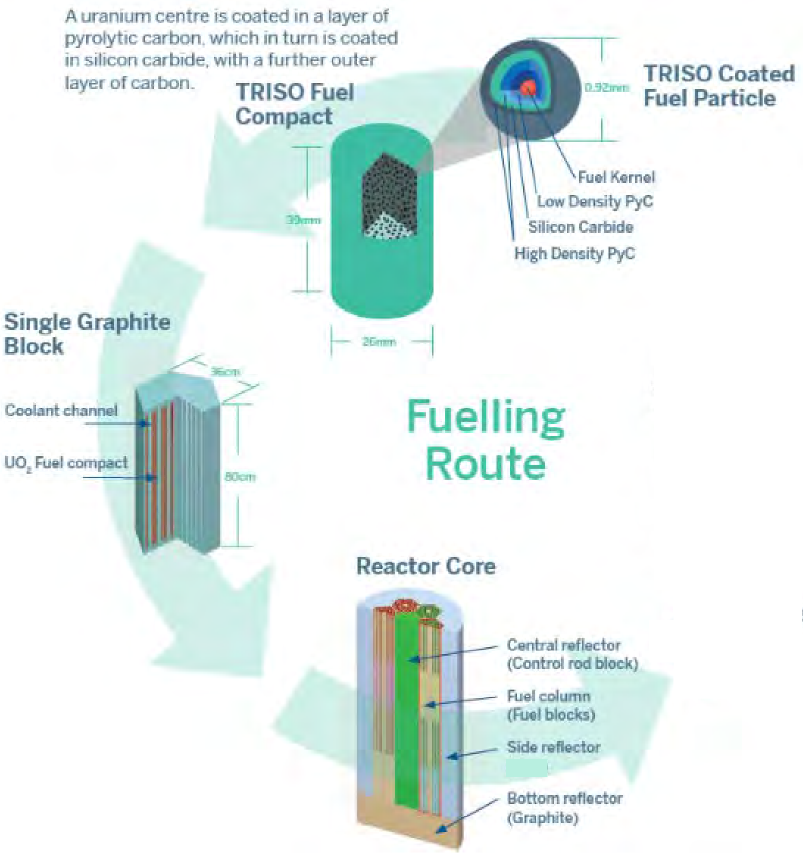
The annular prismatic core is composed of 12 fuel columns, each composed of a stack of 5 hexagonal graphite fuel elements, and arranged in a 3-lobed pattern. The fuel elements contain cooling channels for the helium primary coolant, and closed channels containing cylindrical fuel compacts, each approximately 25 mm long and 12.5 mm in diameter. The central and radial moderating reflectors are formed from graphite blocks, identical in geometry to the fuel blocks such that, if necessary, they can be replaced by the fuel handling machine during refuelling. Fixed graphite blocks provide a cylindrical core-former, and location is provided by a core barrel and a restraint system similar to that used in UK gas-cooled reactors.

(c) Reactivity Control

Fixed burnable absorbers made from B_4C are used for through-cycle reactivity control and power shaping. Control rods are provided to achieve shut-down under all plant states and to allow part-power operation. Although not strictly necessary for nuclear safety, an independent shut-down system is also provided, based on small absorber spheres that can be gravity-fed into secondary shut-down channels. These provide additional regulatory confidence, and offer flexibility during shut-down (for example, if needed, control rods could be replaced early in the cycle without unloading the core).

(d) Fuel Characteristics

The U-Battery fuel compacts contain TRISO coated fuel particles, composed of a uranium kernel coated in successive layers of pyrolytic carbon, silicon carbide, and an outer layer of pyrolytic carbon. The TRISO particles are of a design recently validated as part of the US-DOE's AGR fuel demonstration program in the Advanced Test Reactor at Idaho National Laboratory, with uranium enriched to just under 20%. Although the maximum fuel temperatures in the U-Battery core never approach the well-established safety limit for TRISO fuel with UO_2 kernels (1600ºC), UCO kernels validated in the Advanced Gas-cooled Reactor (AGR) program will be employed because they offer even higher temperature capability, and were validated in greater numbers.



(e) Fuel Handling System

The fuel handling equipment for U-Battery is based on a well-proven pantograph system used in previous HTRs (and fast reactors), together with a fuel carrousel arrangement based on experience with the UK's gas-cooled reactors and on spent fuel handling experience at the UK's Sellafield site. An on-site spent fuel store is provided adjacent to the reactor core to facilitate core unloading prior to refuelling. The natural circulation cooled spent fuel store is also based on experience with similar facilities at Sellafield. Much of the fuel handling equipment is designed to be portable, and will not be left on site during periods of operation. This significantly enhances the resilience of the facility against unauthorised access to fissile material, as well as allowing optimal use of the refuelling equipment to service multiple U-Battery units.

(f) Reactor Coolant System

In common with all previous HTRs, the U-Battery employs a helium primary coolant. The possibility of employing CO_2 was briefly considered early in the project, but irradiation experiments conducted as part of the Manchester-Delft collaboration demonstrated that significant coolant chemistry intervention would be necessary to supress graphite oxidation, and this was not considered to be compatible with minimal (or even no) operating personnel. The use of a secondary steam (Rankine) power cycle was rejected for similar reasons, and also because of the positive reactivity injection that would result from a heat exchanger failure and steam ingress into the primary circuit. A direct-cycle configuration was rejected because of the significant development requirements associated with a helium turbine. The preferred solution has been to adopt an indirect cycle configuration with a nitrogen secondary coolant, which can either pass heat to a tertiary fluid for heat applications, or drive a gas-turbine based on an aero-engine but without the combustion stage and employing a closed-cycle configuration.

(g) Reactor Pressure Vessel

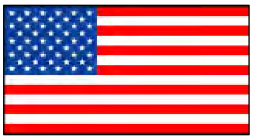
The U-Battery core is housed in a Reactor Pressure Vessel (RPV) manufactured from SA-508/533 steel in order to take advantage of the extensive experience from LWR RPVs. The RPV design is based on the ASME-III design code for a working pressure of 40 bar at 300°C, together with limited transient temperature excursions as permitted by the code. The helium coolant is passed to and returns from the primary heat exchanger via a coaxial duct housed within a cross-vessel, situated towards the bottom of the RPV reactor. Limiting the design to a single major RPV penetration reduces the impact of a major failure of the cross-duct by eliminating the "chimney effect" that could result from multiple penetrations. The RPV wall is maintained within acceptable temperature limits by the cool helium returned from the primary heat exchanger, which passes up the annulus between the RPV and the core barrel before passing into an upper plenum and being directed down through the core, into a lower plenum, and out to the heat exchanger through the inner part of the coaxial duct. The RPV is provided with a bolted head that can be removed for refuelling and inspection.

4. Plant Layout Arrangement

A plant layout has been developed based on a belowground reactor cavity and adjacent spent core storage facility, both served by an overhead crane and collocated with the power conversion module in a vented confinement building. The reactor and heat exchanger are positioned side-by-side, but each in its own compartment. Both these are served by an above maintenance floor, that also allow refuelling and has access to the used fuel cartridge store located next to the reactor cavity. Natural convection is used to cool the spent fuel assisted with the special fuel store ventilation. The turbine generator is located in a separate building (only above ground). The layout make provision for fuel to be loaded / unloaded into fuel casks with access via he fuel handling facility.

5. Development Milestones

2008	The project was initiated by Urenco and the concept design was developed by the Universities of Manchester and Dalton Institute in the UK and Technology University of Delft in the Netherlands.
2011	Feasibility study completed
2017	Memorandum of Understanding signed with Bruce Powers in Canada
2018	Green light to progress to Phase 1 of UK Government's Advanced Modular Reactor Programme
2019	Completed the first stage of the evaluation process in Canadian Nuclear Laboratories' (CNL) invitation to site a first-of-a-kind small SMR in Chalk River, Ontario. Established a service agreement with the Canadian Nuclear Safety Commission for pre-licensing Phase 1 vendor design review.
2020	U-Battery awarded funding by UK Government to conduct design and development to bring to market a new, innovative nuclear technology
2023	Development of detailed design
2025	Construction first-of-a-kind (FOAK) plant
2028	First-of-a-kind U-Battery operating



AURORA (OKLO Inc., United States of America)

MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	OKLO Inc., United States of America
Reactor type	Fast Reactor
Coolant/moderator	Liquid metal/no moderator
Thermal/electrical capacity, MW(t)/MW(e)	4 / 1.5
NSSS Operating Pressure (primary/secondary), MPa	Not pressurised
Fuel type/assembly array	Metal fuel
Refuelling Cycle (months)	Up to 20 years
Design life (years)	20 years per deployment
Plant footprint (m^2)	4180
Design status	Accepted combined license application by the US NRC

1. Introduction

Oklo's first product is the Aurora powerhouse. The Aurora is orders of magnitude smaller than any commercial reactor in the U.S., on the order of a research reactor. The lower power of the Aurora also leads to low decay heat production. For reference, one day after shutdown the reactor is producing 21 kW of decay heat, and 1 month after shutdown, the reactor is producing 7 kW of decay heat. This small amount of heat is generated and dissipated in a module that contains tens of tons of metal, and tens more tons of shielding and other structural material. For comparison, one fuel assembly at Diablo Canyon produces approximately 4 times as much power as the entire Aurora core, and contains approximately 0.5 tons of fuel, cladding, and structural material (or about 0.6 tons when the assembly is immersed in water and the water mass is included).

2. Target Applications

The Aurora is designed to provide affordable, reliable, carbon-free electricity on a microgrid scale. The Oklo product is to provide power on a power purchase agreement basis.

3. Main Design Features

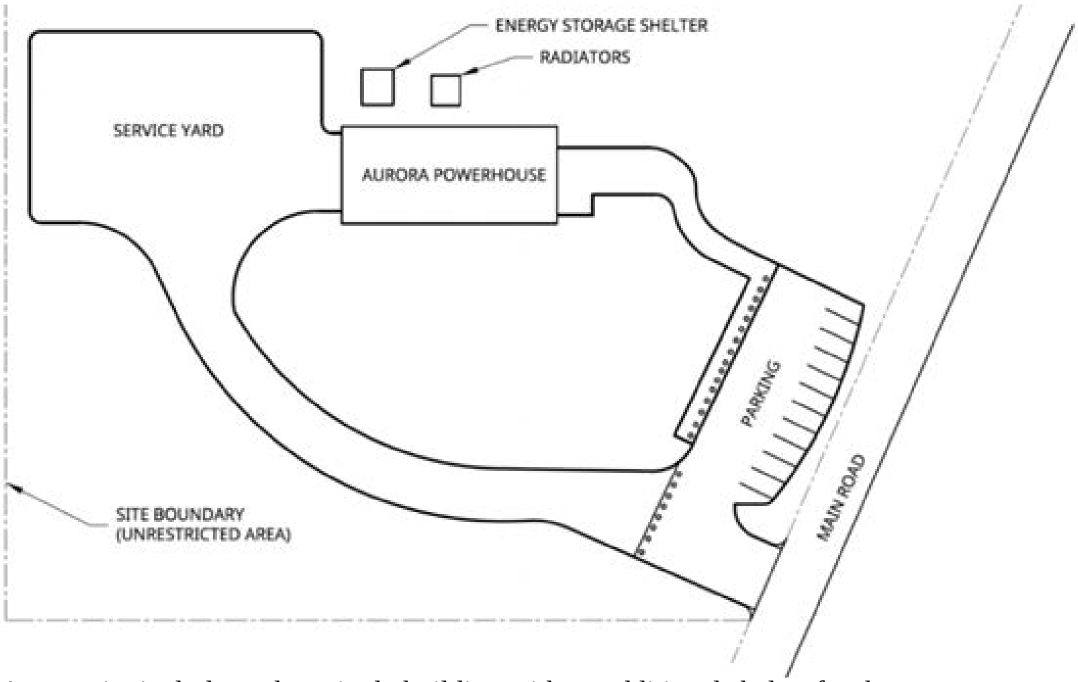
Safety and defense-in-depth are fundamentally accomplished in the Aurora design by its inherent characteristics, including:

- Small size, low power output, low power density, and low decay heat output
- Low fuel burnup, small inventory of fuel, and limited available source term
- Low decay heat term, removed by inherent and passive means
- High thermal conductivity materials reduce temperature hot spots, and large thermal mass provides capacity for heat dissipation
- Inherent reactivity feedbacks ensure reactor power is controlled during overpower or overtemperature events
- Multiple barriers to fission product release
- High thermal conductivity materials reduce temperature hot spots, and large thermal mass provides capacity for heat dissipation
- Ambient pressure system removes sources of pressure and limits driving forces for release
- No use of water cooling, uses dry heat rejection instead.

4. Initial Proposed Site

An Aurora will be sited at Idaho National Laboratory (INL) Site in southeast Idaho, which is referred to as the "Aurora INL site." The Oklo Inc. site use permit request was evaluated by the Department of Energy Office of Nuclear Energy (DOE-NE), a field office of the DOE, through the site use permit process and received a permit on September 26, 2019. This Site Use Permit grants Oklo personnel access to the land on the INL Site leased to Oklo Inc. Oklo Inc. subsidiary Oklo Power LLC (Oklo Power) will own and operate the Aurora at the INL Site.

5. Plant Layout Arrangement



The Aurora site includes only a single building with an additional shelter for the energy storage inverter. The Aurora powerhouse has two floors and has a footprint of less than 5000 ft^2. The first floor of the Aurora powerhouse provides for monitoring of the plant, access control, and maintenance. At the front of the first floor is a visitor area. The power conversion system is located on the first floor and is connected to dry air coolers (i.e. radiators). The radiators are located external to the Aurora powerhouse. The radiators exchange heat with ambient air as the ultimate heat sink for the power cycle. The reactor module is emplaced below the floor of the basement. There is a parking lot for the site as well as landscaping surrounding the site. An energy storage inverter is located in a shelter approximately 50 feet away from the building. There are no structures other than the Aurora powerhouse and the energy storage inverter shelter on the site. The boundary is the site boundary and unrestricted area. The site boundary is the line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee. Unrestricted area, the area outside the site boundary, means an area to which access is neither limited nor controlled by Oklo Power.

6. Design and Licensing Status

Oklo's combined license application was accepted by the U.S. NRC in June 2020.

7. Development Milestones

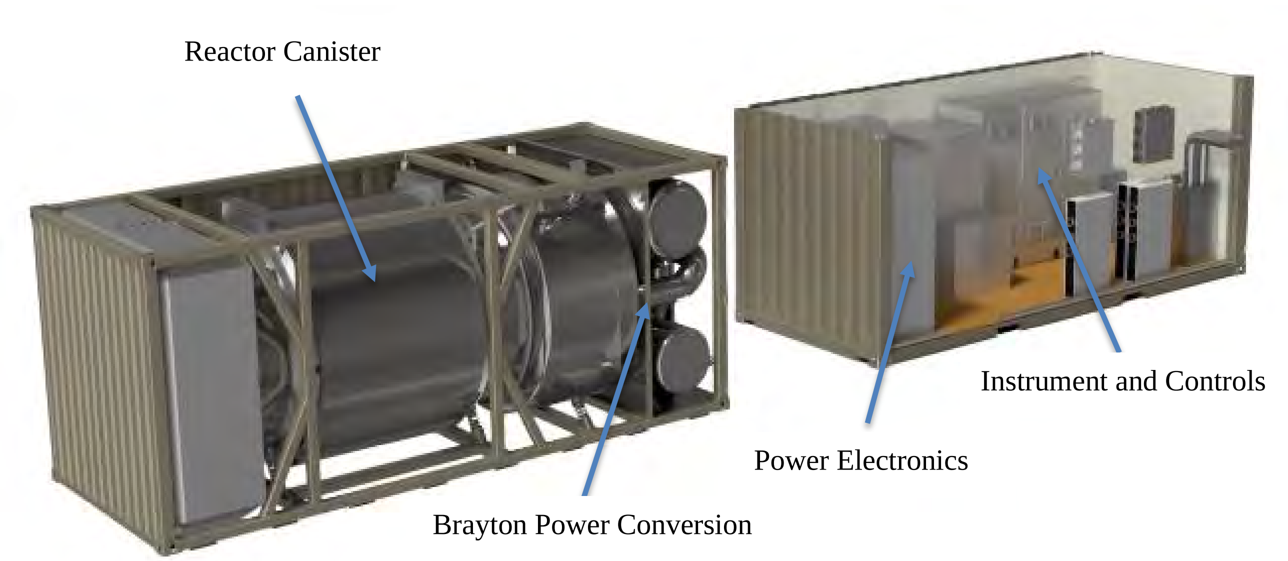
2016	Demonstrated prototype fuel fabrication
2019	Received a site use permit from the U.S. Department of Energy
2019	Awarded access to recovered used fuel from the Idaho National Laboratory
2020	Combined license application submittal to the U.S. Nuclear Regulatory Commission and acceptance for review

Westinghouse eVinci™ Micro Reactor (Westinghouse Electric Company LLC, USA)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse Electric Company LLC, USA
Reactor type	Heat pipe cooled
Coolant/moderator	Heat pipes/ metal hydride moderator
Thermal/electrical capacity, MW(t)/MW(e)	7-12 / 2-3.5
Primary circulation	Heat pipes
NSSS Operating Pressure (primary/secondary), MPa	N/A
Core Temperature (ºC)	NA800
Fuel type/assembly array	TRISO or another encapsulation
Number of fuel assemblies	Monolith core
Fuel enrichment (%)	5 - 19.75
Core Discharge Burnup (GWd/ton)	Not Disclosed
Fuel cycle (months)	>36
Reactivity control mechanism	Ex-core control drums
Approach to safety systems	Inherent and passive safety for shutdown and heat removal
Design Life (years)	40
Plant footprint (m^2)	< 4000
RPV height/diameter (m)	N/A
Seismic design (SSE)	IBC Zone 4 Category F
Fuel cycle requirements / Approach	No onsite refuelling Replace reactor approach
Distinguishing features	Transportable reactor that can operates autonomously
Design status	Conceptual design

1. Introduction

The Westinghouse eVinci Micro Reactor is designed for energy generation in remote or isolated locations. The design can produce both process heat and electricity for remote communities, mining operations, or critical infrastructure installations. The key attribute of the design is its transportability within standard shipping containers. The design is based on heat-pipe reactor technology that has been developed and tested by Los Alamos National Laboratory for space applications. Because of its compact and simplified design, the eVinci Micro Reactor will be manufactured and fueled in a factory, and then transported to an end user site. The figure below shows how the eVinci reactor and power conversion system can be packaged into two standard transport containers. One of the containers houses the reactor and the power conversion system. The other container includes power electronics and the instrumentation & controls (I&C) system.



Westinghouse's eVinci Micro Reactor schematic.

2. Target Application

The eVinci Micro Reactor is designed specifically to serve remote communities, mining operations, or military installations. It combines both heat and power generation capabilities, addressing diverse energy needs of these decentralized and off-grid markets.

3. Main Design Features

(a) Design Philosophy

The design of the eVinci reactor leverages proven heat pipe technology developed by the Los Alamos National Laboratory (LANL) for space application. This uranium-fueled reactor does not use a bulk primary coolant. Instead, heat is removed from its core using passive heat pipes, limiting the number of its moving parts and providing overall plant simplicity. The design utilizes the inherent safety features in the fuel, moderator and heat pipes to enhance safety and self-regulation capability.

(b) Design Overview

The fundamental reactor design is based on a solid core block that includes a matrix of nuclear fuel, moderator and heat pipes that extract the heat from the core region. The reactor core by itself is subcritical. It requires radial and axial reflectors to improve neutron utilization. Reactivity control and shutdown are performed with radial control drums placed around the core. Shutdown can also be performed with shutdown rods that can be inserted into the core block. The control drums are the only moving parts within the reactor canister leveraging the passive heat pipe thermal exchange. The reactor heat transport design utilizes the unique properties of sodium heat pipes. Sodium heat pipes are embedded in the solid core and under normal operation are used to move the core heat to the heat exchanger and associated power conversion system. Heat pipe operation removes the need for coolant pumps in the primary system. The complete reactor and power conversion systems are controlled with an autonomous control system that is based on proven nuclear I&C systems.

The eVinci Micro Reactor has inherent safety features. For example, in case of decay heat removal, the core design within the reactor canister enables heat removal by conduction to the outer sections of the reactor canister that passively allow heat transfer to the surrounding atmosphere (air). The core is designed with negative reactivity feedback, allowing for increased safety in the event of an accident scenario. If external power is lost or the reactor is tripped through its autonomous control system, the control drums automatically rotate to a high neutron absorption position around the core and subsequently shutdown the reactor. This solid core design eliminates many traditional accident scenarios such as high-pressure pipe ruptures or loss of coolant accidents.

An open-air Brayton system was incorporated due to its compactness, technical maturity and reliability. A single shaft gas turbomachinery composed of a compressor, turbine, and alternator operating as a single rotary spool. The high-speed single-shaft architecture results in a very compact power module. An annular recuperate and simplified, low pressure ducting will enable close integration of the engine with the primary heat exchanger. To avoid the use of lubricants, magnetic bearings will allow the engine to operate in any orientation, facilitating optimal integration and packaging. The high-speed alternator acts as a motor to start the engine during regular and black start up. The mechanical configuration is critical to starting the engine with the relatively slow reactor ramp-up and avoidance of thermal stress. Power electronics are combined with the variable speed engine to enable efficient part-load and load following power management.

4. Resilient Features

Resiliency of eVinci Micro Reactor is enabled through packaging within a secure canister, installed in a strong secure vault on site and connected to a micro-grid coordinating both heat and power. A dedicated device called SMART is 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 with other power generation resources.

5. Operational Performances

The eVinci Micro Reactor leverages its autonomous design to reduce the personnel required on site. It is planned that a few personnel would be required per shift for monitoring and security as the autonomous control system will limit the need for operators.

The eVinci design has the capability for load following and grid frequency control. The eVinci autonomous control system in combination with a micro grid, can perform load following for the most demanding situations in remote applications. Most of the load following demands can be executed utilizing the self-control capabilities, reactor control and power conversion control. Other high demand fluctuation will be supported by power resources coupled to the micro grid. The eVinci system has black start-up capability supported by batteries, for instances where the local micro grid cannot support the start-up process.

6. Instrumentation and Control Systems

The Autonomous Control System (ACS) is the primary I&C system of the eVinci Micro Reactor. The ACS facilitates autonomous control by utilizing fiber optic sensors for high fidelity temperature measurement of the reactor, neutron flux sensors for measurement of the reactor core operation, and load following logic programmed into the ACS. The ACS is programmed with functional logic for autonomous operation of the eVinci reactor. Temperature measurements from fiber optic sensors and neutron flux measurements from self-powered neutron flux detectors (SPNDs) determine trip decision and load following logic. The trip decision logic ensures the reactor does not transverse predefined operational limits. Load following logic autonomously adjusts control drum position based on load variances, reactor temperature and neutron flux measurements. The control drum interface logic provides a logical priority between the trip, load following logic and manual system controls. The ACS also collects environmental and operating data during normal operation, the autonomous logic functions control the reactor. The autonomous logic functions relinquish control of the reactor to a manual operator if required. Due to the autonomous control, no specific control room is required.

7. Plant Layout Arrangement

The eVinci Micro Reactor is designed for ease of logistics, with nearly no on-site construction and minimum on-site installation. The eVinci system is nearly 100% fabricated in the factory and packaged in shipping containers to be transported to site via air, water or land. After installation of the container at site, the only connection needed between the boxes are the electrical power conduits or pipe connections if combine heat and power is required. To ensure allowable dose levels are met, integrated shielding is designed for transportation. However, a shielding structure is planned for the reactor container to limit dose rate to operators and the public.

Designed for remote applications, the eVinci Micro Reactor can provide heat and power via a micro grid system. The system can be configured for all the voltage and frequencies that might be required at remote applications. If power demands require several eVinci systems, they can be staged together at a site; with each unit operating independently. Because the eVinci Micro Reactor is similar to a battery design power supply, it does not require on-site refuelling. This eliminates additional onsite facilities and structures that would typically be required for refuelling. After the end of its fuel cycle, the eVinci Micro Reactor will be transported back to the factory for refuelling or for long-term storage.

8. Design and Licensing Status

The eVinci development program has completed the conceptual design phase. Westinghouse drives the maturity of the eVinci Micro Reactor product through evaluating both Technology Readiness Level (TRL) and Manufacturing Readiness Level (MRL). The current TRL and MRL of the eVinci Micro Reactor technical areas are estimated to be at a level 5.

Westinghouse is currently working with both the U.S. Nuclear Regulatory Commission (NRC) and the Canadian Nuclear Safety Commission (CNSC) to license the design and technology. At this point, pre-application discussions and activities have highlighted the licensing efforts to date:

- Request to submit application for Vendor Design Review to Canadian Nuclear Safety Commission (CNSC)
- Submitted eVinci Micro Reactor Licensing Modernization Project (LMP) Demonstration “table top” report as part of a project to evaluate the LMP process. This report was provided to NRC for information.
- Provided the U.S. NRC a Regulatory Issue summary on process for scheduling and allocating resources for

review of a new licensing applications.

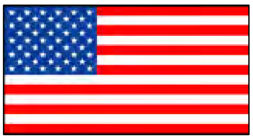
9. Fuel Cycle Approach

The eVinci Micro Reactor core will use either the High-Assay Low Enriched Uranium (HALEU) - Uranium Oxycarbide (UCO) in a tristructural isotropic (TRISO) fuel or other fuel that is encapsulated.

After 3 years of full power operation without refuelling, the eVinci Micro Reactor will be disconnected and transported back to the factory in its original canister for either refuelling and redeployment or for long-term storage, which can be accomplished in the eVinci Micro Reactor canister it itself.

10. Development Milestones

2021	Electric demonstration
2024	First nuclear demonstration



MMR™ (Ultra Safe Nuclear Corporation, United States of America)

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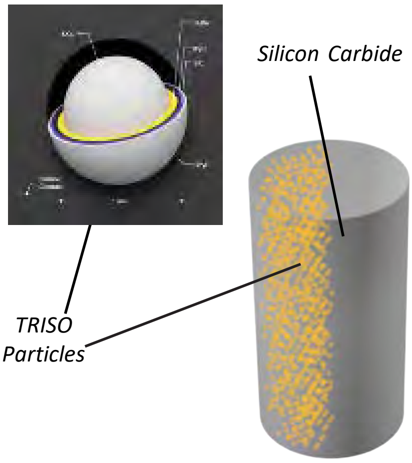
MMR Unit



Fully Ceramic Micro-encapsulated (FCM™) fuel

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Ultra Safe Nuclear Corporation, USA
Reactor type	High Temperature Gas-cooled Reactor / micro-reactor / nuclear battery
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	15 / >5
Primary circulation	Forced circulation
NHSS Operating Pressure (primary/secondary), MPa	3MPa primary, 100kPa secondary
Core Inlet/Outlet Coolant Temperature (°C)	Helium 300 / 630, solar salt stored at 560
Fuel type/assembly array	FCM™ or TRISO graphite /Hexagonal
Fuel enrichment (%)	HALEU 19.75%
Core Discharge Burnup (GWd/ton)	> 60
Refuelling Cycle (months)	Never; for the lifetime
Reactivity control mechanism	Control rod insertion, negative temperature coefficient.
Approach to safety systems	Passive (Category A), no moving parts, does not require fluid or natural convection
Design life (years)	20
Plant footprint (m^2)	130 x 96
RPV height/diameter (m)	8.1 / 3.1
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Fueled once during lifetime.
Distinguishing features	No core meltdown; modular reactor and modular powerplant, adjacent non-nuclear power conversion plant; no EPZ required; load following / fully dispatchable; nuclear reactor isolated from load via molten salt loop; < 6 months assembly at site
Design status	Basic / Preliminary Design

1. Introduction

The MMR™ system is a small modular nuclear energy system that delivers safe, clean and cost effective electricity and heat to remote mines, industry and communities. The energy system consists of two plants, the nuclear plant and the adjacent non-nuclear power plant. The Nuclear Plant is independent of the Adjacent Plant, requiring no supporting services for any event for its safe operation. The Adjacent Plant consists of the equipment and systems that convert the process heat to electrical power or other forms of energy as per client requirements. The Nuclear Plant would generate approximately 15 MW(t) of process heat that could supply electrical power and/or heat to a small community as the potential end user. The electrical power could also be supplied to the area grid, over an anticipated life span of 20 years.

2. Target Application

The MMR power plant is designed for remote communities as standalone micro grid or for heavy industry applications such as process heat and hydrogen production. MMR energy system can scale to match demand with multiple units at the same site.

3. Main Design Features

(a) Design Philosophy

- Safety first
- Low power to surface area ratio
- Low power density

(b) Reactor Core

The reactor core consists of hexagonal graphite blocks containing stacks of FCM™ fuel pellets. The MMR™ reactor core has a low power density (1.24 W/cm^3) and a high heat capacity resulting in very slow and predictable temperature changes. The MMR™ reactor is fueled once for its lifetime and sealed. The graphite core provides neutron moderation and reflection functions.

(c) Fuel Characteristics

The MMR reactor can use either Fully Ceramic Micro-encapsulated (FCM™) fuel or TRISO in graphite fuel. FCM fuel builds on over fifty years of TRISO development efforts in the United States, Germany, and the United Kingdom for gas cooled reactors. TRISO particles are usually compacted into a graphite matrix as it is a low cost and low cross section moderator often used in nuclear reactors. In contrast to traditional TRISO in graphite fuel, FCM fuel replaces the graphite matrix with a dense SiC matrix. This creates an extra barrier to fission product release and improves each TRISO particle’s structural and containment characteristics. SiC provides better fission product containment, irradiation stability, and thermal characteristics compared to graphite. Combining TRISO with a SiC matrix provides an extremely rugged and stable fuel with extraordinary high temperature and irradiation stability.

(d) Fuel Handling System

The MMR is fuelled once in its lifetime. There is no on-site fuel handling or storage.

(e) Reactivity Control

The core provides for areas for insertion of control rods. The MMR reactor core has a low power density and a high heat capacity resulting in very slow and predictable temperature transients. Requirements for the control system of the MMR are un-remarkable. The core has strong negative temperature feedback.

4. Safety Features

In the case of an accident, the MMR is unable to melt down because any excess heat passively dissipates into the environment without any moving parts, fluids, or natural circulation. No active cooling or natural convection is required to maintain safe temperatures.

5. Plant Safety and Operations

The MMR technology has been developed by USNC and is based largely on proven designs with inherent safety features, further augmented with specific novel safety features. The degree of such proven inherent safety design features confers confidence in the operability and safety of the facility, while the novel safety features further enhance the confidence in the technology. Operations are simple with minimal operations and maintenance requirements.

6. Plant Layout Arrangement

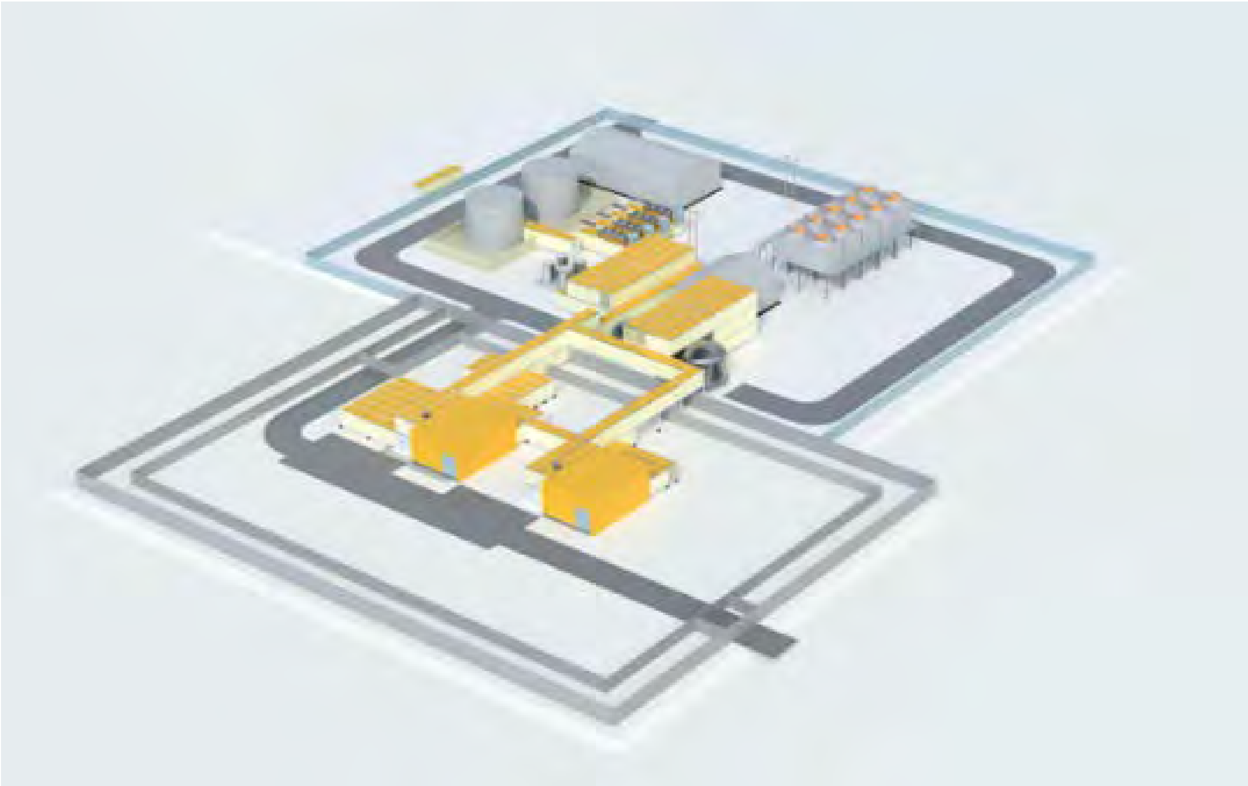
The MMR facility includes a Nuclear Plant containing an MMR reactor, and an Adjacent Plant, which are the main physical works related to the Project.

(a) Nuclear Plant

The Nuclear Plant provides process heat to the Adjacent Plant where it is converted to electrical power and/or heat as per client requirements. The Nuclear Plant contains the Nuclear Building and the Citadel Building, where the MMR reactor and its associated Nuclear Heat Supply System are housed.

i. Nuclear Building. The Nuclear Building is constructed on top of the Citadel Building. The Nuclear Building contains the equipment associated with the main control room and security room, including instrumentation and control equipment heating, ventilation and air conditioning and electrical equipment room, including instrumentation and control equipment radiation change-over area and change room waste storage and decontamination area

ii. Citadel Building. The Nuclear Heat Supply System (that includes the reactor core) is housed in a vertical cylindrical concrete structure, named the Citadel Building. The Citadel Building protects the reactor and the Intermediate Heat Exchanger from hazards (both external and internal to the Citadel Building), and the Citadel Building wall provides biological shielding that mitigates against possible radiation exposure from the reactor.



MMR two-unit Power Plant.

(b) Adjacent Plant

The Adjacent Plant buildings contain the equipment required for the generation of electricity from the heat supplied by the Nuclear Plant and to interface with any customer end-use facilities. Additionally, there are offices, a training and a visitor centre on the Project site.

i. Adjacent Plant Molten Salt System. The Adjacent Plant Molten Salt Heat Storage System acts as an intermediary to transport and store the heat generated in the Nuclear Plant and transfer it to a steam cycle for the purpose of generating power and heat for customer applications. The Adjacent Plant Molten Salt System consists of pumps and pipes containing molten salt as well as hot and cold storage tanks. These tanks serve as an energy storage system and help to regulate the flow of molten salt.

ii. Power Generation and Steam Turbine Generator. The function of this system is to generate electricity from the heat supplied from the Nuclear Plant via the molten salt. The Power Generation System consists of the turbine generator and supporting infrastructure. The Adjacent Plant will have a main electrical grid connection for supply of the electrical power generated via transmission infrastructure. Additionally, there will be an auxiliary grid connection to provide station power when the main connection is not available.

7. Design and Licensing Status

Global First Power (GFP) has submitted a 'License to prepare site initial application' for the MMR Demo power plant at the Chalk River site to the Canadian Nuclear Safety Commission (CNSC) which regulates all nuclear activities in Canada.

8. Fuel Cycle Approach

The MMR^{TM} will be fuelled only once in its lifetime. The Decommissioning phase activities are anticipated to take approximately 6 months. Spent fuel will be stored according to the national storage plan. No processing or conditioning of the spent fuel is required because FCM fuel is already in a geologically stable form.

9. Waste Management and Disposal Plan

All waste will be handled and processed in a responsible and safe manner that ensures minimum exposure to all personnel handling, transporting and processing the waste. Waste will be segregated at source as non-radioactive waste and radioactive or potentially radioactive waste. Waste will be temporarily stored on the Project site in defined areas and transported to authorized processing facilities at appropriate times, dependent on the category and type of waste.

10. Development Milestones

USNC MMR Demo unit at Chalk River (Canadian Nuclear Laboratories)

2011	Secured FCM™ fuel and MMR™ reactor patents
2016	Established R&D and fabrication laboratories
2017	Initiated FCM™ fuel development and qualification
June 2018	GFP submits proposal to CNL, supported by OPG & USNC
February2019	CNL announces GFP proposal enters Stage 3 proponent review process
March 2019	GFP submits license to prepare site initial application to CNSC
2020	License to Prepare Site
__ Future__	__ Future__
2021	Begin Site Preparation
2016-2021	Project Development
2021-2027	Site Preparation and Construction
2023-2052	Plant Operation
2044-2058	Decommissioning
2058-2060	Abandonment

ANNEX I
Summary of Global SMR Technology Development and Deployment

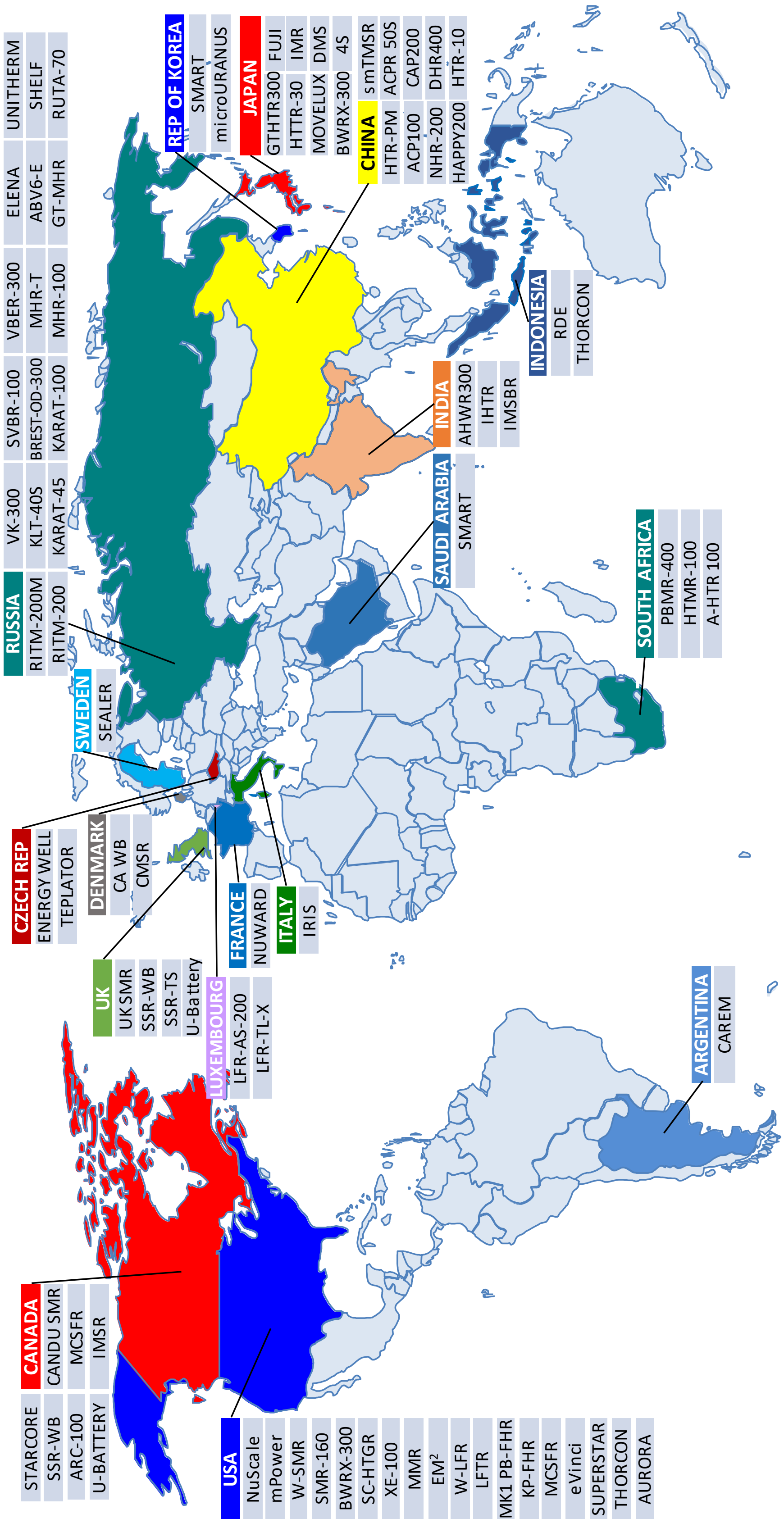


Figure I-1 Global Map of SMR Technology Development

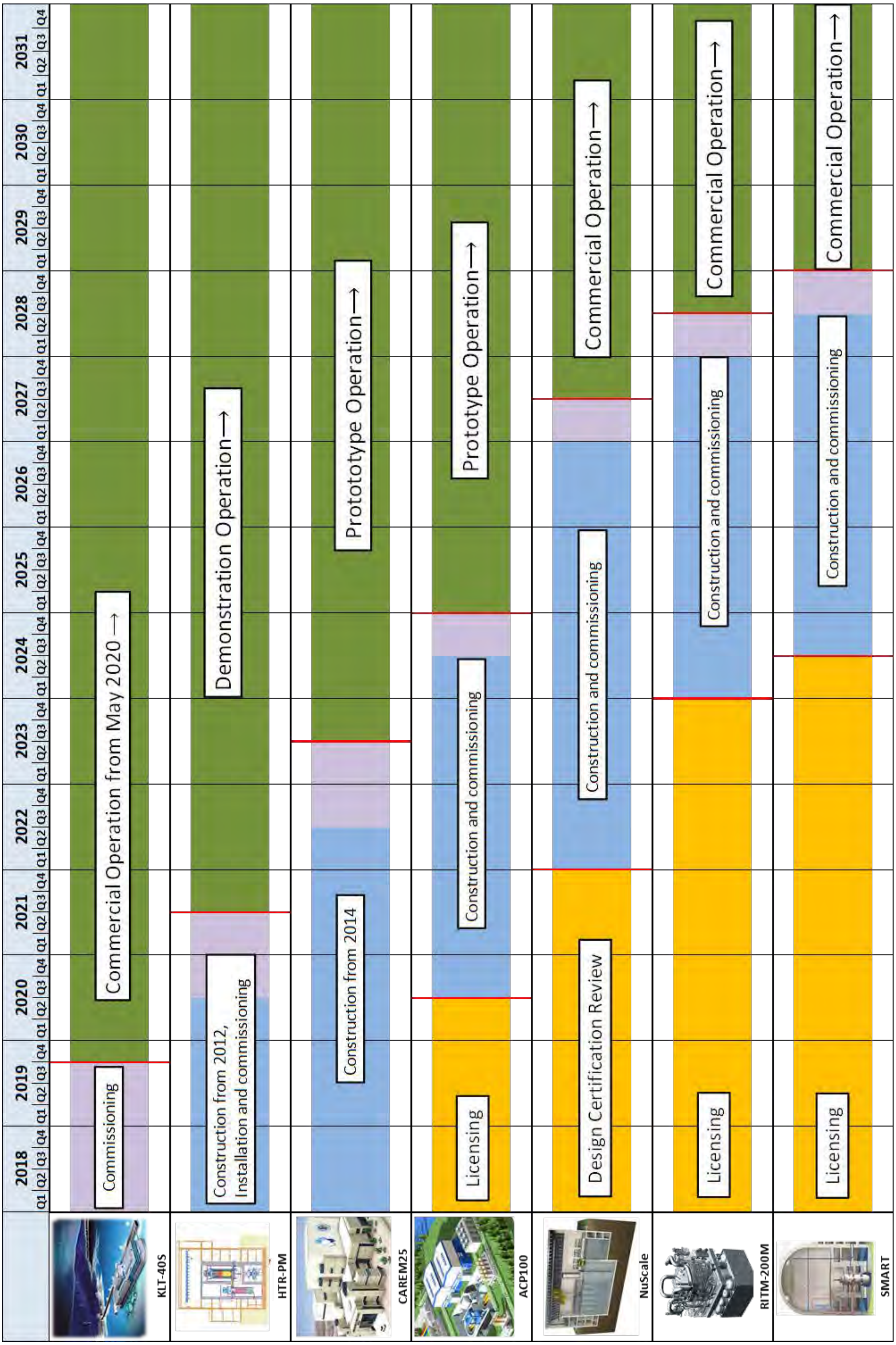


Figure I-2 General Timeline of Deployment as of 2020

Figure I-3 Government and Private Sectors on SMR Technology Development

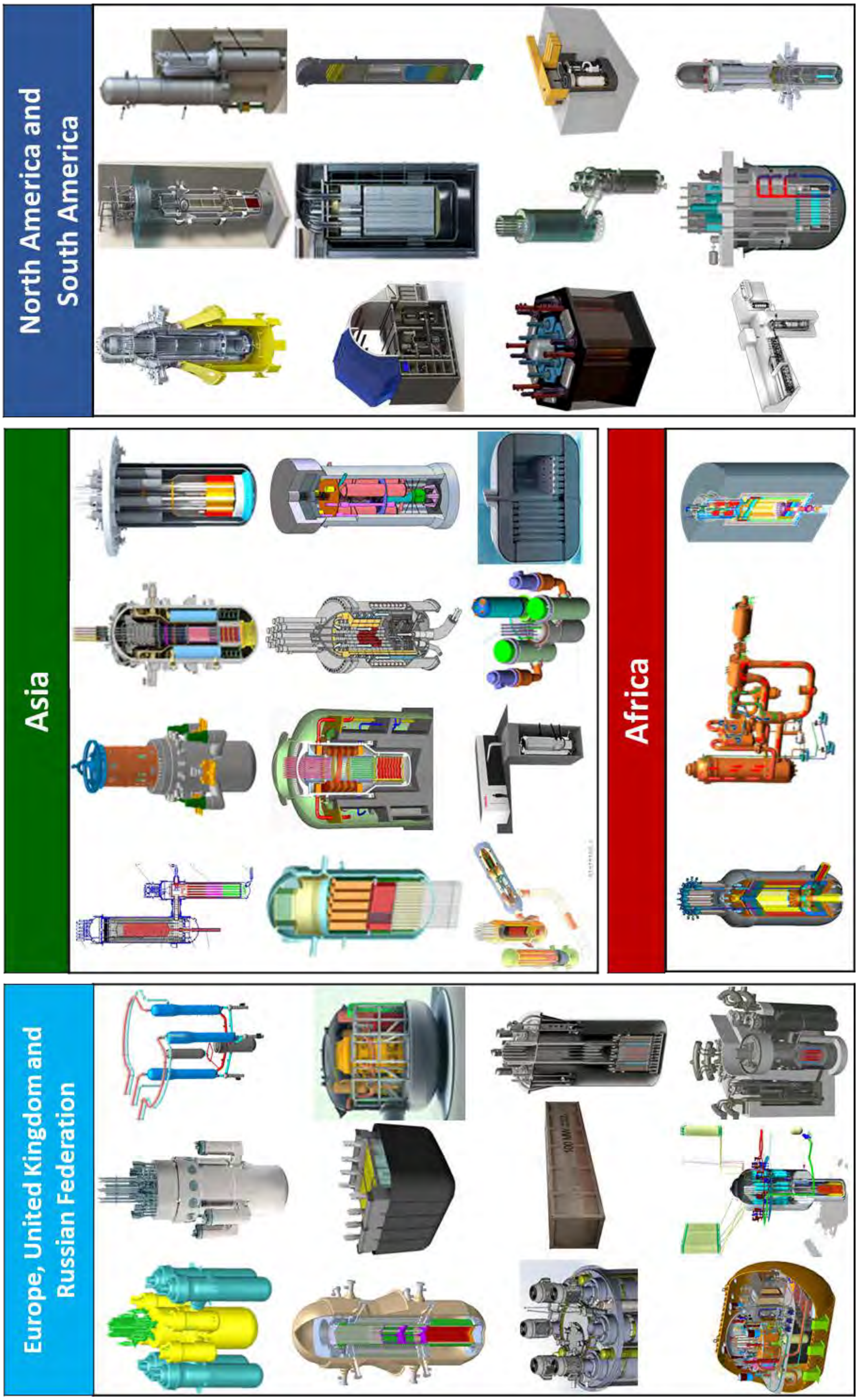


Figure I-4 SMR design and technology across the world's regions

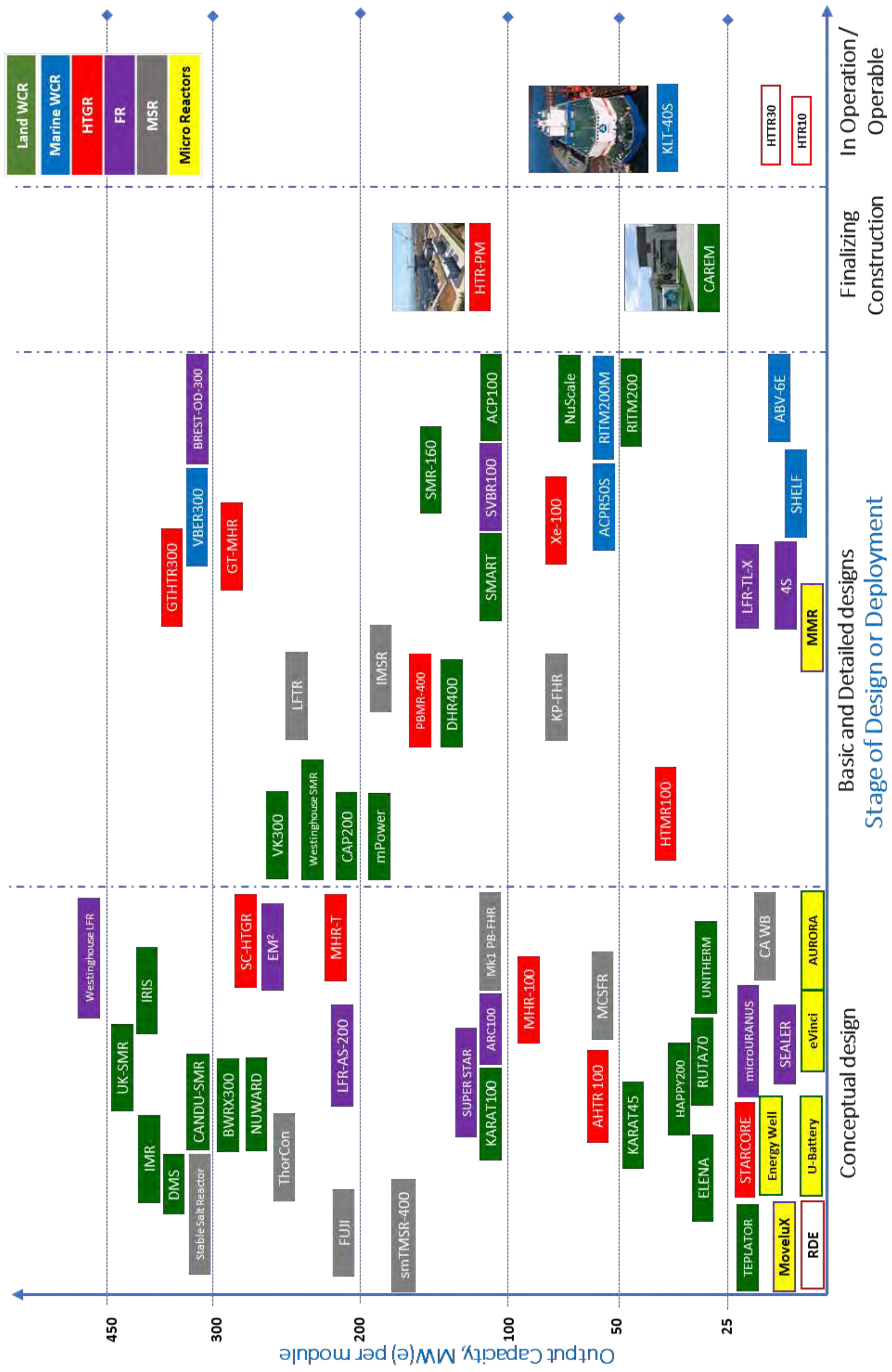


Figure I-5 Stage of Design or Deployment of SMRs in terms of their Output Capacity

ANNEX II
Power Range of SMR Designs

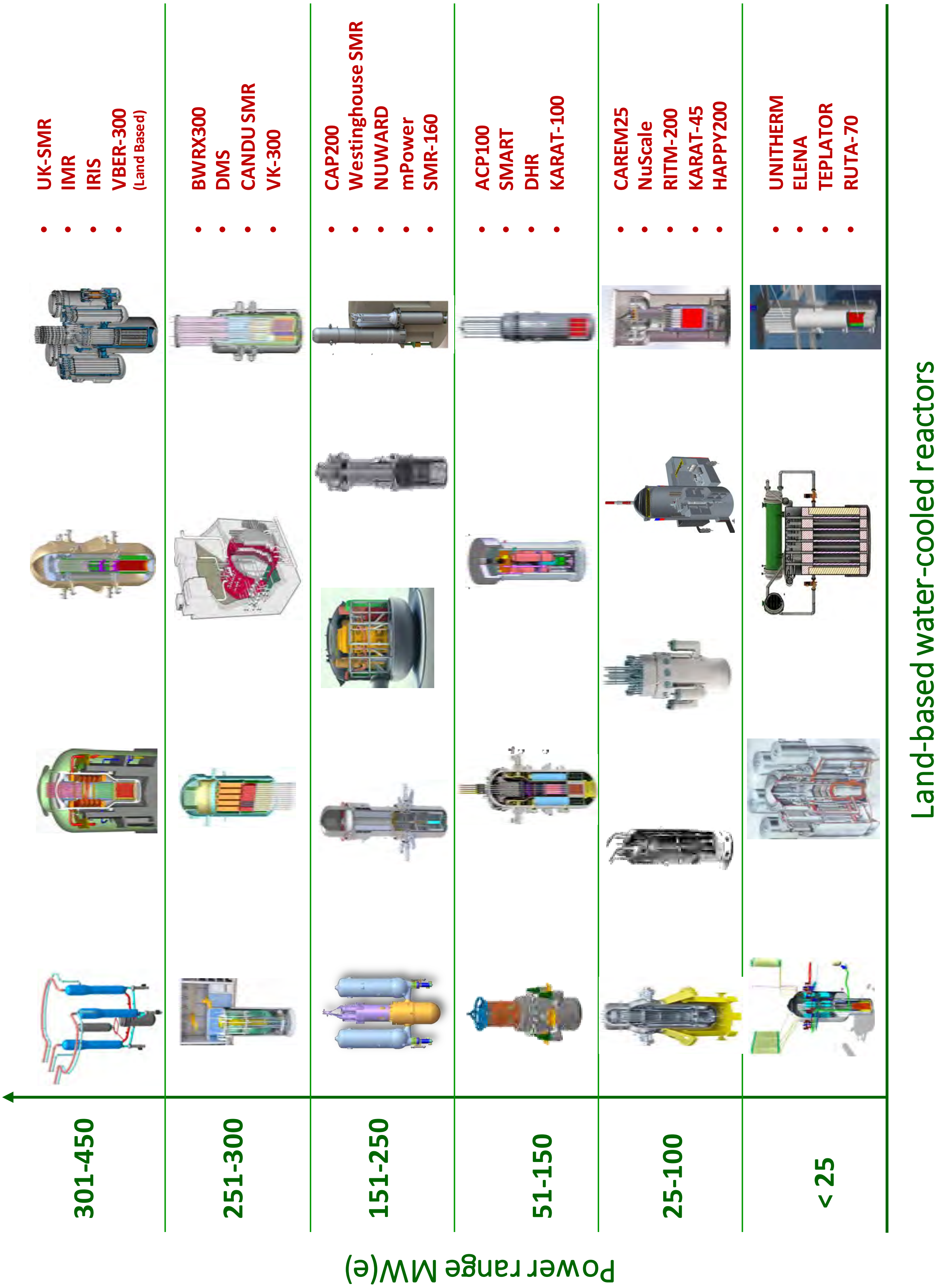


Figure II-1 Power Range of Land-based SMRs

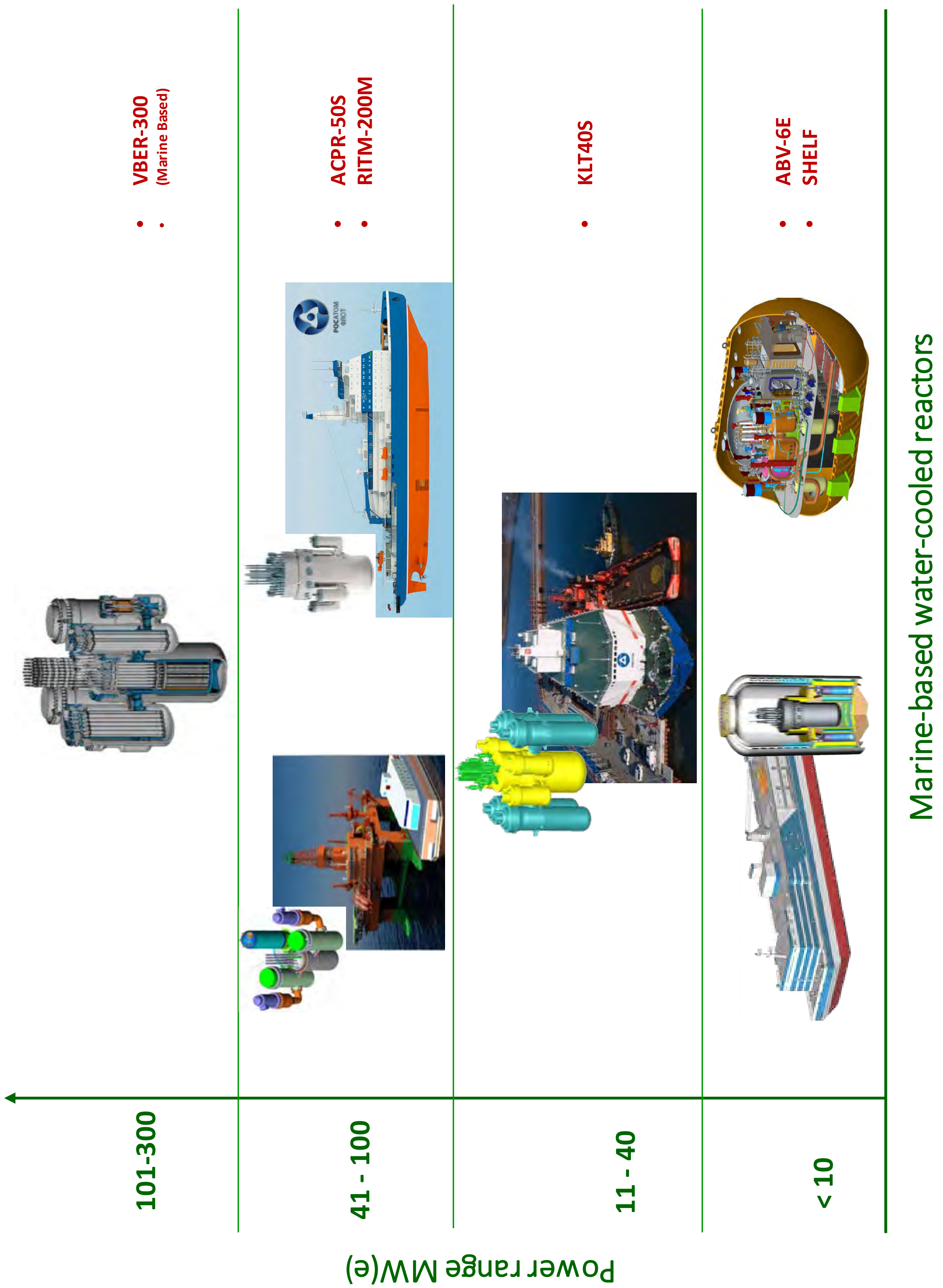


Figure II-2 Power Range of Marine-based SMRs

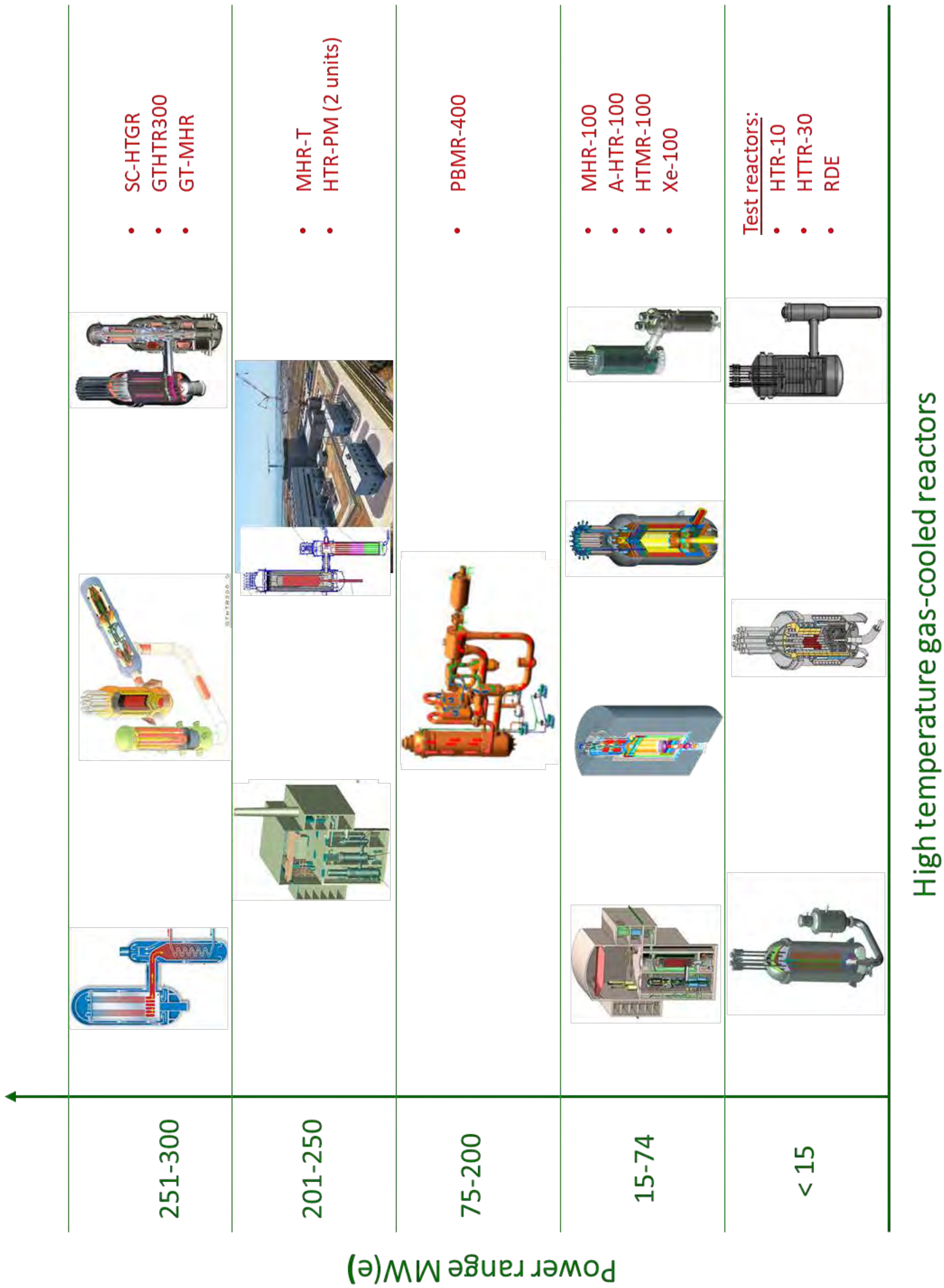


Figure II-3 Power Range of HTGR-SMRs

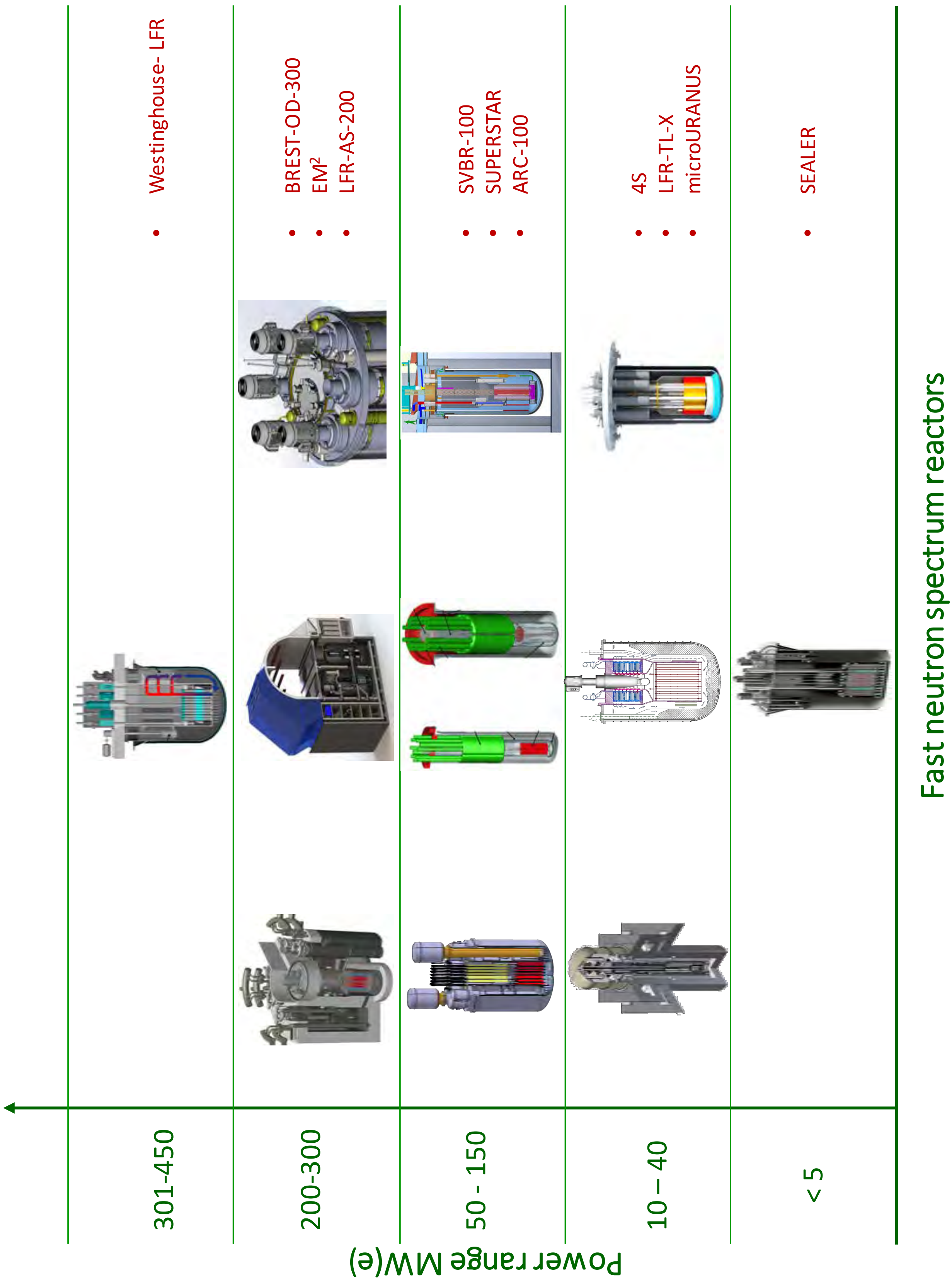


Figure II-4 Power Range of Fast Neutron Spectrum-SMRs

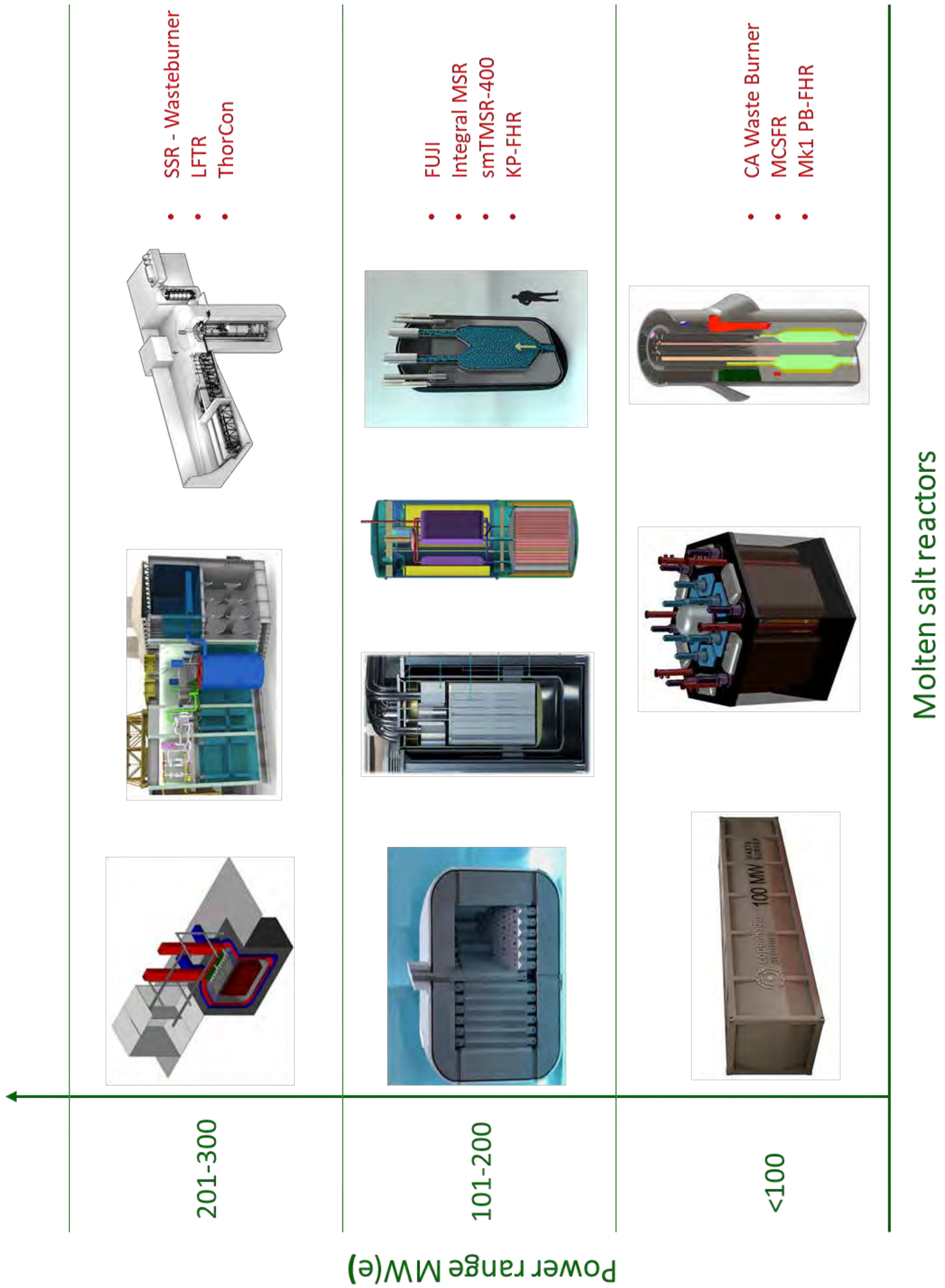


Figure II-5 Power Range of Molten Salt-SMRs

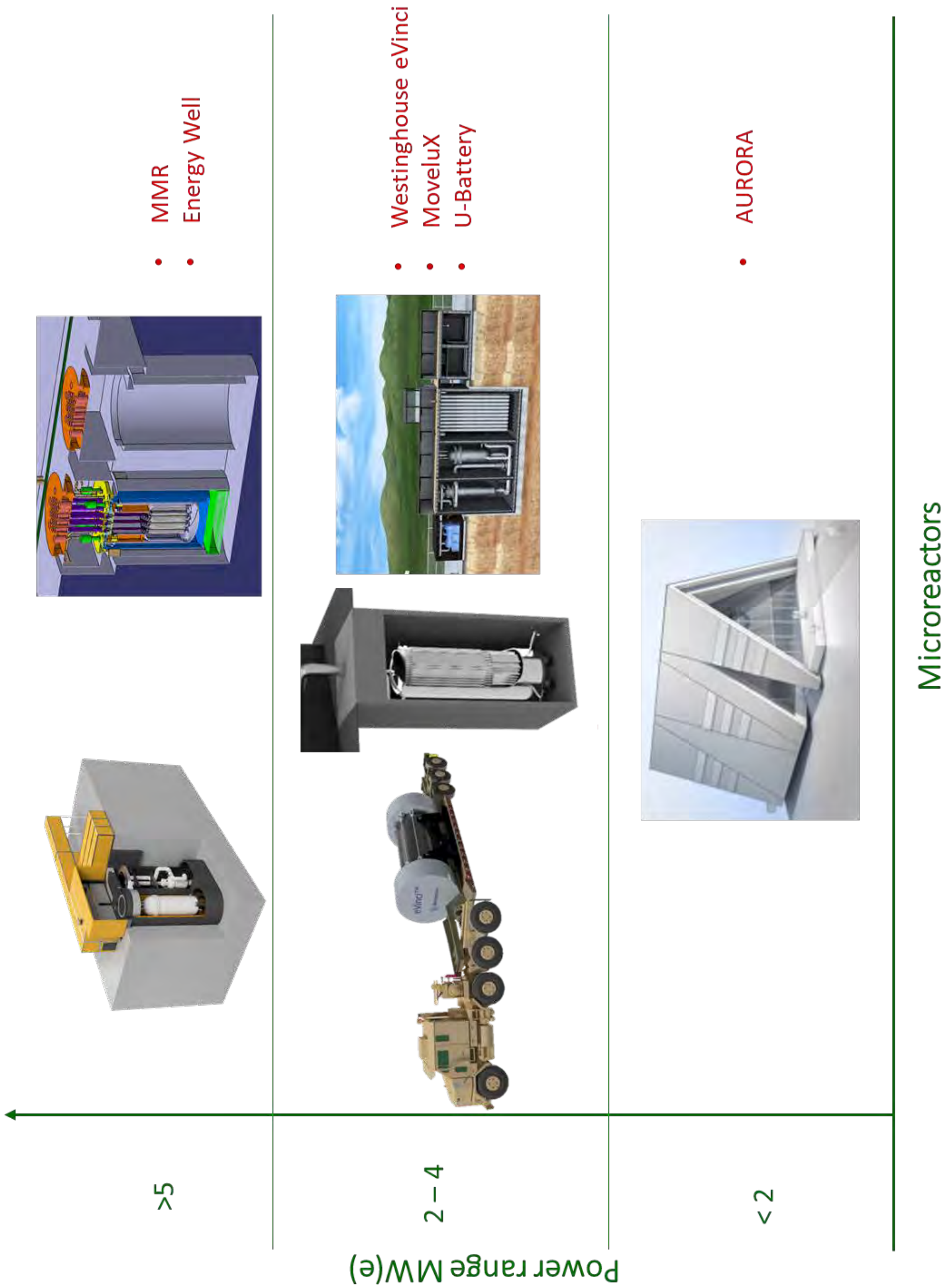


Figure II-6 Power Range of Microreactors

ANNEX III
Comparison of Main Technical Characteristics among several SMR Designs

Table III-1: Comparison of Main Characteristics among Land-based Water-cooled SMR Designs

	CAREM	ACP100	CANDU SMR	NUWARD	SMART	UK-SMR	NuScale	BWRX-300
Country of Origin	Argentina	China	Canada	France	Republic of Korea & Saudia Arabia	United Kingdom	United States of America	United States or Americ and Japan
Design organization(s)	CNEA	CNNC/NPIC	Candu Energy	EDF	KAERI, K.A.CARE	Rolls Royce, Plc.	NuScale Power, Inc.	GE Hitachi & Hitachi GE Nuclear Energy
Reactor Type / Primary Circulation	Integral PWR / Natural Circulation	Integral PWR / Forced Circulation	PHWR with / Calandria	Integral PWR / Forced Circulation	Integral PWR / Forced Circulation	3-loop PWR / Forced Circulation	Integral PWR / Natural Circulation	Boiling Water Reactor / Natural Circulation
Fuel type/assembly array	UO_2 pellet / hexagonal	UO_2 pellet / 17x17 square	37-element fuel bundle	UO_2 pellet / 17x17 square	UO_2 pellet / 17x17 square	UO_2 pellet / 17x17 square	UO_2 pellet / 17x17 square	UO_2 pellet / 10x10 array
Number of fuel assembly	61	57	2064 bundles	76	57	121	37	240
Coolant	Light water	Light water	Heavy water	Light water	Light water	Light water	Light water	Light water
Moderator	Light water	Light water	Heavy water	Light water	Light water	Light water	Light water	Light water
Thermal output, MW(t)	100	385	960	2 x 540	365	1276	200	870
Electrical output, MW(e)	30	125	300	2 x 170	107	443	60 (gross)	270-290
Core inlet temp., ºC	284	286.5	-	380	296	296	265	270
Core outlet temp., ºC	326	319.5	310	307	322	327	321	287
Enrichment, %	3.1	< 4.95	Natural uranium, not enriched	< 5	< 5	4.95 (max)	< 4.95	3.40 (avg) / 4.95 (max)
Fuel cycle, months	24	24	Online refuelling	24	30	18 – 24	24	12 – 24
Reactivity control	Control rods	Control rods + Gd_2O_3 solid burnable	Zone controllers + adjusters	Control rods + Gd_2O_3 solid burnable	Control rods + Soluble boron	Control rods + Gd_2O_3 solid burnable	Control rods + Soluble boron	Rods, Solid Burnable Absorber (B_4C, Hf, Gd_2O_3)
Reactor Vessel's height/diameter, (m)	11 / 3.2	10 / 3.35	N/A - Calandria	13 / 4	18.5 / 6.5	11.3 / 4.5	17.7 / 2.7	26 / 4
Design status	Under construction as Prototype	Detailed Design for construction	Conceptual	Conceptual	Licensed / Standard design approval	Conceptual, Pre-Licensing in the UK	Under design certification review	Pre-Licensing in UK, Canada, and the United States

Table III-2 Comparison of Main Characteristics among Marine-Based Water-cooled SMR Designs

	KLT40S	ACPR50S	RITM-200M	SHELF
Country of Origin	Russian Federation	China	Russian Federation	Russian Federation
Design organization(s)	OKBM Afrikantov	CGNPC	OKBM Afrikantov	NIKIET
Reactor Type	PWR	2-loop PWR	Integral PWR	Integral PWR
Fuel type/assembly array	UO_2 pellet in silumin matrix	UO_2 pellet / 17x17	UO_2 pellet / hexagonal	UO_2 pellet / hexagonal
Number of fuel assembly	121	37	241	163
Coolant	Light water	Light water	Light water	Light water
Moderator	Light water	Light water	Light water	Light water
Thermal output, MW(t)	150	200	175	28.4
Electrical output, MW(e)	35	50	50	6.6
Core inlet temp., ºC	280	299.3	277	270
Core outlet temp., ºC	316	321.8	318	310
Enrichment, %	18.6	< 5	< 20	19.7
Fuel cycle, months	30 – 36	30	Up to 120	Up to 160
Reactivity control	Control rods	Control rods + boron solution	Control rods	Control rods
Reactor Vessel's height/diameter, (m)	4.8 / 2	7.2 / 2.2	8.6 / 3.45	3 / 1.2
Design status	2 reactor-module in Operation in Akademik Lomonosov Floating NPP in Pevek, Russia	Detail Design	Deployed in 6 prototype icebreakers, land-based version available	Basic Design

Table III-3 Comparison of Main Characteristics among High Temperature Gas-cooled SMR Designs

	HTR-PM	GTHTR300	GT-MHR	PBMR-400	Xe-100	SC-HTGR
Country of Origin	China	Japan	Russian Federation	South Africa	United States of America	United States of America
Design organization(s)	INET, Tsinghua University	JAEA	OKBM Afrikantov	PBMR SOC, Ltd.	X Energy, LLC	Framatome
Reactor type	Modular pebble bed HTGR	Prismatic HTGR	Modular Helium Reactor	Modular HTGR	Modular HTGR	Prismatic HTGR
Fuel materials	TRISO Spherical elements with coated particle fuel	UO_2 TRISO ceramic coated particle	Coated particle fuel in compacts, hexagonal prism graphite blocks	Pebble bed with coated particle fuel	UCO TRISO/pebbles	UCO TRISO particle fuel in hexagonal graphite blocks
Coolant	Helium	Helium	Helium	Helium	Helium	Helium
Moderator	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite
Thermal output, MW(t)	2 x 250	<600	600	400	200	625
Electrical output, MW(e)	210	100~300	288	165	82.5	272
Core inlet temp., ºC	250	587-633	490	500	260	325
Core outlet temp., ºC	750	850-950	850	900	750	750
Enrichment, %	8.5	14	14-18% LEU or Weapon grade Pu	9.6% LEU or WPu	15.5	14.5 (avg) 18.5 (max)
Core Discharge Burnup (GWd/ton)	90	120	100-720 (depends on fuel type)	92 (depends on fuel cycle)	165	165
Fuel cycle, months	Online refuelling	48	25	Online refuelling	Online refuelling	½ core replaced every 18 months
Reactivity control	Control rods	Control rods	Control rods	Control rods	Control rods	Control rods
Reactor Vessel's height/diameter, (m)	25 / 5.7 (inner)	23 / 8	29 / 8.2	30 / 6.2	16.4 / 4.88	24 / 8.5
Design status	Finalizing construction	Pre-licensing basic design completed	Preliminary Design completed	Basic Design	Design Certification Review	Conceptual

Table III-4 Comparison of Main Characteristics among Liquid metal-cooled fast neutron spectrum SMR Designs

	BREST-OD-300	4S	SVBR-100	EM^2	ARC-100
Country of Origin	Russian Federation	Japan	Russian Federation	United States	United States
Design organization(s)	NIKIET	Toshiba Corporation	JSC AKME Engineering	General Atomics	
Reactor Type	Liquid metal cooled fast reactor	Liquid metal cooled fast reactor (pool)	Liquid metal cooled fast reactor	Modular high temperature gas-cooled fast reactor	Liquid metal cooled fast reactor (pool)
Fuel type/assembly array	Mixed uranium plutonium nitride	U-Zr alloy	UO_2 / hexagonal	UC pellet / hexagon	U-Zr alloy
Number of fuel assembly	169	18	61	85	99
Coolant	Lead	Sodium	Lead-bismuth eutectic alloy	Helium	Sodium
Thermal output, MW(t)	700	30	280	500	286
Electrical output, MW(e)	300	10	100	265	100
Core inlet temp., ºC	420	355	340	550	355
Core outlet temp., ºC	535	510	485	850	510
Enrichment, %	up to 14.5	< 20	< 19.3	~14.5 (LEU)	13.1
Core Discharge Burnup (GWd/ton)	61.45	34	60	~130	77
Fuel cycle, months	900-1500 days	N/A	7 – 8 years	360	20 years
Reactivity control	Shim and automatic control rods	Movable reflectors / fixed absorber	Control rods	Control rods	Control rods
Reactor Vessel's height/diameter, (m)	17.5 / 26	24 / 3.5	8.2 / 4.53	12.5 / 4.6	15.6 / 7.6
Design status	Detailed design aims for startup in 2026	Detialed design	Detailed design aims for 2026 construction	Conceptual design	Conceptual design

Table III-5 Comparison of Main Characteristics among Molten Salt-cooled SMR Designs

	Integral MSR	CA Waste Burner	ThorCon	FUJI	Stable Salt Reactor	KP-FHR	MCSFR
Country of Origin	Canada	Denmark	Indonesia	Japan	United Kingdom, Canada	United States	United States, Canada
Design organization(s)	Terrestrial Energy, Inc.	Copenhagen Atomic	Martingale, ThorCon Consortium	International Thorium Molten Salt	Moltex Energy	Kairos Power	Elysium Industries
Reactor Type	Molten salt reactor	Molten salt reactor	Thermal molten salt reactor	Molten salt reactor	Static fuelled molten salt fast reactor	Modular, pebble bed, high temperature, salt-cooled reactor	MSR – Fast Chloride
Fuel type/assembly array	Fluoride fuel salt	LiF-ThF_4	UF_4, ThF_4	Molten salt with Th and U	Molten salt fuel in a conventional hexagonal array fuel assembly	TRISO particles in graphite pebble matrix / pebble bed	Molten Chloride Salt
Coolant	Fluoride salt	Fuel salt	NaF, BeF_2 Molten Salt	Molten fluoride	ZrF4/KF/NaF molten salt	Li_2BeF_4 Fluoride salt	NaCl-XCl_2-$UCl_{3/4}$-$PuCl_3$-$FPCl_y$ fuel salt
Moderator	Graphite	Heavy water	Graphite	Graphite	-	Graphite	-
Thermal output, MW(t)	440	0.05 – 0.25	557	450	750	320	125
Electrical output, MW(e)	195	0.1 – 0.25	250	200	300	140	50
Core inlet temp., ºC	620	600	565	565	525	550	650
Core outlet temp., ºC	700	650 - 700	704	704	590	650	750
Enrichment, %	< 5	Inventory of spent nuclear fuel	5.0 (min) 19.7 (max)	2.0 (0.24%U_{233} + 12.0%Th)	4.95	19.75	10% Pu fissile/(Pu+U total) or ~15% enriched HALEU
Fuel cycle, months	84	Continuous	48	Continuous	150	Online refuelling	Online refuelling
Reactivity control	Negative Temperature Coefficient	D_2O level adjustment	Salt flow rate	Control rod, pump speed	Boron carbide	Control Rods	Fuel expansion in/out of core
Reactor Vessel's height/diameter, (m)	10.0 / 3.67	12 / 2.4	10.3 / 7.8	5.4 / 5.34	5 / 5 (width)	15 / 3	9.0 / 4.0
Design status	Basic engineering in progress	Conceptual	Basic design completed	Basic	Conceptual	Conceptual	Conceptual

Table III-6 Comparison of Main Characteristics among Microreactor Designs

	eVinci™	MMR	U-Battery *	Aurora	MoveluX™
Country of Origin	United States	United States	United Kingdom	United States	Japan
Design organization(s)	Westinghouse	Ultra Safe Nuclear Corporation	Urenco	OKLO	Toshiba Corporation
Reactor type	Monolithic core with heat-pipe technology	High Temperature Gas-cooled Reactor	High Temperature Gas-cooled Reactor	Fission battery, fast spectrum	Heat-Pipe cooled and calcium-hydride moderated reactor
Coolant	Heat pipes	Helium	Helium (primary) Nitrogen (secondary)	Liquid metal	None (Sodium heat-pipe cooled)
Moderator	Metal hydride	Graphite	Graphite	N/A	Calcium hydride (CaH_2)
Thermal output, MW(t)	7 – 12	15	10	4	10
Electrical output, MW(e)	2 – 3.5	5	4	1.5	3 - 4
Core inlet temp., ºC	N/A	?		N/A	680
Core outlet temp., ºC	N/A	630		N/A	685
Fuel type/assembly array	TRISO or another encapsulation	FCM	TRISO	metal UZr	Silicide (U_3Si_2) / Hexagonal
Enrichment, %	5 - 19.75	9 – 12		19.75	4.8 – 5.0
Fuel cycle, months	> 36	Continuous		Up to 20 years	Continuous
Reactivity control	Ex-core control drums	Control rods		N/A	Lithium Expansion Module
Reactor Vessel's height/diameter, (m)	N/A	N/A		N/A	2.5 / 6.0
Design status	Conceptual	Conceptual/Basic	Conceptual	Accepted combined license application by the US NRC	Conceptual

ANNEX IV
SMR and Non-Electric Applications

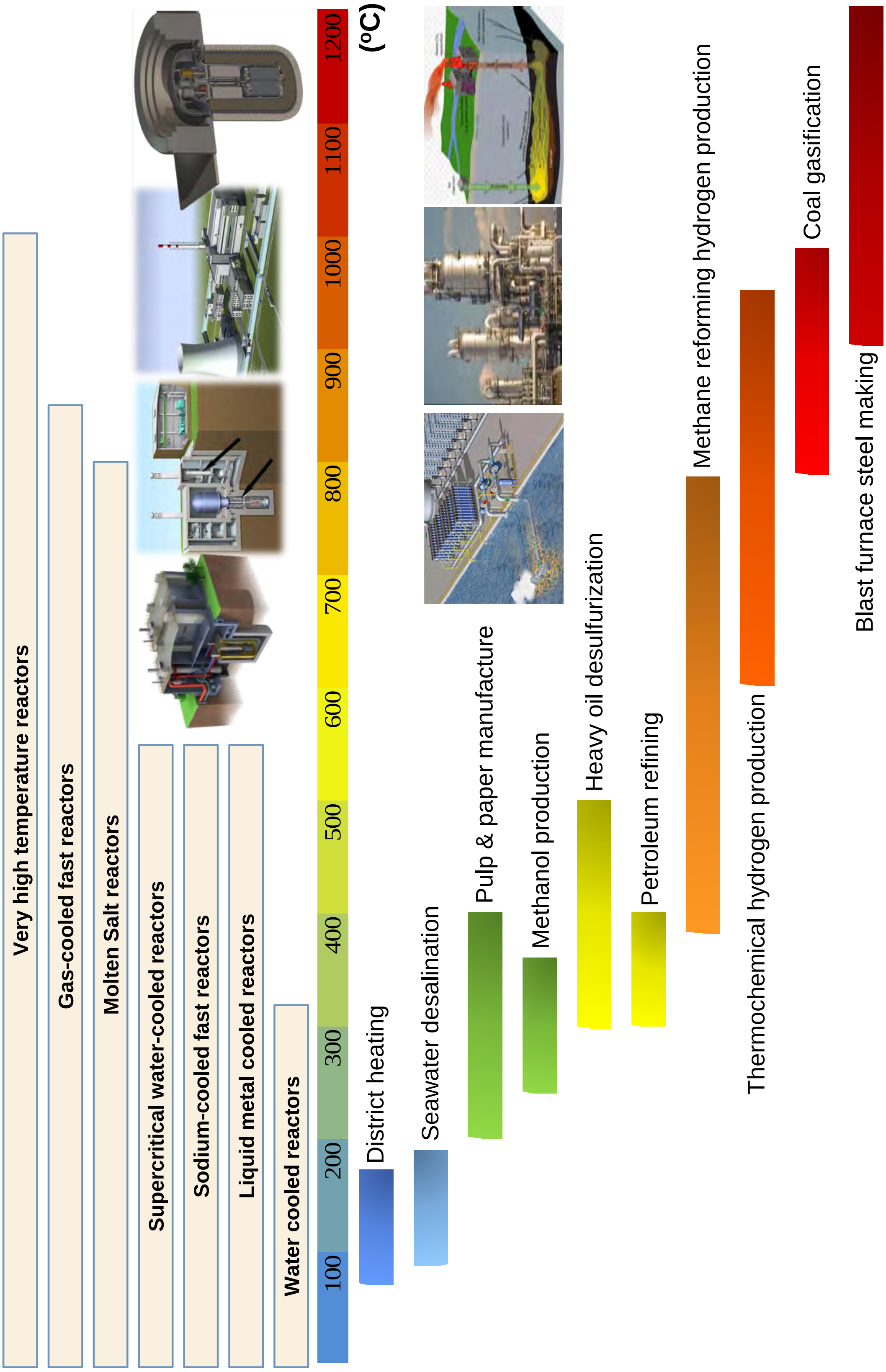


Figure IV-1 Exit Temperature of SMR Designs and their Corresponding Non-Electric Applications

ANNEX V
Dimensions of Reactor Vessels

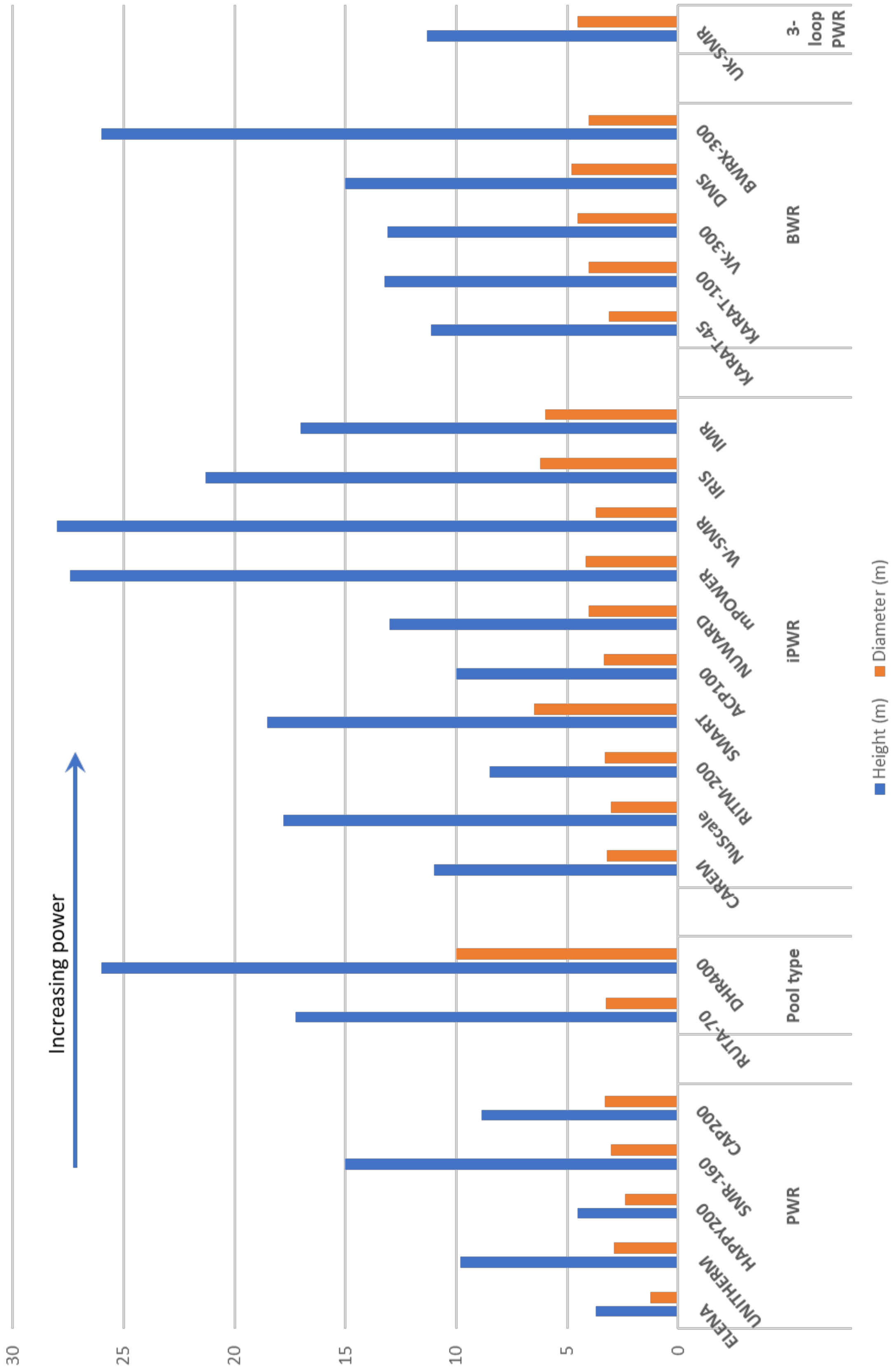


Figure V-1 Dimension of Reactor Vessel of Land-Based Water-cooled SMRs

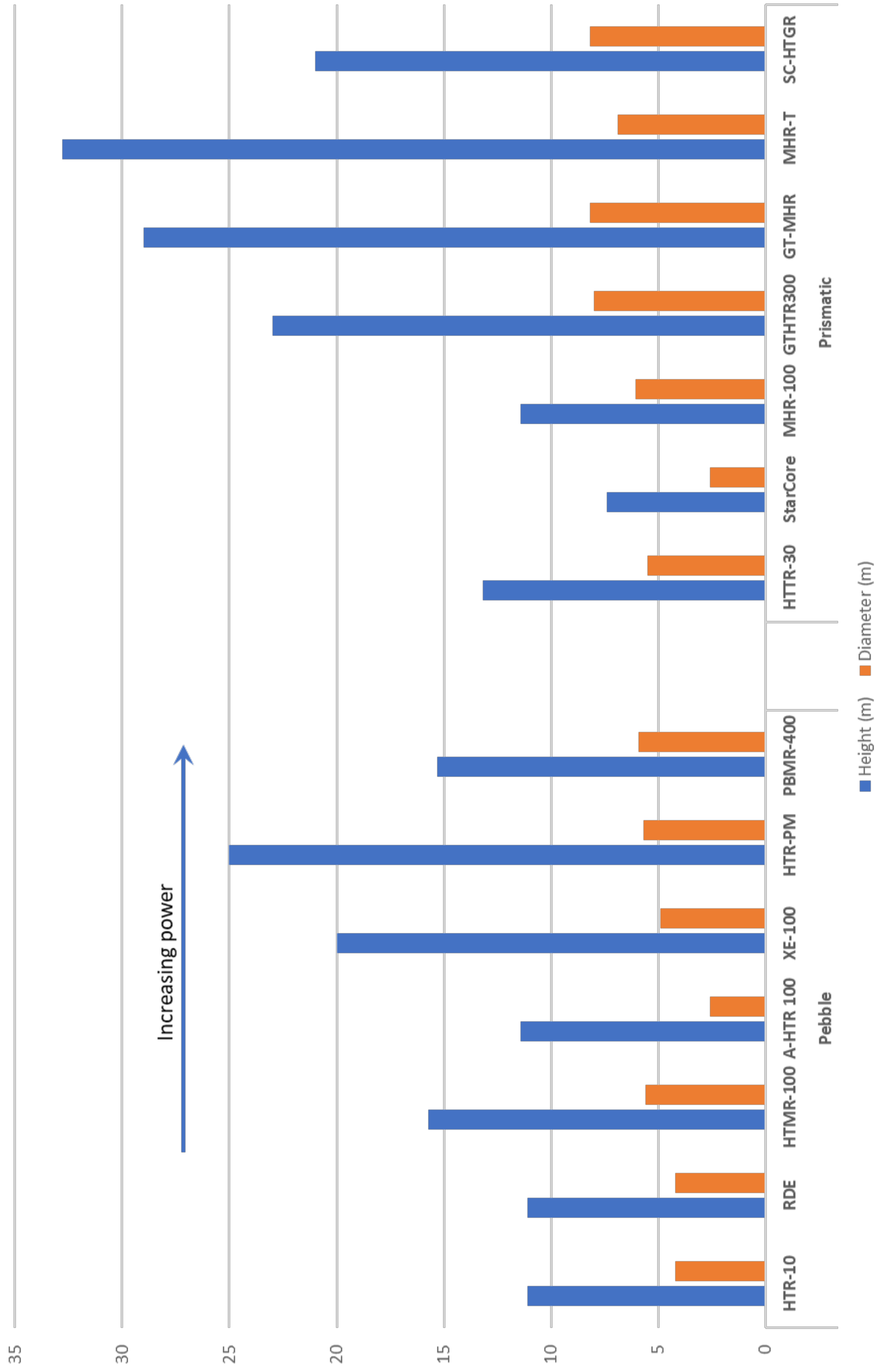


Figure V-2 Dimension of Reactor Vessel of HTGRs

ANNEX VI
Fuel Cycle Approach adopted in SMRs

Introduction

The responsibility for designing and deploying SMRs with a view to full life-cycle impact is well understood, and in particular the associated spent fuel management (SFM), radioactive waste management (RWM) and eventual decommissioning responsibilities are considered from the early design stage. To the extent that certain SMR designs concepts, associated fuel cycle decision and future operational procedures are similar to those used in current operating large NPPs, the SFM and RWM activities can most probably be implemented consistently. The chosen fuel cycle and other features of a few SMRs may further contribute to the reduction of the SFM and RWM needs addressed in the next annex.

A better understanding of the specific fuel cycle approach being proposed by various SMR designs is very important to stakeholders in Member States. Particularly, the knowledge of the fuel cycle will facilitate discussion on the front end requirements and fuel supply, the potential management routes for spent fuel and waste forms (discussed in the annex below), the potential needs for a final waste repository, and Safeguard-by-Design implementation by SMR designers. It should be recognised that fuel cycle choices are often made by the state and this may place restrictions on vendors that wish to deploy a SMR in a country.

SMRs are not only of PWR/BWR water-cooled based reactor types but include many designs utilizing other coolant types or neutron spectra, including the HTGRs, FRs and MSRs represented in this booklet. Many of these designs adopt an innovative approach for their fuel cycle. Most of the SMR designs propose longer operation cycles between refuelling. This does not only benefit availability and capacity factors but at the same time raise issues on fuel (and cladding where applicable) performance (at higher burnup), other aspects such a core maintainability on-line access to service key components and the different steps for managing the spent fuel (mainly storage, transport and recycling routes). Some of the new fuel cycles and new fuels proposed include Th/U_{233}, reprocessed uranium, MOX, Pu disposition, transuranic waste (TRU) fuels and the direct use of spent LWR fuel. Other innovations, like the use of pyro-processing is another area of interest related to SMR fuel reprocessing and recycling. It is important to highlight that some of these innovative routes are not industrially implemented, a few are at demonstration stage and most of them at R&D level, with a low TRL.

Land-based SMRs

For water-cooled SMRs, various open fuel cycles are adopted depending on the design approach and objectives. Water-cooled based SMRs typically utilize existing LWR fuel design with U_{235} enrichment below 5% to achieve a range of discharged burnups as low as 30 GWd/tHM for district heating reactors, and between 40 up to 75 GWd/tHM for electricity generation, to attain refuelling cycles of 18, 24, 36 or even up to 48 months. The ultimate objective is to have an optimized capacity factor of higher than 90% while minimizing the operation and maintenance (O&M) costs. Longer fuel cycles improve the overall plant availability but could be at the expense of a somewhat reduced discharge burnup.

Typical PWR-type SMR use fuel with enrichment lower than 4.95%, propose a refuelling cycle of 18-24 months and replace half of the assemblies every cycle to optimize the overall fuel economics while maximizing the discharge burnup. A 3-batch refuelling design, on the other hand, improves the discharge burnup and fuel utilization. The target refuelling outage is between 18 to 36 days. A natural circulation integral-PWR design for a multi-unit plant, for example, adopts a 24-month refuelling cycle and a 3-batch in-out refuelling scheme. This means that during the refuelling process, one-third of the fuel assemblies are removed from a reactor module and placed in the spent fuel pool, one-third fresh assemblies are introduced while at the same time the remaining assemblies are re-shuffled in the core.

Recently, a new SMR design of the CANDU-type pressurized heavy-water cooled heavy-water moderated was introduced into this Booklet. The standard design uses a once-through open fuel cycle based on natural uranium, producing a very low residual uranium spent fuel with low heat generation. It is also designed to be able to burn recovered uranium from LWRs, a valuable addition to any light-water fleet. The use of MOX and Thorium fuels is also possible with proper customization.

There are also converted PWR type SMR designs originally deployed for marine-based SMRs. These designs have enrichment close to the 20% limit of LEU to ensure long time operation in remote areas while still satisfying international proliferation goals. The target refuelling cycle is as long as 72 – 84 months. In the framework of transportable SMR design, a factory-fabricated reactor vessel can be delivered to the site loaded with fresh fuel. This initial load is designed to provide the whole NPP lifetime without refuelling.

This booklet also reports three (3) BWR-type SMRs with natural circulation. Typically, the standard BWR fuel designs with an open cycle is used. During the refuelling outage 15% to 25% of the bundles in the core are replaced by fresh fuel with the result that they stay in the core for several cycles, until they are discharged. When removed from the core, spent fuel is stored in the fuel pool inside the reactor building for 6 to 8 years then it is transferred to storage casks that can be removed from the reactor building and stored outside.

In general, the water-cooled SMR power plants provide a spent fuel pool using storage racks for 6 to 10 years after operation before the decommissioning. A longer storage option of up to 20-30 years could be proposed for some designs, depending on owner's requirements. The spent fuel pools are connected to the ultimate heat sink through dedicated engineered safety features. The spent fuel storage racks include not only sufficient storage for many years of operation, but also to store potential defective fuel assemblies and for non-fuel core components such as control rod assemblies.

Ultimately, the fuel cycle can be tailored to customer requirements in compliance with the regulatory requirements both in the country of origin and host countries.

Marine-based water-cooled SMRs

The first SMR design that completed construction, was connected to the grid and is now in commercial operation is the marine-based PWR-type KLT-40S of the Akademik Lomonosov floating power unit, deployed in Pevek, Russian Federation. The objective of the design is to supply cogeneration of electricity and process heat in difficult-to-access remote regions. Therefore, the adopted fuel cycle should be commensurate to the purpose of deployment. In Russia, there are at least five (5) marine based SMRs designs (included in this booklet) and their variants. To achieve a long refuelling cycle of 30-36 months, a near 19% fuel enrichment is required and 45 GWd/tHM core discharge burnup is achieved. Refuelling is performed 14 days after reactor shutdown when the levels of residual heat releases from spent fuel assemblies have come down to a required level. No special maintenance or refuelling ships are necessary. A single batch fuel loading is done to attain a maximum operation period between refuelling. In other words, fresh fuel is loaded in all the core positions replacing the burned fuel assemblies.

The RITM-200M based optimized floating power unit (OFPU) is also designed as a transportable SMR power plant that has a refuelling cycle up to 120 months. The OFPU is delivered to the site with fresh fuel in its reactors. After completion of the fuel cycle, the OFPU returns to the country of origin together with the spent fuel in its reactors. All operations for production, post-reactor maintenance and reprocessing of spent nuclear fuel are performed in the country of origin.

HTGR-type SMRs

The objective of modular high temperature gas cooled reactor (mHTGR) designs is catastrophe free SMRs built on inherent safety characteristics with the focus on accident free fuel. The mHTGR comes either as pebble-bed or prismatic core designs and adopts TRISO coated particle fuel. The next SMR to start operation is in China and expected in 2021 is the 2-unit pebble bed type HTR-PM that can generate

210 MW(e). Either online refuelling (pebble bed) or long refuelling cycles (prismatic) of 25 to 48 months are adopted by near-term deployable mHTGRs with the U_{235} enrichment ranging from 8.5% to 18.5%. The typical discharge burnup of the mHTGR is substantially higher than that of typical water-cooled SMRs. It ranges between 80 to 720 GWd/tHM depending on fuel type and fuel cycle. The higher enrichment is needed for the increased burnup target but also to compensate for the typically larger neutron leakage from this graphite moderated and reflected core with its low-power density but also relatively small active core diameters (needed to limit fuel accident temperatures). Although the core diameters are physically larger than typical LWR SMRs, the core is neutronically much smaller leading to the aforementioned higher nett leakage.

Both open and close fuel cycles can be adopted with options including UO_2, MOX, Pu-burning (or disposition) and the thorium cycle. Most of these fuels have been developed, manufactured and demonstrated (or at least tested) in the past HTGR projects. In HTR-PM, for example, if recycling is adopted, the spent fuel spheres would be dismantled, and the nuclear fuel would be reprocessed (in normal reprocessing facilities, that may have to be adjusted to receive the material). Currently there are still no routes available or decided to manage graphite wastes.

Most mHTGR reactor will start with a once through open cycle and no reprocessing will be done. The generated spent fuel shows some of the favourable proliferation resistance characteristics of these reactors, one being that the total plutonium and Pu_{239} assay (per unit of energy produced) is much lower due to better in-situ utilization. In the case of pebble bed reactors, fuel that has not reached the target discharge burnup is recycled again through the reactor and remains in the closed safeguarded fuel handling system until it is classified as spent fuel; and then deposited into a spent fuel cask.

With higher thermal efficiency and high fuel burnup, mHTGRs support sustainability for once-through fuel cycles. Most of the core designs are 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. The TRISO coated particle fuel could also be recycled when and if such process is mandated and economically viable, although the reprocessing routes are still at R&D and demonstration stages.

Fast Neutron Spectrum-type SMRs

This booklet reports a dozen of liquid-metal cooled fast neutron spectrum SMRs and a gas-cooled fast SMR. In general, compared to water-cooled SMRs, FR-type SMRs have a higher enrichment of ranging from 14% to slightly less than 20% U_{235} (some fuels also with plutonium loadings) with core discharge burnup in the range of 60 to 80 GWd/tHM to realize longer fuel cycles of up to 30 years.

Within the SMR category, BREST-OD-300 is a lead-cooled fast neutron reactor facility to start construction in Seversk, Russia with completion schedule by 2026. The project is to enable a closed nuclear fuel cycle (CNFC) for full utilization of the energy potential of natural raw uranium. Mixed nitride fuel with high density and thermal conductivity ensures optimum reproduction of fuel in the core (core reproduction ratio ~ 1.05) to compensate for burnup. The fuel type considered for the first core and the first partial fuel reloads of the BREST-OD-300 fast reactor is nitride of depleted uranium mixed with plutonium, whose composition corresponds to that of irradiated spent fuel from VVER's following reprocessing and subsequent cooling for about 25 years. After completion of the initial stage the reactor will operate a closed fuel cycle. For the production of new fuel, it uses the products from the reoprocessing of its own spent fuel.

The EM^2 gas-cooled fast reactor uses an open fuel cycle with LEU/DU vented fuels that can exceed 30 years operation without refuelling or shuffling. This should lead to reduced cost and decreased proliferation risk, while achieving high fuel utilization with low mass of waste streams. The core is capable of burning used LWR, plutonium, and thorium fuels. The spent fuels are stored in storage facility at site. After cooling, the used fuel will be directly disposed of or recycled. When the closed fuel cycle scheme is applied, the spent fuel would undergo pyroprocessing for metallic fuels and would be refabricated as fresh fuel for reuse. There is no reprocessing plant based on pyro technology under operation so far. This technology is still under demonstration stage with low TRLs and technical issues to be solved as materials corrosion, etc.

Molten salt fuelled SMRs

Liquid fuel molten salt reactors (MSRs) have different characteristics from that of solid-fuelled reactors. In MSRs, as no fuel structure or cladding is required, the fuel is not subject to failures due to high burnup or mechanical damage. The fuel is already in a molten state so there is no risk of fuel melting (with the severe consequences that may challenge the vessel integrity and possible fission products release). Molten-salt reactors reported in this booklet adopts different fuel cycles that aim for longer fuel cycle up to 150 months, online refuelling (adding of fuel containing molten salt) and continuous operation. Enrichment levels vary from those designs with less than 5% enrichment to some with a higher-level enrichment up to 19.7%.

In the Integral MSR, that has recently undertaken a vendor design review in Canada, the core-unit will be replaced at the end of its 7-year fuel cycle. The de-fuelled core-unit is then moved by 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 its place ready to begin operations when needed (or once the second operating core-unit unit reaches its end-of-cycle and is shut down).

The Copenhagen Atomics Waste Burner has an initial fuel composition of LiF-ThF_4-PuF_4, where the U_{233} production benefits from the high number of excess neutrons from the plutonium fissioning. As the thorium fuel cycle converges towards equilibrium the breeding process benefits from U_{233} superior neutron economy.

Another MSR design is ThorCon, a thorium converter. It adopts enriched uranium of minimum 5% and maximum 19.7%, with discharge burnup of up to 12.4 GWd/tHM. The ThorCon core will require 5.3 kg of 19.7% enriched uranium and 9.0 kg of thorium per day to be added to the core to realize the 8-year core life. During the 8-year fuel cycle, a portion of the fertile thorium is converted to fissile U_{233} which then becomes part of the fuel.

General issue of this fuel cycle is the management of the spent fuel which is a mixture of fuel and molten salt difficult to manage as a high-level waste and the potential reprocessing routes through pyro technologies.

Microreactors

The microreactors reported in this booklet are of different families, heat-pipes, HTGR, molten salt cooled HTR, and liquid-metal cooled fast reactor. The range of electrical output is between 1.5 to 5 MW(e), the adopted enrichment is between 4% up to 19.75%, with long fuel cycle between 36 months up to 20 years. Limited information of the fuel cycle aspects of microreactors are available.

The eVinci uses heat pipes for heat transfer and HALEU Uranium Oxycarbide (UCO) in tristructural isotropic (TRISO) encapsulated fuel is one option that can be used. After 3 years of operation without refuelling, the microreactor will be disconnected and transported back to the factory in its original canister for either refuelling and redeployment or for long-term storage.

Another product called the Micro Modular Reactor (MMR) uses Fully Ceramic Micro-encapsulated (FCM™) fuel that replaces the graphite matrix around the typical TRISO coated mHTGR fuel. This creates an extra barrier to fission product release and improves each TRISO particle’s structural and containment characteristics. The reactor will be fuelled only once in its lifetime of 30 years.

Table VI-1 Fuel Cycle Approach adopted by SMR Designs

Fuel Cycle Approach Categories	SMR designs by type of coolants and technology characteristics				
	Water-cooled Reactors	**HTGRs**	**Liquid-metal cooled and Fast Reactors**	**Molten Salt Reactors**	**Microreactors**
Open Fuel Cycle	CAREM, ACP100, SMART, NuScale, CANDU-SMR	HTR-PM, GTHTR300, PBMR, GT-MHR, Xe-100, SC-HTGR	EM^2	Integral MSR, SmTMSR-400	All designs
Close Fuel Cycle	SHELF		BREST-300-OD, 4S, SVBR-100	FUJI, LFTR, CA Waste Burner (later generation), and MCSFR	eVinci®, MoveluX
Longer Refuelling Cycle > 24 months	SMART, HAPPY200, ABV-6M, RITM-200, SHELF	HTR-PM (online refuelling), GTHTR300	MicroURANUS, W-LFR, SEALER and EM^2	CA Waste Burner	
Enrichment < 5%	CAREM, NuScale, VBER, NUWARD and ACPR50S			Integral SMR Stable Salt Reactor	MoveluX
5% ≤ Enrichment ≤ 15%		HTR-PM, PBMR, GTHTR300	BREST-300-OD, 4S, EM^2, ARC100, Superstar	ThorCon	Energy Well
Enrichment > 15%	KLT-40S, RITM, SHELF, ABV-6M	MHR-T, MHR-100, GT-MHR, SC-HTGR, Xe-100,	SVBR, SEALER, LFR-TL-X, W-LFR	ThorCon	eVinci®, Aurora, MMR
Spent Fuel Processing and Conditioning			BREST-300-OF, 4S, SVBR	SmTMSR-400	
Use of Thorium-cycle and/or Plutonium Disposition		HTMR-100, GTHTR300, GT-MHR, SC-HTGR, and possibly for all	LFR-AS-200, Superstar	FUJI, LFTR, Integral MSR, CA Waste Burner, ThorCon, Moltex SSR and SmTMSR-400	
Use of Spent Fuel as Fuel		GTHTR300	BREST-300-OD	Moltex SSR and CA Waste Burner	

ANNEX VII
Spent Fuel, Waste Management and Disposal Plans adopted for SMRs

Introduction

For the first time, the IAEA booklet on SMR includes brief information on the waste management approach, technology and disposal plan adopted by SMRs. The adopted approaches depend on the particular SMR design and more importantly a country's existing spent fuel and radioactive waste management plans and practices. Of the 72 SMR designs reported in this booklet, many of them are still in conceptual design phase. Given the early stage of development, the key question is how far can SMR designs substantially reduce radioactive waste throughout the lifecycle of the plant – construction, operation and decommissioning? This is particularly relevant for designs which may generate new radioactive waste stream or for which disposition paths may not be readily available. The existing national policy and strategy of a country are also significant in determining radioactive waste management (RWM) approaches. Establishing responsibilities and funding mechanisms, as well as setting out technological and programmatic decision making on RWM are all essential considerations for the developers of SMRs.

The safe, secure and sustainable management of spent fuel and radioactive waste arising from the operations of nuclear power reactors, as well as from associated fuel cycle activities and ultimately from the decommissioning of these nuclear power reactors is key to the future of nuclear energy. The fuel cycle approach (addressed in the previous annex) is a major factor influencing the type of waste and associated properties that need to be managed; and that in addition to the specific options for reactor design and operations and ultimate decommissioning of the SMR, also influence what type of waste and waste forms are generated (i.e., waste is not limited to decision of direct disposal or reprocessing).

Technical solutions already exist ranging from reprocessing and recycling, to conditioning of the types of spent fuel currently in use, to the knowhow how to dispose spent fuel and high level waste in deep underground repositories. Partitioning and transmutation are advanced processes which have the potential to further reduce the impact of nuclear waste.

The emergence of SMRs presents an opportunity to demonstrate a new paradigm, in which effective RWM should be considered and incorporated early in the conceptual design stage. Solutions for managing spent fuel and radioactive waste arising from SMRs will be one of the important factors for Member States to consider when selecting a technology. This approach will help address uncertainties related to the back end of the fuel cycle, reduce costs and enhance public acceptance of nuclear power. The experience built up over decades, from many types of reactor (experimental, research and power), provides a most valuable resource for considering the RWM aspects of SMRs. The advances in computer codes, particularly as applied in the simulation and modelling of a wide range of phenomena in the design, performance and long-term conditions of new reactors systems, is a major enabling factor in assessing RWM during the design stage.

Some SMR designs have features that could reduce the tasks associated with spent fuel management. These designs adopt longer refuelling cycles of 3 to 7 years, in comparison to 1 to 2 years for conventional power plants. Some designs are even intended to operate for up to 30 years without refuelling. However, even in such cases, there will be some spent fuel left, which will have to be properly managed. For most of the innovative non-water cooled SMRs, the designers propose to either use the existing infrastructure or to adapt it for the new radioactive waste stream. Countries with established nuclear power programmes have been managing their spent fuel for decades. They have gained extensive experience and have proper policies, strategies and infrastructure in place. For these countries, management of spent fuel arising from SMRs, should not pose a challenge if they opt to deploy SMRs based on current technologies. Since specific plans and limits on a final repository may already be in place it may impose restrictions and a barrier to some fuel cycles and its specific waste forms.

Land-based Water-cooled SMRs

Land-based water-cooled SMR developers in general adopt RWM plans similar to that of operating advanced water-cooled reactors. Facilities for waste treatment and storage are provided. For instance, in SMART, that is planned for deployment in Saudi Arabia, the liquid radioactive waste will be processed through demineralizer package to minimize the shipment volume of solid waste; while the gaseous radwaste system performs holdup and release in a controlled manner. In water-cooled SMRs, operation without soluble boron in the primary coolant allows a significant reduction in the environmental discharges and concomitant simplification of the waste treatment systems.

Advances are also being made in the dry storage technologies. Holtec International, the developer of the SMR-160 , has developed a Multi-Purpose Canister called MPC-37 that enables the on-site storage of all spent fuel for the life of the plant within an array of the HI-STORM UMAX modules, an underground vertical storage cask design.

For decommissioning, SMR design organizations also plan systematic dose reduction to personnel in decontamination and decommissioning. In integral PWRs, the longer distances from the fuel, and the additional material and components between the active core and the RPV, can decrease the fast neutron fluence on the RPV by a significant factor compared to a loop-type PWR. This essentially eliminates vessel embrittlement, the need for surveillance coupons for periodic in-service inspection of the RPV. It will also reduce the activation of the steel components significantly.

Marine-based Water-cooled SMRs

For the Russian's floating nuclear power plants (FNPP) to be supplied to other countries, such as and the RITM-200M, onsite refuelling is not required. Spent fuel will be taken back to the supplier. The spent fuel is initially stored on board the FNPP and then will be returned to the Russian Federation.

HTGR-type SMRs

Due to higher thermal efficiencies and increased burnup, HTGRs produce about 40% less high-level waste per unit of energy produced, including significantly less plutonium, especially just one quarter of the Pu_{239} content compared to a single-pass typical LWR cycle. The storage and disposal requirements largely depend on the volume, activity and fission product decay heat, that could be up to 50 times lower (per volume) for HTGRs as compared to LWRs. In this form it may be ideal to dispose of since this may warrant a different and much more cost effective approach to packaging and final disposal (lower specific source term and heat production generation) with radioactivity already contained in coated particles with a stable silicon carbide layer (that will last for more than a million years).

If disposed as part of a larger waste programme (designed for LWRs), the direct disposal of HTGR fuel spheres or elements may take up a relative larger part of the facility than expected. In comparison the HTGR spent fuel volumes are much larger requiring much more space. For example, the fuel volumetric content of uranium in a pebble fuel sphere is significantly less than 1%. So even though spent fuel spheres may provide a stable multi-barrier containment with very low heat generation, volume reduction may be attractive. Separation of the coated particles from the graphite matrix (or fuel block) then become a more attractive option since the coated particles present excellent radioactivity containment characteristics also for long term storage or disposal. Reprocessing is also an option. Although many studies have also been performed to dispose of, condition or reprocess the large volumes of graphite from past gas-cooled reactor projects, none has been implemented on commercial scale and direct disposal may be the preferred option.

In general, HTGR-SMRs have a dedicated waste handling system to store and discharge low- and medium-level liquid and solid radioactive waste generated in all operating modes. For decommissioning, spent fuel spheres are removed from the spent fuel tanks and loaded into the spent fuel transport casks for final disposal.

Liquid metal-cooled Fast neutron spectrum SMR

The fast neutron spectrum reactors are well known for their potential to substantially reduce the total radiotoxicity of nuclear waste. Most long-lived waste is re-cycled and burned as fuel. This is still a major consideration in pursuing fast spectrum SMRs but other aspects such as very long cycle is also important. The sodium-cooled 4S SMR simplified-plant design contributes to waste reduction during operation and decommissioning. Some of the FR-SMR designs also progressively apply the radiation-equivalent approximation (in relation to natural occurring raw materials background doses) in its radioactive waste disposal studies. This requires the development of fuel that can accommodate the recycling of the minor actinides.

Molten Salt-fuelled and -cooled SMRs

In all the (thermal spectrum) MSRs included in the booklet, the gaseous fission products are actively removed and stored (to decay). Several designs propose the removal of the fuel salt (after cooling) to a central facility for reprocessing or conditioning and disposal. The residual fuel waste will be transported to geological repositories when the plant is in the decommissioning stage. Actinides such as U/Pu/Th/MA separated at the off-site reprocessing facility are recycled to MSR and the FPs and salt are stored to cool at the disposal facility. In this way (in theory) the lifetime of the waste can be reduced to the few hundred years to decay (excluding the few very long-lived fission products).

Several MSR designers are looking at the opportunity to mine the fuel salt for valuable radioisotopes that can be sold for medical or industrial applications. A design, called SSR-W300 burns most of the common nuclear spent fuel wastes available in the world today, thus potentially significantly contributing to its reduction, if it can be deployed on large scale.

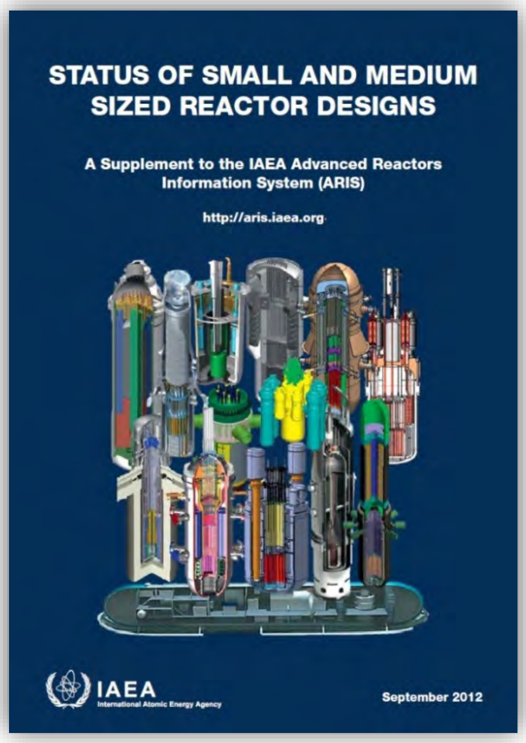
Micro SMRs

In the case of microreactors, radioactive waste management approaches mostly follow that of the reactor family and coolant-type. For example, the Fully Ceramic Micro-encapsulated (FCM™) fuel from the MMR design may be treated similar to the HTGRs coated particle fuels. Research and development on the radwaste management of some unique fuel forms should be further pursued.

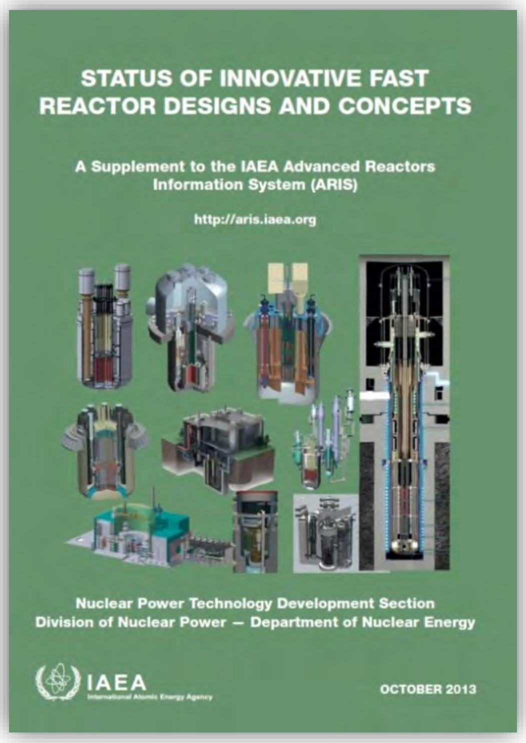
Table VII-1 Waste Management and Disposal Plan adopted by SMR Designs

Waste Management and Disposal Plan Categories	SMR designs by type of coolants and technology characteristics				
	Water-cooled Reactors	**HTGRs**	**Liquid-metal cooled and Fast Reactors**	**Molten Salt Reactors**	**Microreactors**
Volume Reduction and Conditioning	Volume reduction of all waste types are the common principle.	Coated particle separation from graphite will reduce volume by up to a factor of 100.	- Simplified designs contribute to waste reduction. - Development of fuel to include recycled minor actinides to reduce spent fuel lifetime.	- Actinides such as U/Pu/Th/MA separated at the off-site reprocessing facility to be recycled. - Fission products and salt are stored to decay or disposed of.	Small volumes relative to rest of nuclear programme for nuclear power nations.
Waste Processing	For land-based WR-SMRs: similar to that of operating advanced WCRs, using the available technology solution.	- Low and medium level waste from plant operation will be conditioned by different process technologies. - Possible graphite recycling, C_{14} separation process	The required processes have been studied by advanced countries as part of there (large) fast reactor programmes.	Gaseous fission products are removed during operation and stored onsite to decay.	All radioactive waste generated from operation will be transferred to the designated waste area, to be categorized and packaged for removal from the NPP site.
Storage Approach, Spent Fuel Pool Cooling Mechanism	- Approach are very similar to that of the current fleet of large LWRs - In general, designers have plan to safely store, handle and dispose of all the spent fuel including the on-site storage.	- With higher thermal efficiencies the radiotoxicity and decay heat will be lessened by 50% for HTGRs as compared to LWRs. - Dry storage with natural convection after short material active cooling. - Facilities for long-term storage of spent fuel and solid radwaste are in the NPP complex	Spent fuel will be stored until reprocessing and fuel cycle closure becomes economically viable.	- Development of special fuel cask or fuel is cooled within the reactor (tank or pot). - KP-FHR with TRISO pebble fuel, the waste is packaged in multi-purpose canisters for dry interim storage and subsequent off-site transportation for direct geologic disposal or recycling.	Typically, microreactors have life-time core loading and therefore no storage in the plant.
Spent fuel take back; **Market potential**	- For marine-based WR-SMRs deployed in host countries, spent fuel will be taken back to the vendor's country, e.g. Russian Federation. - For embarking countries, spent fuel processing can be done in the reactor supplier's country.	- Spent fuel take back is to date not considered in HTGRs - With TRISO fuels, the core has low power density. If affordable, accident tolerant fuel can be used.	Spent fuel take back is to date not considered in LMFR-SMRs.	For some MSR designs, if deployed in host countries, spent fuel and reactor modules can be taken back to the vendor's country,	Most designers propose fuel to be handled by central facility.

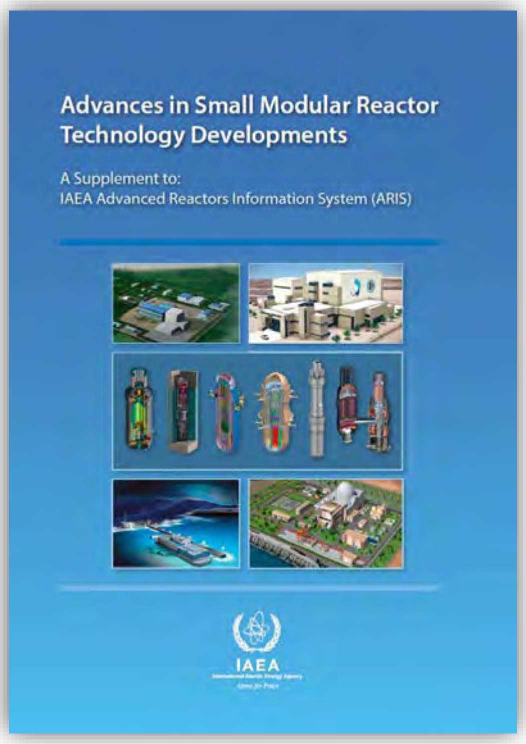
ANNEX VIII
Bibliography



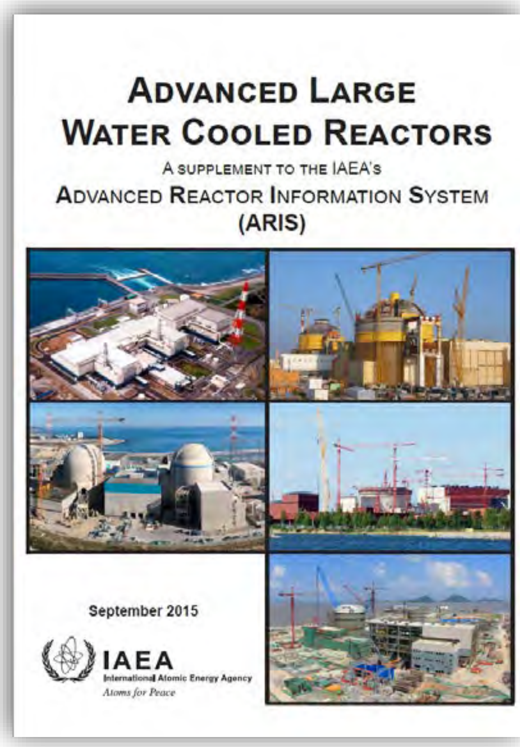
- Contained status, design description and main features of 32 selected SMR designs;
- Sorted by type/coolant: iPWR, PHWR, GCR, and LMFR;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), FBNR (Brazil), CNP-300 (China), Flexblue (France), IMR (Japan), SMART (Republic of Korea), ABV-6M (Russian 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**

- Contained status, design description and main features of 56 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 September 2018**

ANNEX IX
Acronyms

AC	Alternating Current
ADS	Automatic Depressurization System
AGR	Advanced Gas-cooled Reactor
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
ARIS	Advanced Reactor Information System
ASEC	Air Heat Sink for Emergency Cooldown
ASIV	Amphora-Shaped Inner Vessel (in LFR design)
ASME	American Society of Mechanical Engineers
ATS	Automation System
AST	Auxiliary Standby Transformer
ATWS	Anticipated Transient Without Scram
BCR	Back-up Control Room
BDBA	Beyond Design Basis Accident
BOL	Beginning of Life
BOP	Balance of Plant
BPV	ASME Boiler and Pressure Vessel code
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium (Canadian Pressurized Heavy-water Reactor)
CBR	Core Breeding Ratio
CCF	Common Cause Failure
CCS	Containment Cooling System
CCWS	Component Cooling Water System
CEDM	Control Element Drive Mechanism
CNPP	Cogeneration Nuclear Power Plant
CRDM	Control Rod Drive Mechanism
CES	Containment Enclosure Structure
CHE	Containment Hydrogen Control and Filtration Exhaust System (in ACPR50S design)
CHR	Containment Heat Removal System
CI	Conventional Island (Turbine-Generator Building)
CIS	Containment Isolation System
CMT	Core Makeup Tank
CNPP	Cogeneration Nuclear Power Plant
CNSC	Canadian Nuclear Safety Commission
CNV	Cylindrical Containment Vessel (in NuScale design)
CPRSS	Containment Pressure and Radioactive Suppression System (in SMART design)
CPS	Control and Protection System
CRA	Control Rod Assemblies
CRDM	Control Rod Drive Mechanism
CS	Containment Structure
CSG	Compact Steam Generator
CSS	Control Safety System
CSS	Control and Support Safety System
CST	Coolant Storage Tank
CSTS	Condensate Storage and Transfer System
CTS	Chemical Technological Sector
CV	Containment Vessel
CVCS	Chemical and Volume Control System
CWS	Chilled Water System
DAS	Diverse Actuation System
DBA	Design Basis Accident

DBC	Design Basis Condition
DBE	Design Basis Earthquake
DC	Direct Current
DCA	Design Certification Application (a licensing term in the United States)
DCIS	Distributive Control and Information System
DCS	Distributed Control System
DHRS	Decay Heat Removal System
DID	Defence in Depth
DLOFC	Depressurized Loss of Forced Cooling
DRACS	Direct Reactor Auxiliary Cooling System
DU	Depleted Uranium
DVI	Direct Vessel Injection
D/G	Diesel Generator
EAB	Exclusion Area Boundary
EBI	Emergency Boron Injection
EBT	Emergency Boration Tank
ECCS	Emergency Core Cooling System
ECDS	Emergency Cooling Down System
ECT	Emergency Cooldown Tank
EDG	Emergency Diesel Generator
EFPD	Effective Full Power Day
EHRS	Emergency Heat Removal System
ENTSO	European Network of Transmission System Operators for Electricity
EOL	End of Life
EPZ	Emergency Planning Zone
ESF	Engineered Safety Feature
ESWS	Essential Service Water System
ETS	Energy Transport System
FA	Fuel Assembly
FBR	Fast Breeder Reactor
FCD	First Concrete Date
FDT	Fuel salt Drain Tank
FE	Fuel Element
FEED	Front-End Engineering Design
FGCS	Fission Gas Collection System
FHR	Fluoride-salt cooled High Temperature Reactor
FHS	Fuel Handling System
FMCRD	Fine Motion Control Rod Drive (in BWR)
FNPP	Floating Nuclear Power Plant
FOAK	First of a Kind
FPGA	Field Programmable Gate Arrays
FPHE	Formed Plate Heat Exchanger
FPU	Floating Power Unit
FRPS	First Reactor Protection System (in CAREM)
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
FSS	Free Surface Separation
FSS	First Shutdown System (in CAREM)
GCB	Generator Circuit Breaker
GDA	Generic Design Assessment
GDCS	Gravity Driven Cooling System
GDWP	Gravity Driven Water Pool
GFR	Gas-cooled Fast Reactor
GTG	Gas Turbine Generator
GV	Guard Vessel

HALEU	High-Assay Low Enriched Uranium
HE	Heat Exchanger
HEU	High Enriched Uranium
HFE	Human Factors Engineering
HIPS	Highly Integrated Protection System
HGDPV	Hot Gas Duct Pressure Vessel
HLMC	Heavy Liquid Metal-Cooled
HHTS	Hybrid Heat Transport System (in IMR design)
HPB	Helium Pressure Boundary
HPCF	High Pressure Core Flooder (in BWR)
HRSG	Heat Recovery Steam Generator System
HTF	Heat Transfer Fluid
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
HTS	Heat Transport System (in CANDU and PHWR)
HVDC	High Voltage Direct Current
HX	Heat Exchanger
IC	Isolation Condenser
ICRDM	Internally Driven Control Rods Drive Mechanism (in IRIS design)
IHX	Intermediate Heat Exchanger
IPIT	Intermediate Pressure Injection Tanks
IRWST	In-Containment Reactor Water Storage Tank
I&C	Instrumentation and Control
IHP	Integrated Head Package
IRACS	Intermediate Reactor Auxiliary Cooling System
IST	Integrated System Test
IVR	In-Vessel Retention
LBB	Leak Before Break
LBE	Lead-Bismuth Eutectic coolant
LC	Lead Coolant
LCOE	Levelized Cost of Electricity
LEU	Low Enriched Uranium
LFR	Lead-cooled Fast Reactor
LFTR	Liquid-Fluoride Thorium Reactor
LLSF	Low Level Safety Functions (in CAREM)
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPFL	Low Pressure Core Flooder (in BWR)
LWR	Light Water Reactor
MA	Minor Actinides
MCR	Main Control Room
MCSFR	Molten Chloride Salt Fast Reactor
MHT	Main Heat Transport
MOX	Mixed Uranium-Plutonium Oxide fuel
MCP	Main Coolant Pump
MSA	Moisture Separator Reheater
MSFR	Molten Salt Fast Reactor
MSR	Moisture Separator and Reheater
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
MSK	Medvedev-Sponheuer-Karnik scale (a macroseismic intensity scale used in Russia)
MSSV	Main Steam Safety Valve
MWS	Makeup Water System
MWS	Metal and Water Shielding
MW(e)	Mega Watt electric

MW(t)	Mega Watt thermal
NDHP	Nuclear District Heating Plant
NFC	Nuclear Fuel Cycle
NGCC	Natural Gas Combined Cycle
NHP	Nuclear Heating Plant
NHSS	Nuclear Heat Steam Supply System
NI	Nuclear Island (Reactor Building)
NPP	Nuclear Power Plant
NPS	Nuclear Power System
NSSS	Nuclear Steam Supply System
NTEP	Nuclear Thermoelectric Plant
OBE	Operating Basis Earthquake
OCP	Outside Containment Pool
OPEX	Operating Expenses
OTSG	Once-Through Steam Generators
OTTO	Once Through Then Out
O&M	Operation and Maintenance
PAR	Passive Autocatalytic Re-Combiners
PAS	Passive Containment Air Cooling System
PBMM	Pebbled Bed Micro Model (in PBMR design)
PC	Primary Containment
PCCS	Passive Containment Cooling System
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
PFB	Passive Feed and Bleed System
PGA	Peak Ground Acceleration
PHTS	Primary Heat Transport System
PHWR	Pressurized Heavy Water Reactor
PHX	Primary Heat Exchanger
PLC	Programmable Logic Controller
PLOFC	Pressurized Loss of Forced Cooling
PLS	Plant Control System
PMG	Plant Main Generator
PMS	Protection and safety Monitoring System
PORV	Power-Operated Relieve Valve
PRHRS	Passive Residual Heat Removal System
PSAR	Preliminary Safety Analysis Report
PSGC	Passive Steam Generator Cooler
PSIS	Passive Safety Injection System
PSWS	Plant Service Water System
PWR	Pressurized Water Reactor
RIA	Reactivity Insertion Accident
RCAB	Reactor Containment and Auxiliary Building
RCCS	Reactor Cavity Cooling System
RCI	Reactor Coolant Inventory and Purification 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 (in ACP100)
RFA	Robust Fuel Assembly

RHRS	Residual Heat Removal System
RP	Reactor Plant
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTNSS	Regulatory Treatment of Non-Safety System
RV	Reactor Vessel
RVACS	Reactor Vessel Auxiliary Cooling System
RW	Radioactive Waste
RWST	Reactor Water Storage Tank
sCO_2	supercritical CO_2
SAS	Small Absorber Sphere
SBO	Station Black-Out
SDS	Shut Down System (in CANDU and PHWR)
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator
SGPV	Steam Generator Pressure Vessel
SIS	Safety Injection System
SIT	Safety Injection Tanks
SLCS	Standby Liquid Control System
SMR	Small Modular Reactor
SNF	Spent Nuclear Fuel
SSC	Systems, Structures and Components
SSLC	Safety System Logic and Control
SRPS	Second Reactor Protection System (in CAREM)
SSE	Safe Shutdown Earthquake
SSG	IAEA Specific Safety Guide
STSG	Spiral-Tube Steam Generator
TC	Turbo Compressor
T/G	Turbine/Generator
TCU	Thermal Conversion Unit
TEG	Thermoelectric Generator
TES	Thermal Energy Storage
TEU	Thermoelectric Unit
TGP	Turbine Generator Package
TM	Turbo Machine
TRISO	Triple Coated Isotropic
TRL	Technology Readiness Level
UAT	Unit Auxiliary Transformer
UCO	Uranium Oxy Carbide
UHS	Ultimate Heat Sink
UPS	Uninterrupted Power Supply (System)
VCS	Vessel Cooling System
VDR	Vendor Design Review (a licensing term in Canada)
V&V	Verification and Validation
WATSS	Waste to Stable Salt
WDS	Waste Disposal System
WPu	Weapon-Grade Plutonium
WWER	Water-cooled Water-Moderated Power Reactor (Russian PWR)
YSZ	Yttria Stabilized Zirconia pellets

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