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

SMALL MODULAR REACTOR TECHNOLOGY CATALOGUE

2024 Edition

A Supplement to the non-serial publication:
Small Modular Reactors: Advances in Developments 2024

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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 nuclear operating 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 and many other stakeholders such as future operators, potential industrial users, industries in the nuclear supply chain and academics keep expressing their interest in information about advanced SMR designs and concepts, as well as current development trends. The IAEA Division of Nuclear Power supports Member States interested in SMRs by offering a methodology to model energy systems with innovative nuclear technologies, and assess their sustainability, helping them develop the necessary nuclear infrastructure for their deployment and providing engineering support, including supply chain consideration. The activities on SMRs are further supported by specific activities on advanced reactor technology development including fast and high temperature gas cooled reactors, as well as their non-electric application.

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

Booklets (or catalogues) on the status of SMR technology developments have been published biannually since 2012 with the objective 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 microreactors with thermal power typically up to 30MWth. The content on the specific SMRs is provided by the responsible institute or organization and is reproduced, with permission, in this booklet.

This SMR catalogue completes the IAEA Advanced Reactor Information System (ARIS), which can be accessed at http://aris.iaea.org. The ARIS database has been upgraded in September 2024 to be searchable and incorporate comparison tools. The database will be dynamically updated with contribution from SMR technology developers in the near future.

This publication was developed by Nuclear Power Technology Development Section, Division of Nuclear Power of the IAEA Department of Nuclear Energy in cooperation with SMR technology developers from its Member States. The IAEA officer responsible for this publication was B. Lepouzé 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 the needs of the Member States are issued.

The world will need to harness all low-carbon sources of energy in order to meet the 2015 Paris Agreement goal of limiting the rise in global temperatures to well below 2°C 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 baseload power.

In this context, there is an increasing interest in small modular reactors (SMRs): Their small size makes them suitable for smaller grids and easier to finance. Their modular design also holds the promises of a shorter construction span. The IAEA has been supporting this interest by publishing biannually an SMR booklet since 2012, addressing issues such as the non-electric applications of SMRs, their economic challenge, the fuel cycle approach for SMRs, the waste management associated with their operation and the prospect of their decommissioning, not to mention the listing of the latest development of the different designs.

This year, in the framework of the International Conference on SMR and their application organized in Vienna in October 2024, the IAEA published a non-serial publication (*Small Modular Reactors and their Applications: Advances in SMR Developments 2024*) focusing on these issues. The 2024 edition of the booklet, titled *Small Modular Reactor Technology Catalogue*, can be considered as a supplement to the abovementioned non-serial publication and will solely focus on providing a brief technical description of the key SMR designs under different stage of development and deployment.

As illustrated below, the listed designs are grouped under 4 different technology lines (water-cooled reactors, High temperature gas-cooled reactors, liquid metal cooled fast neutron spectrum reactors and molten salt reactors) with 2 additional categories (floating nuclear power plants and microreactors) which can rely on the abovementioned technology lines.

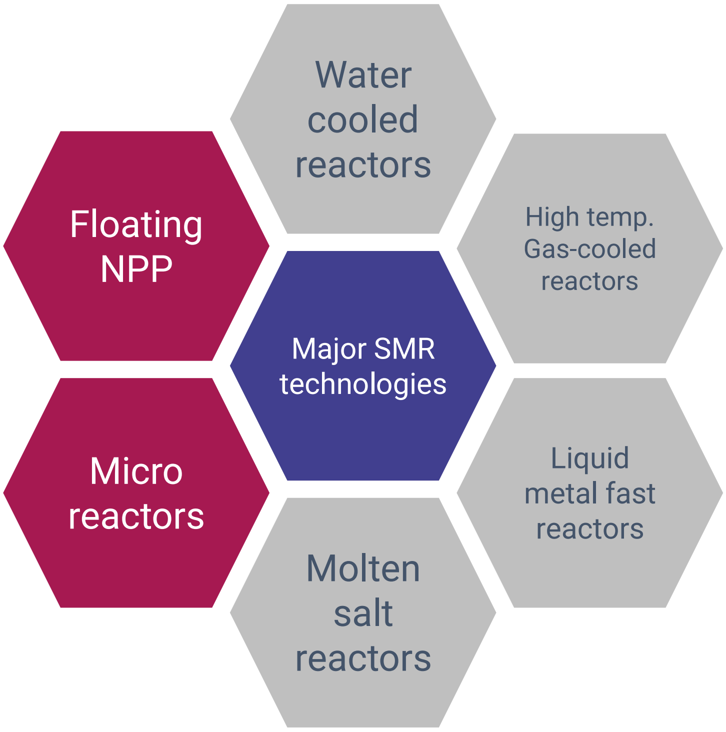

Figure 1: Major technology lines of the SMR technologies

Since 2012, the SMR booklet has been listing an increasing number of designs, with the latest edition featuring eighty-three designs. Although close to a hundred designs could have been listed in the 2024 edition, only active designs with demonstrated sustained development were selected. Even among these active designs, not all of them are expected to develop into real commercial products, as some designs are developed as proofs of concept or study material.

This publication comprises of six (6) parts as follows:

Part One: Land-based water-cooled SMRs. This part presents the key 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 fourteen (14) water-cooled SMR designs from 10 Member States described in this catalogue that comprises integral-PWRs, compact-PWRs, loop-PWRs, BWRs and pool-type reactors for district heating.

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 catalogue presents six (6) marine based water-cooled SMRs, some of them have been deployed as nuclear icebreaker ships.

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. Fourteen (14) HTGR-type SMRs are described in this catalogue.

Part Four: Fast Neutron Spectrum SMRs. This part presents ten (10) 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.

Part Five: Molten Salt SMRs. This part highlights eleven (11) SMR designs from molten salt fuelled and cooled advanced reactor technology (MSRs). 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.

Part Six: Micro-sized SMRs. This catalogue 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). Microreactors may 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. Thirteen (13) microreactor designs are included and discussed in this catalogue.

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

The table below summarizes the designs listed in this publication:

Table 1: Design and Status of SMRs included in this catalogue

Design	Output MW(e)	Type	Designers	Country	Status
PART 1: WATER COOLED SMALL MODULAR REACTORS (LAND BASED)					
ACP100	125	PWR	CNNC	China	Under construction
AP300	330	PWR	Westinghouse Electric Company, LLC	United States of America	Basic design
BWRX-300	300	BWR	GE-Hitachi Nuclear Energy and Hitachi GE Nuclear Energy	United States of America, Japan	Detailed design
CAREM	30	PWR	CNEA	Argentina	Under construction

HAPPY200	200 MW(t)	PWR	SPIC	China	Detailed Design
i-SMR	170	PWR	KHNP & KAERI	Republic of Korea	Conceptual Design
NuScale Power Module	77	PWR	NuScale Power Inc.	United States of America	Detailed design
NUWARD	170	PWR	EDF, CEA, TA, Naval Group	France	Basic Design
PWR-20	20	PWR	Last Energy	United States of America	Detailed design
RITM-200N	55	PWR	JSC “Afrikantov OKBM”	Russian Federation	Detailed design
SMART	107	PWR	KAERI and K.A.CARE	Republic of Korea, and Saudi Arabia	Detailed design
STAR	10	PTWR	Star Energy	Switzerland	Basic design
Rolls-Royce SMR	470	PWR	Rolls-Royce	United Kingdom	Detailed design
SMR-300	320	PWR	Holtec International	United States of America	Conceptual Design
PART 2: WATER COOLED SMALL MODULAR REACTORS (MARINE BASED)					
ABV-6E	9	Floating PWR	JSC Afrikantov OKBM	Russian Federation	Detailed design
ACP100S	125	Floating PWR	CNNC	China	Basic design
BANDI	60	Floating PWR	KEPCO E&C	Republic of Korea	Conceptual design
KLT-40S	2 × 35	Floating PWR	JSC Afrikantov OKBM	Russian Federation	In Operation
RITM-200M	50	Floating PWR	JSC Afrikantov OKBM	Russian Federation	Detailed design
VBER-300	325	Floating NPP	JSC Afrikantov OKBM	Russian Federation	Detailed design
PART 3: HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS					
EM2	265	HTGR	General Atomics	United States of America	Conceptual design
FMR	50	HTGR	General Atomics	United States of America	Conceptual design
GTHTR300	300	HTGR	JAEA	Japan	Basic design
GT-MHR	288	HTGR	JSC Afrikantov OKBM	Russian Federation	Basic design
HTGR-POLA	11.5	HTGR	NCBJ	Poland	Basic design
HTMR-100	35	HTGR	STL Nuclear	South Africa	Basic design
HTR-10	2.5	HTGR	INET, Tsinghua University	China	Operational
HTR-50S	17.2	HTGR	JAEA	Japan	Conceptual design
HTR-PM	210	HTGR	INET, Tsinghua University	China	In operation
HTTR	30 (t)	HTGR	JAEA	Japan	In operation
MHR-100	87	HTGR	JSC Afrikantov OKBM	Russian Federation	Conceptual design
MHR-T	205.5	HTGR	JSC Afrikantov OKBM	Russian Federation	Conceptual design
PeLUIt-40	10	HTGR	BRIN	Indonesia	Conceptual design
Xe-100	82.5	HTGR	X-Energy LLC	United States of America	Basic Design
PART 4: FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS					
4S	10	LMFR	Toshiba Corporation	Japan	Detailed design
ARC-100	100	Sodium-cooled	ARC Nuclear Canada, Inc.	Canada	Conceptual design
Blue Capsule*	50	Sodium-cooled	Blue capsule technology	France	Conceptual design
BREST-OD-300	300	Lead-cooled	NIKIET	Russian Federation	Under construction
HEXANA	150	Sodium-cooled	Hexana	France	Conceptual design
LFR-AS-200	200	Lead-cooled	NewCleo	Italy/France	Conceptual design
OTRERA 300	110	Sodium-cooled	Otrera Energy	France	Conceptual design
SEALER-55	55	Lead-cooled	BlyKalla	Sweden	Conceptual design
SVBR-100	100	Lead-Bismuth	JSC AKME Engineering	Russian Federation	Detailed Design
Natrium	345	Sodium-cooled	Terrapower	United States of America	Conceptual Design

PART 5: MOLTEN SALT SMALL MODULAR REACTORS					
CA Waste Burner	100 MW(t)	MSR	Copenhagen Atomics	Denmark	Detailed design
CMSR	110	MSR	Seaborg Technology	Denmark	Conceptual design
Flex Reactor	24	MSR	Moltex Energy	United Kingdom	Basic design
FUJI	200	MSR	International Thorium Molten-Salt Forum: ITMSF	Japan	Conceptual design
IMSR 400	195	MSR	Terrestrial Energy Inc.	Canada	Detailed design
KP-FHR	140	Modular Pebble-bed HTR Salt Cooled Reactor	KAIROS Power, LLC.	United States of America	Under construction
Stable Salt Reactor - Wasteburner	300	MSR	Moltex Energy	United Kingdom / Canada	Conceptual Design
Stellarium	110	MSR	Stellaria Energy	France	Conceptual design
ThorCon	250	MSR	ThorCon International	Indonesia / United States	Conceptual design
Thorizon	100	MSR	Thorizon BV	Netherlands / France	Conceptual design
XAMR	40	MSR	NAAREA	France	Conceptual design
PART 6: MICROREACTORS					
AMR	3	HTGR	Power Cell Micro Reactor Pty	South Africa	Conceptual design
Aurora	1.5	FR	OKLO, Inc.	United States of America	Basic design
ELENA	68 kW(e)	PWR	National Research Centre “Kurchatov Institute”	Russian Federation	Conceptual design
Energy Well	8	MSR	Centrum výzkumu Řež	Czech Republic	Conceptual design
eVinci	5	Heat Pipe	Westinghouse Electric Company, LLC.	United States of America	Under Development
HOLOS-QUAD	10	HTGR	HolosGen	United States of America	Detailed design
HOLOS-MONO	10	HTGR	HolosGen	United States of America	Detailed design
Jimmy	10 MW(t)	HTGR	Jimmy Energy SAS	France	Detailed design
MMR	15	HTGR	Ultra Safe Nuclear Corporation	United States of America	Basic design
MoveluX	3	Heat Pipe	Toshiba Corporation	Japan	Conceptual design
Pylon D1	1MW(t)	HTGR	Ultra Safe Nuclear Corporation	United States of America	Conceptual design
SHELF	10	iPWR	NIKIET	Russian Federation	Basic design
Unitherm	6.6	PWR	NIKIET	Russian Federation	Conceptual design

* Blue capsule design is metal-cooled but is a thermal spectrum reactor.

Detailed design descriptions of these designs can be found in the IAEA Advanced Reactor Information System (ARIS, http://aris.iaea.org).

WATER COOLED
SMALL MODULAR REACTORS
(LAND BASED)

ACP100 (CNNC,China)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNNC (NPIC), 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 / 17 × 17 square pitch
Number of fuel assemblies in the core	57
Fuel enrichment (%)	< 4.95
Refuelling cycle (months)	24
Core discharge burnup (GWd/ton)	< 52
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)	0.3 g
Fuel cycle requirements/approach	Temporarily stored in spent fuel pools
Distinguishing features	Integrated reactor with tube-in-tube once through steam generator, nuclear island underground
Design status	Construction in progress

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. Design Philosophy

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

4. Main Design Features

(a) 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.

(b) Reactor Core

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

(c) 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).

(d) 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.

(e) 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.

(f) 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.

(g) 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.

(h) Primary pumps

The ACP100 uses canned-motor pumps as reactor coolant pumps which are directly mounted on the RPV nozzle. The shaft of the impeller and rotor of the canned-motor pump is contained in the pressure boundary, eliminating the seal LOCA of reactor coolant pump.

5. Safety Features

(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) Spent Fuel Cooling Safety Approach / System

The In-Refuelling Water Storage Tank (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. 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.

(e) 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.

6. Plant Safety and Operational Performances

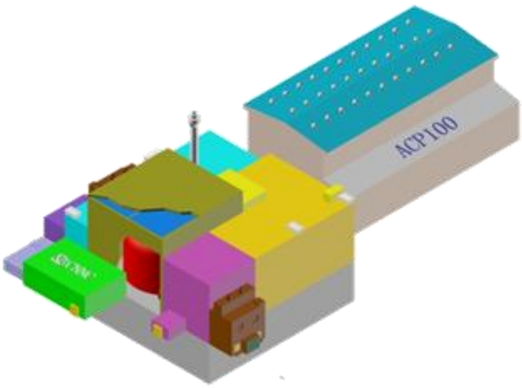
Nuclear safety is always the first priority. The ultimate goal of nuclear safety is to establish and maintain an effective defence that can effectively protect people, community, and environment from radioactive disaster. To be specific, the design and operation of ACP100 ensures that radiation dose 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.

7. Instrumentation and Control System

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.

8. 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.



Plant layout arrangement

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.

9. Testing Conducted for Design Verification and Validation

Seven test research of verification tests have been finished, such as Control rod drive line cold and hot test, Control rod drive line anti-earthquake test, Internals vibration test research, Fuel assembly critical heat flux test research, Passive emergency core cooling system integration test, CMT and passive residual heat removal system test research, Passive containment heat removal testing.

10. Design and Licensing Status

The ACP100 preliminary safety assessment report (PSAR) is approved by National Nuclear Safety Authority and detailed engineering design is in progress. Changjiang nuclear power site, Hainan, China, was chosen to build the first of a kind (FOAK) ACP100 demonstration project. FCD in July 2021. Construction period of FOAK 55 months, target commercial operation in 2026.

Site construction drawing in 2024

11. Fuel Cycle Approach

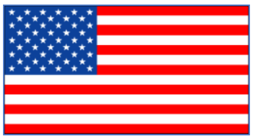
Spent fuel processing is similar to other nuclear power plants. It is temporarily stored in spent fuel pools. Waste management approach and disposal plan is similar to other nuclear power plants.

12. Waste Management and Disposal Plan

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

13. 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, Site Preparation started.
2020	Apply for authorize to Changjiang nuclear power site, Hainan, China.
2021	FCD on Changjiang nuclear power site, Hainan, China.
2022	Star nuclear island installation.
2023	The core equipment (RPV and OTSG) hoisted into position
2024	The outer dome of containing vessel hoisted into position, begin cold test.
2026	Target commercial operation.



AP-300 (Westinghouse Electric Company LLC, United States of America)

AP300 SMR site rendering

AP300 SMR applications

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse, United States of America
Reactor type	PWR SMR
Coolant/moderator	Water
Thermal/electrical capacity, MW(t)/MW(e)	990/330
Primary circulation	Forced with 2 Reactor Coolant Pumps
NSSS Operating Pressure (primary/secondary), MPa	15.4/7.35
Core Inlet/Outlet Coolant Temperature (°C)	302/325
Fuel type/assembly array	UO2 fuel/ 17x17 square lattice
Number of fuel assemblies in the core	121
Fuel enrichment (%)	Less than 5 wt. % U-235 in U
Core Discharge Burnup (GWd/ton)	N/S
Refuelling Cycle (months)	36 (Up to 48)
Reactivity control	Rods, soluble boron
Approach to safety systems	Passive (as in the AP1000)
Design life (years)	80
Plant footprint (m^2)	8300 m^2 (Power block buildings, bounding value)
RPV height/diameter (m)	11/4
RPV weight (metric ton)	275
Seismic Design (SSE)	design basis safe shutdown earthquake 0.3g pga
Distinguishing features	
Design status	Basic design

1. Introduction

Westinghouse AP300 SMR is a 330 MWe class (990 MWth core thermal power), Generation III+, one-loop light water Pressurized Water Reactor (PWR) based on the licensed AP1000 plant technology which has demonstrated industry leading reliability. It is the only SMR utilizing deployed, operating, and advanced reactor technology. The plant design includes passive safety systems and extensive plant simplifications to enhance the safety, construction, operation, and maintenance of the plant. Safety systems use natural driving forces such as pressure, gravity, natural circulation, and convection. Active components such as pumps, fans, or diesel generators are not required, and the plant is designed to function without safety-grade support systems such as AC power, component cooling water system, service water system, or HVAC.

2. Target Application

Benefiting from the experience gained on Westinghouse AP1000 plant projects, Westinghouse offers the highest level of delivery certainty for the AP300 SMR projects. These benefits include technology maturity, licensing confidence, established design and modularization, proven construction and commissioning approaches, resolution of first-of-a kind (FOAK) challenges, and incorporation of lessons learned. All of which provide a high-level of certainty in our team's ability to achieve the overall project schedule in a safe and quality-focused manner to deliver a plant that meets all operational parameters.

3. Design Philosophy

The plant has a well-defined design basis backed by extensive engineering analyses and testing performed as part of the AP1000 plant reactor development. High-level design characteristics of the plant are:

- Leveraged AP1000 reactor design and licensing experience to achieve deployment by early 2030's;
- Maximized use of demonstrated solutions to reduce cost and risk with no plant prototype needed since passive plant systems and power generating system components are based on proven Westinghouse technology;
- Westinghouse 17x17 RFA fuel product that has demonstrated excellent performance in its extensive operating experience in the United States and in more than 50 plants worldwide;
- Proven Westinghouse-supplied fuel and core design solutions capable of extended operating cycle up to 4 years;
- Based on licensed & globally operating AP1000 technology;
- Simplified plant design and world class outage performance resulting in short refueling outages;
- Availability, minus planned outages, is expected to be in excess of 97%;
- Enveloping site parameters are defined to ensure the standard plant design is suitable for a wide range of site locations.

In the extremely unlikely event of a core melt, in-vessel retention of core debris provides high confidence that containment failure and radioactive release to the environment will not occur due to ex-vessel severe accident phenomena.

4. Main Design Features

(a) Nuclear Steam Supply System

The NSSS for the AP300 SMR is a Westinghouse-designed pressurized water reactor. The AP300 SMR is designed around a conventional 1 loop, 1 steam generator primary system configuration with 1 hot leg, 2 reactor coolant pumps (RCP) directly mounted to the steam generator lower head, and two cold legs. The AP300 SMR design uses the same, proven, state-of-the-art RCPs as in the AP1000 plant. The system also includes a pressurizer, interconnecting piping, and the valves and instrumentation necessary for operational control and the actuation of safeguards. The reactor coolant system piping includes the same connection and shapes of the AP1000 design.

(b) Reactor Core

The AP300 SMR core design consists of 121 fuel assemblies, laid out in a typical PWR core configuration. The active fuel stack height is 12 feet, consistent with the majority of today's operating PWRs. The core thermal power rating is 990 MWth resulting in a relatively low core power density that supports extended operating cycle lengths, up to 4 years, with efficient fuel management solutions based on current fuel and licensing framework.

(c) Reactivity Control

The AP300 SMR has 45 Control Rod Drive Mechanisms and incorporates both reduced-worth ("grey") and heavy-worth ("black") Rod Cluster Control Assemblies. In addition to providing rapid shutdown capability, these banks are utilized for implementing the mechanical shim (MSHIM) operational strategy. The MSHIM operation and control strategy, adopted from and proven in the AP1000 PWR, greatly simplifies reactor operations by automatically compensating reactivity changes from fuel depletion by movement of selected control banks, thereby minimizing changes to the coolant soluble boron concentration from multiple times a day in standard plants operation to only every few weeks in the AP300. Additionally, the MSHIM strategy allows to perform load following manoeuvres without changes to the soluble boron concentration during the manoeuvre, which extends load following capabilities and reduces the burden on the Chemical and Volume Control System and related effluents.

5. Safety Features

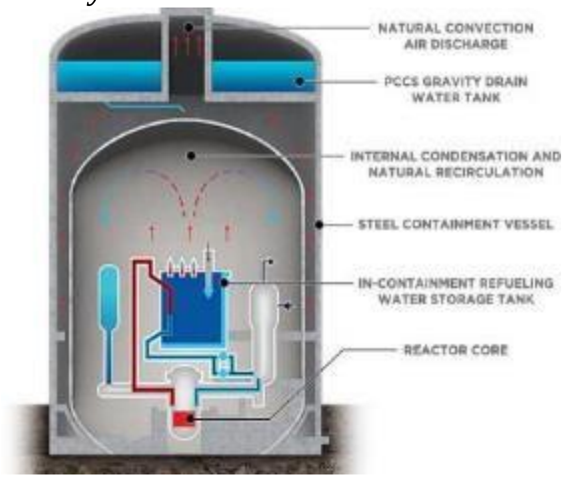
(a) Engineered Safety System Approach and Configuration

The Westinghouse AP300 SMR plant design includes passive safety systems and extensive plant simplifications to enhance the safety, construction, operation, and maintenance of the plant. Safety systems use natural driving forces such as pressure, gravity, natural circulation, and convection. Safety systems do not use active components such as pumps, fans, or diesel generators, and are designed to function without safety-grade support systems such as AC power, component cooling water system, service water system, or HVAC. The number and complexity of operator actions required to control the safety systems are minimized; the

approach is to eliminate the need for operator action rather than automate it. The goal is to ensure the plant safe state to be reached without the need for operator action for all the design basis events.
The AP300 SMR passive safety systems do not require the large network of safety support systems found in current generation nuclear power plants. The safety systems are designed to operate for at least 72 hours without AC power. When AC power is available, the plant passive systems can be supplemented with simple, active defense-in-depth systems and equipment. While important to the safe normal operation of the plant, the active systems are not necessary for the safe shutdown of the reactor following a design basis event.

(b) Containment and Spent Fuel Cooling Safety Approach / System

Similar to the AP1000 plant design, the AP300 SMR has a Passive Core Cooling System (PXS) and Passive Containment Cooling System (PCS). These passive safety-related systems use natural driving forces such as pressure, gravity, natural circulation, and convection and are designed to operate without the use of active equipment such as pumps and AC power sources. Like the AP1000 plant, these passive features do require a one-time alignment of valves upon actuation of the specific components from the plant protection system, when setpoints on key process parameters are reached. Notably, in the AP300 SMR the fuel storage area is located with fuel racks located in the lowest area of the In-Containment Refueling Water Storage Tank (IRWST). As a result, the AP300 SMR does not have a spent fuel pool located outside of containment.



6. Plant Safety and Operational Performances

Westinghouse together with the world’s leading civilian nuclear power plant vendors, have adopted a common set of principles that reflect global best practices in connection with the export of nuclear power plants. Known as the Nuclear Power Plant Exporters’ (NPPE) Principles of Conduct, the principles articulate the nuclear power plant industry’s shared high standards in the areas of safety, security, environmental protection and spent fuel management, compensation in the unlikely event of nuclear-related damage, non-proliferation, and ethics. This type of voluntary code of conduct has never existed in the nuclear industry. Westinghouse is committed to the peaceful use of nuclear energy and the promotion of proliferation-resistant designs and taking IAEA safeguards requirements into account in the reactor design phase. The AP300 SMR is designed to the highest criteria to provide assurance for the protection of the health and safety of the public against radiological sabotage.
The key design objectives of the AP300 Physical Security System (SES) include the following:

- Providing access control capability for the plant protected and vital areas by limiting entry to authorized personnel, vehicles, and material only.
- Providing video surveillance and assessment capability.
- Providing a centralized command-and-control computer network and communications for the plant security response force.
- Providing or utilizing communications capability for the plant security response force, operations, and emergency response personnel.
- Providing deterrence and delay of attempted unauthorized entry into the plant protected and vital areas.
- Providing hardened defensive positions for protection of the plant security response force from the elements of the design basis threat.
- Ensuring adequate lighting as necessary for essential elements of the surveillance and assessment security systems and to allow effective response by the plant security response force.
- Providing a power supply as necessary for subsystems of the security system, including but not limited to: intrusion detection, access control, and video assessment/surveillance security systems.
- Providing detection of attempted unauthorized entry to the plant protected and vital areas.

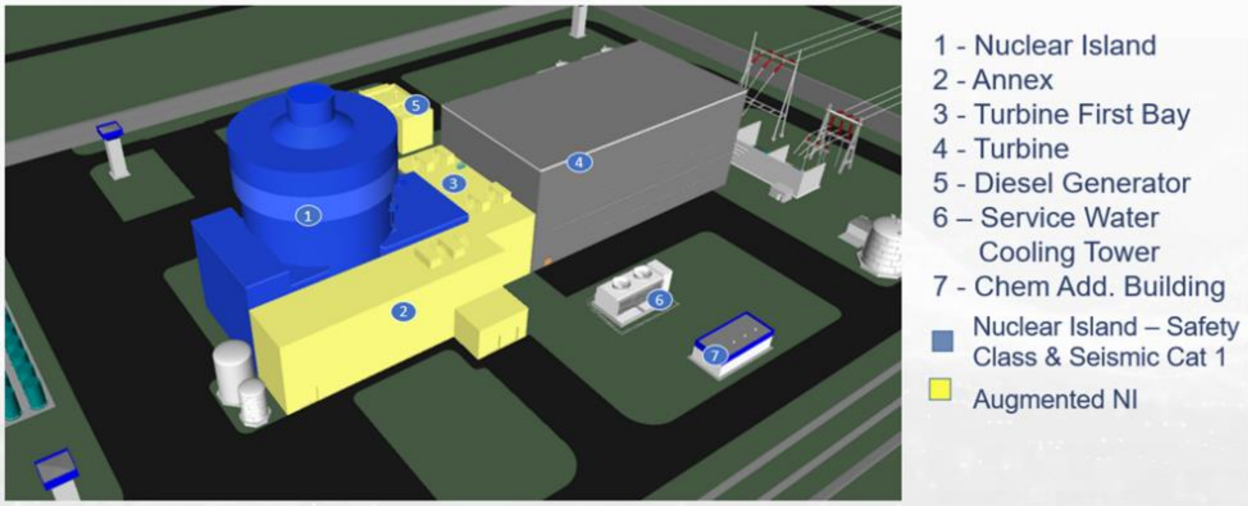
7. Instrumentation and Control Systems

The 4-division AP300 SMR safety system utilizes the Common Q™ platform. The Common Q™ platform is the standard Westinghouse safety system platform previously approved by the NRC with over 20 years of operating experience. The plant uses a proven Distributed Control System (DCIS) platform with improvements in supporting server technology using a high availability virtualized system that reduces the volume of independent servers and increases reliability. The Diverse Actuation System (DAS) also applies the non-safety platform from the DCIS using separate network segments and different applications. The non-safety platform is diverse from the Common Q safety platform, and DAS uses independent and dedicated sensors diverse from the Common Q safety system.

8. Plant Layout Arrangement

The AP300 SMR overall plant arrangement utilizes building configurations and structural designs to minimize the building volumes and quantities of bulk materials (concrete, structural steel, rebar) consistent with safety, operational, maintenance, and structural needs. The plant arrangement provides separation between safety-related and non-safety-related systems to preclude adverse interaction between safety-related and non-safety-related equipment. Separation between redundant safety-related equipment and systems provides confidence that the safety design functions can be performed. Minimizing the overall plant footprint with focus on the Nuclear Island (NI), the AP300 SMR locates all the safety-related structures, systems and components in the Nuclear Island. Structures located off the Nuclear Island are neither safety-related nor seismic Category I.
The AP300 SMR buildings are grouped as shown below:

- Nuclear Island
  - Containment and Shield Building
  - Auxiliary Building
- Nuclear Island Supporting Buildings and Structures
  - Annex Building
  - Diesel Generator Building
- Turbine Island
  - Turbine Building First Bay
  - Turbine Building



9. Testing Conducted for Design Verification and Validation

The AP300 SMR passive safety features rely on extensive testing performed as part of the AP1000 reactor development program which ensures the safety features will perform as designed and analyzed. Performance has been further demonstrated through extensive full-scale testing as part of commissioning the lead AP1000 reactors. Tests were also conducted during the AP600 and AP1000 Design Program to provide input for plant design and to demonstrate the feasibility of unique design features. Tests for the AP600/AP1000 design certification and design program were devised to provide input for the final safety analyses, to verify the safety analysis models and to provide data for final design and verification of plant components. No fewer than fifteen different test facilities for design basis accident testing were used to demonstrate passive core and containment cooling and to collect data for code validation for the Westinghouse passive plant technology. Further, the U.S. NRC has conducted extensive, independent testing of the technology in five test facilities confirming design basis accident results obtained by Westinghouse as well as a significant number of beyond design basis accidents. Finally, to support the startup and operation of the first wave of AP1000 plants, several tests were performed of the passive safety systems. The results of these tests agree well with the scale model testing performed during the AP1000 plant development effort, providing further evidence to support the AP300 licensing.

10. Design and Licensing Status

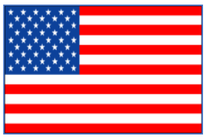
AP300 is using a proven technology based on AP1000. So far (in 2024) six AP1000 units are operational resulting in over 24 reactor years of demonstrated operational excellence and their design, construction, testing data, licensing, and operational experience feed into the AP300 program. The AP300 plant design complies with all international design and licensing regulations including US, EUR and URD. Lessons learned from the AP1000 licensing process with the US NRC and NNSA, Generic Design Assessment in the UK and Pre-project design review by CNSC in Canada will be incorporated into the AP300 SMR design approval process and will enable a more streamlined, informed approach to worldwide compliance. The AP300 SMR design is currently progressing through the design development stages with conceptual design completed in 2023, and the basic design phase in progress and culminating in 2025 with the submittal of the Design Certification Documentation to the regulators. Regulator certification is expected in 2027, which will align with the standard plant deployment readiness. Site specific project preparation is expected to commence in 2027 with site specific design & licensing activities, as well as long lead item procurement. The AP300 SMR target is to be ready for construction in 2030 with the Nth of a kind (NOAK) unit construction schedule being 36 months (first nuclear concrete to ready for fuel load).

11. Fuel Cycle Approach

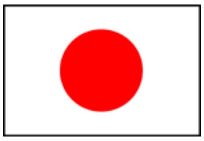
The reference fuel assembly design for the AP300 SMR is based on the Westinghouse 17x17 Robust Fuel Assembly-2 (RFA-2) bearing 264 fuel rods in a 17x17 square array. The center position in the fuel assembly has a guide thimble that is reserved for in-core instrumentation, with guide thimbles in the remaining 24 positions. The guide thimbles are joined to the top and bottom nozzles of the fuel assembly and provide the supporting structure for the fuel grids. A number of proven design features have been incorporated in the AP300 fuel assembly design that improve fuel performance, enhance fuel reliability, enable better fuel cycle economics and provide additional margin.

12. Development Milestones

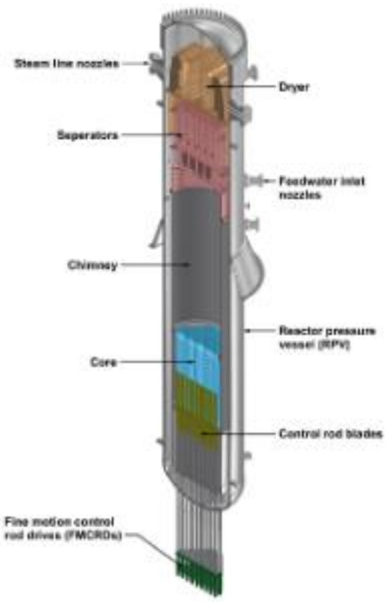
2023	Conceptual Design Complete
2024	Start of Basic Design; Pre-licensing NRC interactions with White Paper and Topical Report Submittals; Submit UK Generic Design Assessment (GDA) Entry Application to DESNZ
2025	Design Certification Application (DCA) Submittal to US NRC; Initiate and Conclude GDA Step 1 (Initiation) with UK ONR
2026 - 2027	Detailed Design Completion, Initiate and Conclude GDA Step 2 (Fundamental Assessment) with UK ONR, NRC Issues Final Safety Evaluation for Design Certification & Standard Plant Ready to Deploy
2027 – 2030	Target Long Lead Item Procurement, Site Specific Design & Licensing, Ready for Construction
2033 - 2034	Target AP300 Commercial Operation



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



Reactor Pressure Vessel



Control Room

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	GE Hitachi Nuclear Energy, U.S.A. and Hitachi-GE Nuclear Energy, Japan
Reactor type	BWR
Coolant/moderator	Light water/light water
Thermal/electrical capacity, MW(t)/MW(e)	870 MW(t) / 300 MW(e)
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	7.17 MPa(abs) (primary)
Core Inlet/Outlet Coolant Temperature (°C)	270 / 288
Fuel type/assembly array	UO_2 / Square lattice
Number of fuel assemblies in the core	240
Fuel enrichment (%)	3.8% (average) / 4.95% (maximum)
Core Discharge Burnup (GWd/ton)	~ 55
Refuelling Cycle (months)	12 – 24
Reactivity control	Control rods
Approach to safety systems	Defense-in-Depth approach with passive safety systems
Design life (years)	60
Plant footprint (m^2)	11,550
RPV height/diameter (m)	27 / 4
RPV weight (metric ton)	650
Seismic Design (SSE)	0.3
Distinguishing features	Simplified BWR design, natural circulation, passive safety features, cost-competitive with natural gas plants
Design status	Detailed Design

1. Introduction

The GEH Hitachi Nuclear Energy BWRX-300 is a Small Modular Reactor (SMR) that uses natural circulation and passive safety systems. It produces 870 MWth and, in electricity production only configuration, approximately 300 MWe net. The BWRX-300 is the tenth generation Boiling Water Reactors (BWR) and continues the multi-generational journey of simplification in design and operations and maintenance. It leverages several features and many of the licensing aspects of the 1,520 MWe Economic Simplified Boiling Water Reactor (ESBWR). The BWRX-300 is intended for flexible and cost-competitive energy generation.

2. Target Application

Target applications include base load electricity generation, load following electrical generation, hydrogen production, synthetic fuel production, district heating and other process heat applications.

3. Design Philosophy

The BWRX-300 utilizes natural circulation and passive safety systems that rely on natural phenomena rather than active mechanical components. It leverages the U.S. NRC-licensed ESBWR design and aims to minimize construction, operation, maintenance, and decommissioning costs. The safety strategy is based on the International Atomic Energy Agency’s Defence-in-Depth methodology, incorporating five defence lines to ensure robust safety margins and resilience against various initiating events.

4. Main Design Features

(a) Nuclear Steam Supply System

BWRX-300 Nuclear Steam Supply System (NSSS) employs a Reactor Pressure Vessel (RPV) with a tall chimney to facilitate efficient natural circulation for core cooling, eliminating the need for recirculation pumps. The water entrained in the steam generated in the core is removed by steam separators and the steam dryer before existing the RPV and flowing to the steam turbine. Integral Isolation Vales that are directly flanged to RPV are deployed for all flow paths greater than 25 mm. Decay heat is removal is performed by the Isolation Condenser System (ICS) for at least seven days for all design basis accidents.

(b) Reactor Core

BWRX-300 reactor core is a vertical cylinder containing low-enriched uranium GNF2 fuel assemblies that are designed for efficient thermal-hydraulic stability. GNF2 fuel is commercially available with more than 26,000 bundles delivered to the BWR fleet. Control rods are inserted and withdrawn from the bottom of the core for reactivity management. The flow in the core is driven by natural circulation. The core design provides flexibility and minimal fuel cycle disruptions with cycle lengths of 12 to 24 months depending on customer needs.

(c) Reactivity Control

BWRX-300 reactivity control is managed with Fine Motion Control Rod Drives (FMCRDs) that enable precise control during normal operation and rapid hydraulic insertion during emergencies. Control rods containing neutron-absorbing materials and burnable absorbers in the fuel rods help regulate core reactivity. The system ensures core stability and safety by allowing fine-tuned adjustments and quick responses to potential reactivity changes.

(d) Reactor Pressure Vessel and Internals

BWRX-300's RPV assembly consists of a pressure vessel with removable head, internal components, and appurtenances. Key internals include the chimney, steam separators, and steam dryer, all scaled to optimize thermal output and natural circulation. The RPV and internals utilized decades of best practices and lessons learned from the RPV fleet.

(e) Reactor Coolant System and Steam Generator

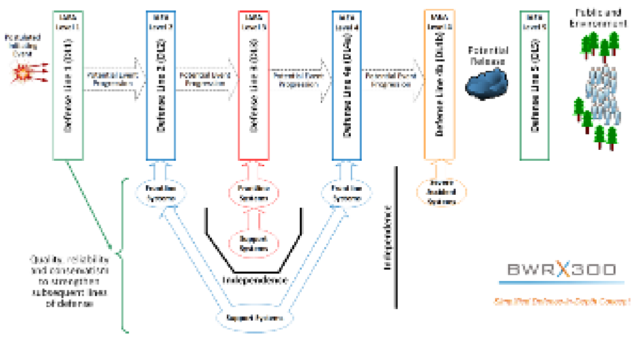
The BWRX-300 Reactor Coolant System (RCS) is natural circulation driven and provides cooling of the reactor core in all operational states and all postulated off normal conditions. The BWRX-300 leverages natural circulation modelling and operational information from the ESBWR and the Dodewaard BWR in the Netherlands. The relatively tall RPV drives natural circulation driving forces to produce abundant core coolant flow. The BWRX-300 does not need steam generators.

(f) Pressuriser

BWRX-300 does not use a separate pressurizer, as it a direct cycle steam system.. Instead, pressure control is managed through the design of the RPV with a large water and steam inventory and modulation of the main steam turbine control valves and turbine bypass valves.

(g) Primary pumps

Relies on natural circulation within the RCS to drive coolant flow through the reactor core. It eliminates the need for active mechanical pumps, utilizing gravity and thermal buoyancy to circulate coolant effectively.



Defence-in-Depth Concept

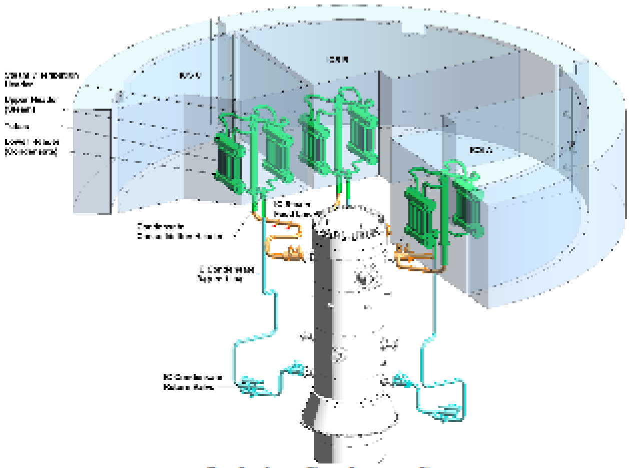
5. Safety Features

(a) Engineered Safety System Approach and Configuration

BWRX-300 adheres to IAEA safety guidelines and Defence-in-Depth approach, integrating multiple safety layers to prevent and mitigate accidents. It features advanced passive systems like Isolation Condenser Systems and a large Reactor Pressure Vessel to manage heat and maintain core cooling with minimal operator input. Safety goals include a core damage frequency of less than 1E-7 per year and a large release frequency of less than 1E-8 per year for the reference site, supported by robust emergency planning and risk assessments.

(b) Safety Approach and Configuration to Manage DBC

BWRX-300 employs a defence-in-depth strategy to manage Design Basis Conditions (DBC), utilizing multiple safety layers to protect against potential failures. It includes physical barriers, such as the RPV and containment structures, along with redundant safety systems for detecting and mitigating accidents. It also incorporates advanced passive systems like Isolation Condensers for heat removal, ensuring reliable safety margins and minimal reliance on operator actions during accidents.



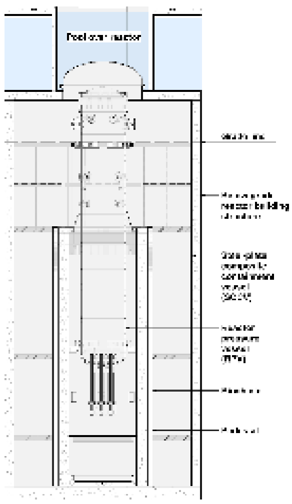
Isolation Condenser System

(c) Safety Approach and Configuration to Manage DEC

BWRX-300 also manages Design Extension Conditions (DEC) through a Defence-in-depth approach, which incorporates multiple layers of protection to prevent or mitigate severe accidents. It integrates active systems, like the Primary Protection System (PPS) for detecting and mitigating accidents, and Severe Accident Management Guidelines (SAMGs) to manage extreme events and ensure effective response while maintaining containment integrity.

(d) Containment System

BWRX-300's Primary Containment System (PCS) is a vertical cylindrical structure, about 17 meters in diameter and 40 meters high, which encloses the Reactor Pressure Vessel (RPV) and associated systems. Designed with a pressure of 414 kPaG and a peak transient temperature of 166°C, the PCS ensures radioactive fission products, steam, and water are contained, featuring a design leakage rate of ≤0.35 wt.%/day. The Containment Cooling System (CCS) and Passive Containment Cooling System (PCCS) manage temperature and pressure, with the CCS using active cooling and the PCCS relying on natural circulation and condensation for heat removal during accidents or loss of power.



Dry Containment

(e) Spent Fuel Cooling Safety Approach / System

BWRX-300 uses a multi-layered approach to ensure the safe cooling of spent fuel. The Spent Fuel Pool cooling is primarily managed by a combination of active and passive systems. The active system includes a dedicated Spent Fuel Pool Cooling System with redundant heat exchangers and pumps to ensure continuous heat removal.

6. Plant Safety and Operational Performances

BWRX-300 prioritizes safety with a defence-in-depth approach, integrating active and passive systems like Isolation Condensers and integral RPV isolation valves. It ensures robust safety margins with a Core Damage Frequency (CDF) of <1E-7 and Large Release Frequency (LRF) of <1E-8 per year for the reference site.

7. Instrumentation and Control Systems

BWRX-300 features a 300 MWe class steam turbine and generator operating at 3000/3600 rpm at 21 kV. Its Emergency Power System ensures 72-hour battery-backed DC and UPS AC power with provisions for portable generators. The DCIS is divided into three independent Safety-Class 1 (SC1) trains for protection and separate and independent triple modular redundant SC2, SC3 and SCN3 systems for plant controls. The control room complies with IEC and IEEE standards.

8. Plant Layout Arrangement

BWRX-300 plant layout includes a central Reactor Building (RB) with a cylindrical shaft, surrounded by the Control Building (CB), Turbine Building (TB), and Radwaste Building (RW). The RB, a Seismic Category 1 structure, houses the RPV and primary containment. The CB contains control systems, the TB houses turbine and generator systems, and the RW contains the radioactive waste systems.

BWRX-300 Plant Layout

9. Testing Conducted for Design Verification and Validation

BWRX-300 undergoes design verification and validation through prototype testing, component testing, and system integration testing. Comprehensive performance tests simulate operational and emergency scenarios to ensure system reliability and compliance with regulatory standards.

10. Design and Licensing Status

BWRX-300 is in various stages of licensing and design validation worldwide. It has undergone pre-application reviews with the UK ONR, Canada's CNSC, and the U.S. NRC. In Canada, the License to Construct was filed for the lead BWRX-300 in October 2022 with issuance expected in early 2025. In the U.S., five Licensing Topical Reports have been approved with two more in the final stages of approval. The firs U.S. Construction Permit Application is planned for submittal in early 2025. The BWRX-300 is also under Generic Design Assessment in the UK, with Step 2 completion scheduled in 2025. The Polish Regulator has issued Decision in Principle for six sites to deploy BWRX-300s. Early pre-application discussions are ongoing in several other countries.

11. Fuel Cycle Approach

BWRX-300 utilizes a standard BWR fuel cycle approach with a focus on efficiency and safety. It employs low-enriched uranium (LEU) fuel assemblies. The design supports fuel cycles of 12 to 24 months, enhancing fuel efficiency and waste minimization.

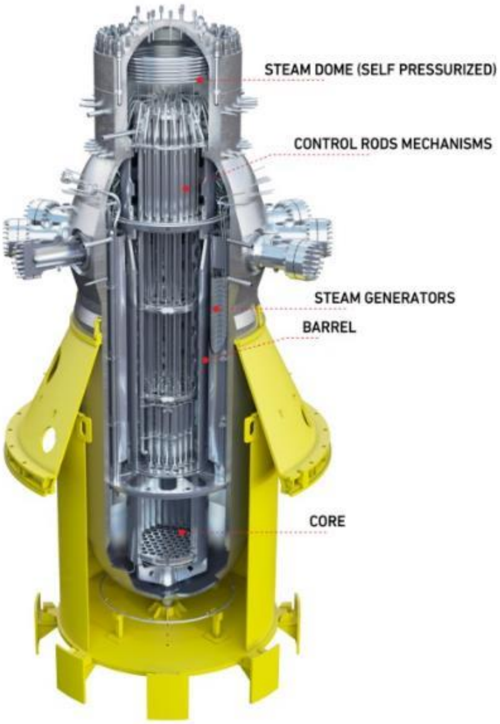
12. Waste Management and Disposal Plan

BWRX-300 is designed for optimal resource use and waste management, aiming to minimize non-renewable resource consumption and reduce radioactive waste generation. It features advanced systems like the Liquid Waste Management System (LWM) and Offgas System (OGS) to minimize environmental releases and ensure low doses to personnel.

13. Development Milestones

2017	Initial Conceptual Design	Complete
2019	Pre-Application review of Licensing Topical Reports (U.S.A)	Complete
2021	Conceptual Design Phase	Complete
2022	Preliminary Design Phase	Complete
2022	License to Construct application submitted (Canada)	Complete
2023	Vendor Design Review (VDR) completed (Canada)	Complete
2024	Generic Design Assessment (U.K.) started	On-Track

CAREM (CNEA, Argentina)



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 (CAREM25)
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	12.25 / 4.7 (CAREM25)
Core Inlet/Outlet Coolant Temperature (ºC)	284 / 326 (CAREM25)
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	61 (CAREM25)
Fuel enrichment (%)	3.1% (CAREM25)
Core Discharge Burnup (GWd/ton)	24 (CAREM25)
Refuelling Cycle (months)	14 (CAREM25)
Reactivity control mechanism	Control rod driving mechanism (CRDM) only
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	36 000 (CAREM25)
RPV height/diameter (m)	11 / 3.2 (CAREM25)
RPV weight (metric ton)	267 (CAREM25)
Seismic Design (SSE)	0.25g (CAREM25)
Fuel Cycle Requirements or Approach	390 full-power days and 50% of core replacement (CAREM25)
Distinguishing features	Core heat removal by natural circulation, pressure suppression containment
Design status	Under construction (CAREM25)

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 level of safety and economic competitiveness. CAREM is an integral PWR type NPP, based on indirect steam cycle with features that simplify the design and support the objective of achieving a higher level of safety.

CAREM25 is the demonstration prototype of CAREM SMR, and 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. 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 steam production and condensation in the vessel, without a separate pressurizer vessel. 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; and Safety systems relying on passive features.

4. Main Design Features

(a) Nuclear Steam Supply System

CAREM is an integral reactor. Its high-energy primary system (core, steam generators, primary coolant and steam dome) is within 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.

(b) Reactor Core

The reactor core of CAREM25 has hexagonal cross section fuel assemblies. There are 61 fuel assemblies with 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.

(c) Reactivity Control

Core reactivity is controlled using Gd_2O_3 as burnable poison in specific fuel rods and movable absorbing elements belonging to the power adjustment and reactivity 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.

(d) Reactor Pressure Vessel and Internals

Reactor Pressure Vessel (RPV) of CAREM25 is 11 meters high and 3.4 meters in diameter, with a variable thickness of 13cm to 20cm. The RPV is made of forged steel with an internal stainless steel liner.

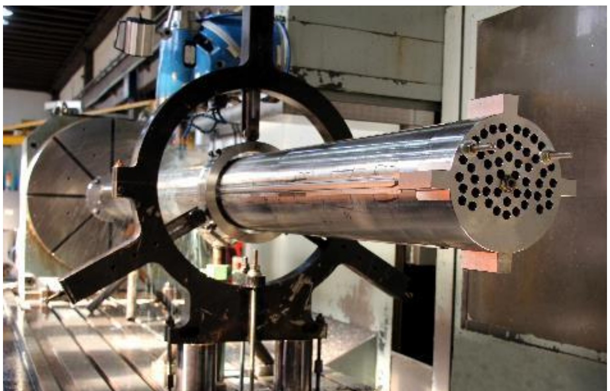
CAREM25 RPV fabrication

(e) Reactor Coolant System

The reactor´s coolant system of CAREM is fully contained in the RPV. Core cooling is achieved using natural circulation phenomenon with no primary coolant pumps. The reactor´s core heats the coolant causing it to flow upwards through the riser. Then it turns downwards to flow into the steam generators, reducing its temperature. Finally, it flows to the downcomer and the core, closing the circuit.

(f) Steam Generator

In CAREM25, twelve (12) 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 4.7MPa. The secondary system circulates upwards within the tubes, while the primary coolant moves in counter-current flow. The steam generators are designed to withstand the primary pressure. 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.



CAREM25 steam generator fabrication

(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.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Defence in Depth (DiD) principle was the basis of this process. Appropriate criteria were defined for DiD internalization in the design taking into account CAREM general design characteristics, to prevent, control and mitigate the postulated events. Western European Nuclear Regulators Association proposal for DiD levels definition was adopted, which allows the consideration of Multiple Failures Events (Level 3B) as part of DiD Level 3 for the prevention of core damage. A strategy was defined for each one of the levels. For Level 2 the CVCS can control a medium loss of coolant event and LOHS. Two stages for level 3A and level 3B are established. For level 3A, the first stage is accomplished by means of passive safety systems, while the second stage is implemented by active systems in order to achieve the final safe state. For Level 3B, which is used in case of failure in Level 3A, the first stage is also accomplished by systems with passive processes while the second stage is composed by simple systems with external water supply. For Level 4 -postulated severe accident mitigation- provisions are considered to guarantee hydrogen control and RPV lower head cooling with the aim of in-vessel corium retention.

CAREM´s safety system consists of two reactor protection systems (RPS), two shutdown systems, passive residual heat removal system (PRHRS), safety and depressurization valves, low pressure injection system and a containment of pressure suppression type. The First Shutdown System (FSS) consists of 9 fast shutdown rods and 16 reactivity adjust and control rods located over the core, which fall by gravity when needed. While the Second Shutdown System consists of a gravity assisted high-pressure injection of borated water from two high pressured tanks, which is activated automatically when failure of the FSS is detected.

The different systems, structures and components (SSCs) that contribute to the reactor´s safety are classified according to the identification of low-level safety functions (LLSF) -derived from the fundamental safety functions- and safety functional groups of SSCs that fulfill those functions.

(b) Decay Heat Removal System

During the grace period of 36 hours, core decay heat removal is assured by one out of two PRHRS in the case of loss of heat sink (LOHS) or Station Black-out (SBO). SBO is considered to be a design basis event. After that period, redundant active systems provide heat removal from the RPV and the containment to the final heat sink. Two redundant diesels provide energy supply for these systems. Despite the low frequency of an SBO longer than 36 hours, provisions for that scenario are supplied by simple systems supported by a fire extinguishing system or self-powered pumps. The PRHRS are heat exchangers formed by parallel horizontal U-tubes (condensers) coupled to common headers.

(c) Emergency Core Cooling System

In case of a loss of coolant event, a Medium Pressure Injection System delivers water into the RPV to keep the core covered during the grace period. This system consists of two redundant borated water accumulators connected to the RPV. When the pressure in the reactor vessel becomes relatively low, the rupture disks, that isolate the accumulator tanks from the RPV will break. After the grace period and in case the loss of the coolant could not be isolated, an active system provides long term water injection into the RPV. Two redundant diesels provide energy supply for these systems.

(d) Spent Fuel Cooling Safety Approach/System

In case of a SBO and during the grace period, longer than the reactor one, the water of the spent fuel pool is the heat sink. After this period, a chain of redundant active systems provides heat removal to the final heat sink. Two redundant diesels provide energy supply for these systems.

(e) Containment System

The cylindrical containment vessel with a pressure suppression pool is a 1.2m thick reinforced concrete external wall with a carbon steel liner and withstands earthquakes of 0.25g. It is designed to withstand the pressure of 0.5MPa. The heat sink is located inside the containment, this provides protection for extreme external events during the grace period. After grace period, redundant active systems with diesel generator support, will provide suppression pool cooling and containment depressurization.

6. Plant Safety and Operational Performances

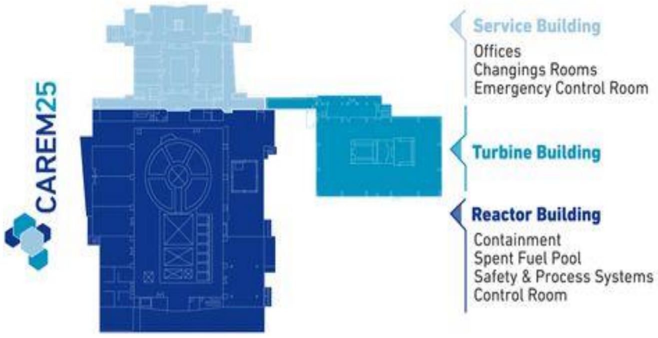
The natural circulation of coolant produces different flow rates in the primary system according to the power generated and removed. Under different power transients, a self-correcting response in the flow rate is obtained.

Due to the self-pressurizing of the RPV, the system keeps the pressure close to the saturation pressure. 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.

7. 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.

8. Plant Layout Arrangement



CAREM plant layout

CAREM25 construction site

9. Testing Conducted for Design Verification and Validation

A full-scale loop has been built in order to test the innovative control rod drive mechanisms. This facility is operating at the same parameters (pressure and temperature) as RPV plant design conditions and is was designed for reactivity adjustment and control rods calibration.

10. Design and Licensing Status

CAREM25 basic design is completed. CAREM25 detail design is 90% progress. CAREM25 was granted a construction license. CNEA is working on the commissioning license request.

11. Fuel Cycle Approach

The fuel cycle can be tailored to customer requirements, as a reference CAREM25 has an open fuel cycle design of 390 full-power days and 50% of core replacement.

12. 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.

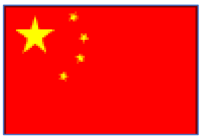
13. Plant Economics

As CAREM25 is a prototype, plant economics is not provided. CNEA is working on economic studies for the commercial module.

14. 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
2006	Argentina Nuclear Reactivation Plan listed the CAREM25 project among priorities of national nuclear development
2009	CNEA submitted its preliminary safety analysis report (PSAR) for CAREM25 to the ARN
2011	Start-up of a high pressure and high temperature loop for testing the innovative hydraulic control rod drive mechanism

2011	Site excavation work beginning
2012	Civil design beginning
2013	Construction license
2014	Civil works started
2024	Electromechanical erection (start)
2028	First criticality



HAPPY200 (SPIC, China)

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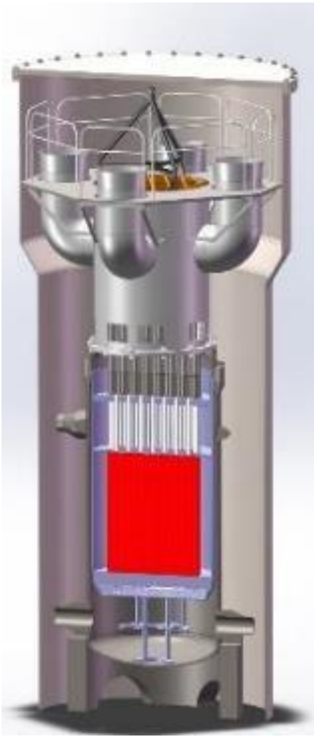
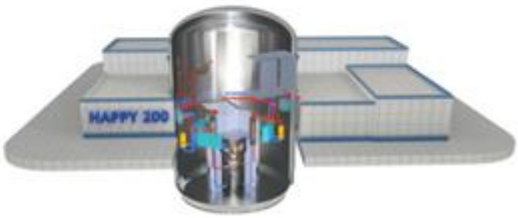




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	State Power Investment Corporation, Ltd. (SPIC), China
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	200 / 0 (thermal power only)
Primary circulation	Forced (2 pumps)
NSSS Operating Pressure (primary/heating), MPa	0.6 / 0.8
Core Inlet/Outlet Coolant Temperature (°C)	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.

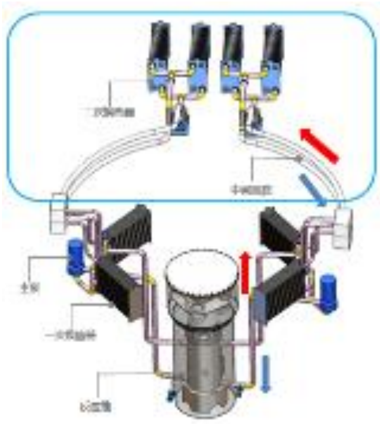
3. Design Philosophy

The HAPPY200 draws on the mature operating experience of light water-cooled reactors, pool reactors and passive nuclear safety technology. Both RCS and ESF system are operated under low pressure and low temperature spectrum, and the design parameters can be fine-tuned to adapt to local civil heating requirement. HAPPY200 adopts proven technology and equipment, such as truncated fuel assembly, plate-type heat exchanger, etc. These equipments have full operational record, high reliability and high maintainability. The plant is to be built half-underground, with reactor core vessel totally submerged in a large underground pool, to elimination of external-event challenge. And 5 FP shielding and barriers are deployed to ensuring radioactivity isolation from terminal users, as well as the surrounding public. 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.

4. Main Design Features

(a) Primary Circuit

During normal operation, the 80℃ inlet water enters into the core from the bottom, and the water is then heated to a temperature of 120℃. 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℃ 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.

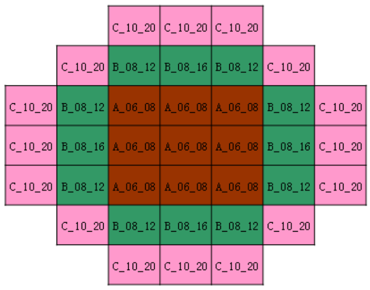


(b) Reactor Core and Fuel

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 ^{235}U 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.

(c) 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.



(d) 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.

(e) 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.

5. Safety Features

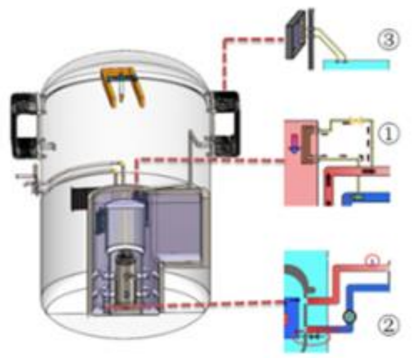
The primary safety objective of HAPPY200 design is practical elimination of core melting and technical cancellation of off-site emergency. To achieve this safety objective, the safety concept of HAPPY200 is based on inherent safety features, the defence in depth principle, the use of passive systems 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.

6. 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.

7. 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.

8. 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 principal 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.

9. Testing Conducted for Design Verification and Validation

The HAPPY200 reactor system is in general designed based on proven technology and method. The spectrum of reactor core and coolant temperature and pressure is decreased consequently both under the normal operation and accident conditions, which resulted in larger thermal margin compared to conventional LWR design criteria, as well as possible cooling and flooding of reactor core accessible to low pressure system, as part of ESF features. To date, 3 important test facilities, one for secondary reactor shutdown system validation testing and one for engineering design validation of PFB and PHR system, and in addition, a large steel containment for validation of PAC system has been built and ready for performing of validation test. Besides, a test for critical heat flux of truncated fuel assembly is also planed for near-term construction.

10. 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

11. 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.

12. 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.

13. Development Milestones

2015	Start market investigation and conceptual design (changes)
2016	Conceptual design
2019	Start pre-project work in north of China
2024	Project suspended for the time being

i-SMR (KHNP and KAERI, Republic of Korea)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	i-SMRDA, KHNP&KAERI Republic of Korea
Reactor type	Integral PWR
Coolant/moderator	Light Water / Light Water
Thermal/electrical capacity, MW(t)/MW(e)	520 / 170
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15.5
Core Inlet/Outlet Coolant Temperature (^{o}C)	286 / 321
Fuel type/assembly array	UO_2 pellet / 17 X 17 square
Number of fuel assemblies in the core	69
Fuel enrichment (%)	< 5.0
Core Discharge Burnup (GWd/ton)	< 62.0
Refuelling Cycle (months)	24
Reactivity control	Control rod, burnable absorber rods, moderator temperature coefficient
Approach to safety systems	Fully passive
Design life (years)	80
Plant footprint (m^2)	NA
RPV height/diameter (m)	29.7 / 5.5
RPV weight (metric ton)	650
Seismic Design (SSE)	0.5g
Distinguishing features	Boron-free operation At least 72 hours for core cooling without any safety grade AC/DC power
Design status	Standard Design in progress

1. Introduction

In order to promote sustainable development and achieve carbon neutrality, the i-SMR is on development in Republic of Korea with enhanced safety and economic efficiency compared to conventional nuclear power plants, offering flexibility for multipurpose use and harmonization with renewable energy. The i-SMR is an innovative small modular PWR producing 520MW thermal power. The i-SMR has an integral Reactor Coolant System (RCS) configuration that eliminates large pipes for connection of the major components. The major primary components such as a core with 69 fuel assemblies, four Reactor Coolant Pumps (RCPs), a helical once-through Steam Generator (SG), and a Pressurizer (PZR) are installed in a single Reactor pressure Vessel (RV) and equipped in a compact steel Containment Vessel (CV). Due to the integral arrangement, the possibility of a Large Break Loss of Coolant Accident (LB-LOCA) is inherently eliminated.

2. Target Application

The i-SMR is a multi-purpose reactor for electricity production, heat supply for industries, seawater desalination and hydrogen production. The i-SMR is four module design, and its total electricity is comparable to substitute the conventional fossil power plant.

3. Design Philosophy

The i-SMR implements design simplification and a elimination of a large break loss-of-coolant accident by integrating the primary coolant system. The i-SMR incorporates the boron-free operation for enhancing safety and economics. The fully passive safety system is adopted without any safety-grade electrical power and operator's action to achieve a high level of safety goals.

4. Main Design Features

(a) Nuclear Steam Supply System

The i-SMR has an integral Reactor Coolant System (RCS) configuration that eliminates large pipes for connection of the major components. The major primary components, such as a core with 69 fuel assemblies, four (4) Reactor Coolant Pumps (RCPs), helical once-through Steam Generator (SG), and Pressurizer (PZR) are installed in a single Reactor pressure Vessel (RV) and equipped in a compact steel Containment Vessel (CV). Due to the integral arrangement, the possibility of a Large Break Loss of Coolant Accident (LBLOCA) is inherently eliminated.

(b) Reactor Core

The reactor core generates heat by a controlled nuclear fission reaction and transfers the generated heat to the reactor coolant. The rated core thermal power is 520 MWth. It consists of 69 fuel assemblies, 49 Control Rod Assemblies (CRAs), and 20 In-Core Instrument (ICI) assemblies. The core is designed for the refueling cycle to be 24 months with a maximum fuel rod burn up of 62,000 MWd/MTU.

(c) Reactivity Control

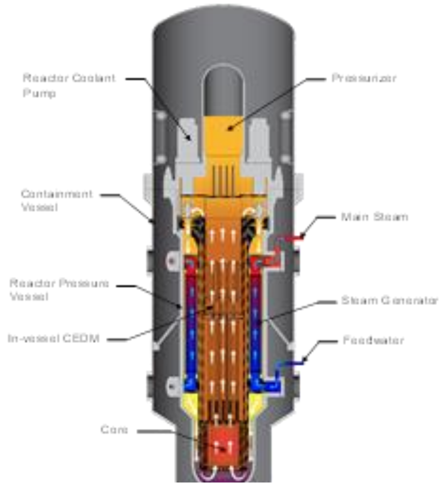
Reactivity control during normal operation is achieved by control rods, burnable absorbers, and moderator temperature. The i-SMR adopts boron-free operation and eliminates significant amount of component and maintenance activities related to boron generation and concentration. The primary reactivity control system is the control rods and burnable absorbers. Burnable poison rods are introduced to give flat radial and axial power profiles, which results in an increased thermal margin of the core. The i-SMR adopts an In-Vessel Control Rod Drive Mechanism (IV-CRDM) which excludes the rod ejection accident. A large number of control rods in i-SMR core assures a relatively high control rod worth. In accordance with the implementation of a boron-free operation, the secondary reactivity control system uses the Moderator Temperature Coefficient (MTC) of a boron-free core with a large negative MTC.

(d) Reactor Pressure Vessel and Internals

The Reactor pressure Vessel (RV) is a cylindrical pressure vessel manufactured vertically, serving as a pressure boundary for RCS along with the PZR. To minimize welds, the RV is primarily produced through forging. The interior of the RV is clad with austenitic stainless steel to enhance corrosion resistance. The internals of RV consist of the Core Support Barrel (CSB), which supports the nuclear fuel assemblies and secures the flow path; a helical SG; the Upper Guide Structure (UGS) installed inside the CSB; and the Internal Barrel Assembly (IBA) designed to support the IV-CRDM and ICI guide tube.

(e) Reactor Coolant System and Steam Generator

The RCS transfers heat generated from the core to the secondary system through the SG and acts as a barrier that prevents the release of reactor coolant and radioactive materials to the public. The RCS and its supporting systems are designed with sufficient core cooling margin for protecting the reactor core from damage during all normal operation and Anticipated Operational Occurrence (AOO). The reactor coolant flows upward through the core, inside CSB assembly including UGS assembly, turns downward through the RCP, SG shell side, and then returns back into the core through the lower plenum. The forced circulation flow of the reactor coolant is formed by four (4) RCPs installed at the upper side of the reactor vessel. By this coolant circulation, the core heat is delivered to the secondary system via the SG.



Flow Path of RCS

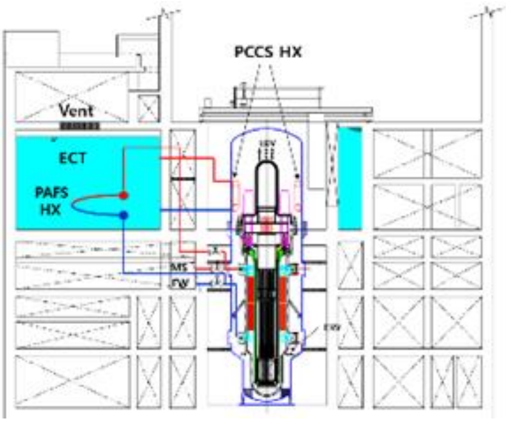
The SG, which is installed inside the RV, is a helical once-through type in which the reactor coolant and secondary coolant flow outside and inside the SG tubes, respectively. The purpose of the SG is to generate superheated steam during normal operation of the reactor and to remove residual heat of the primary system during reactor shutdown or in an upset event. The tubes of the SG are helical-designed to provide the maximum heat transfer area in limited space. The primary coolant flow and secondary coolant flow inside reactor vessel are treated as a single path for the normal operation, respectively. A portion of helical tubes may participate in the heat transfer process during accident condition.

(f) Pressuriser

The pressurizer for i-SMR is located at top-side of the RV, because this configuration has advantages for pressurizing the reactor coolant in a subcooled state and collecting non-condensable gases. The pressurizer is designed as a steam pressurizer where saturated steam and water co-exist. The steam pressurizer has an advantage that the simple control schemes are provided by two-phase phenomena during the transients.
Overpressure protection of the RCS is provided by the PSVs to discharge pressurizer steam during over-pressurization transients. The pressurizer has sufficient steam volume to limit that water level from reaching the valve nozzles during in-surge transients.

(g) Primary pumps

The i-SMR has four (4) RCPs vertically installed at upper shell of the RCH. Each RCP is an integral unit consisting of a canned asynchronous three phase motor. Since a canned motor pump does not require pump seals, its characteristic basically eliminates possibility of the Small Break Loss of Coolant Accident (SBLOCA) associated with a pump seal failure. The RCPs circulate reactor coolant through the SG to exchange heat with secondary feedwater flowing in tube side of the SG. The reactor coolant is cooled as it passes through the SG, and is then returned to the core where it is reheated. The RCP penetrates through the flange-type nozzles of the RCH in vertical direction and inserted into the nozzles formed in the UGS and the CSB.



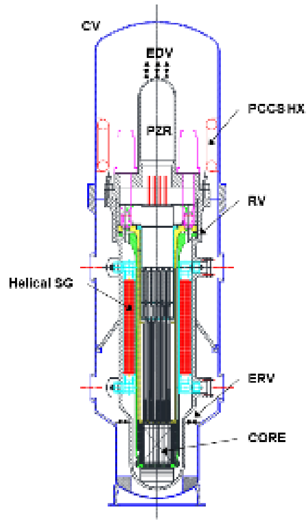
PAFS and PCCS schematics

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The Engineered Safety Features (ESF) of i-SMR, sensible mixture of proven technologies and advanced design features, are designed to function automatically on demand and to cope with Design Basis Accidents (DBAs) and Design Extension Condition without significant fuel degradation (DEC-A) for mitigation. These safety functions to control, mitigate, or terminate such accidents keep radiation exposure levels below the regulatory limits. The main systems of ESFs are as following;

- Passive Auxiliary Feedwater System (PAFS)
- Passive Emergency Core Cooling System (PECCS)
- Passive Containment Cooling System (PCCS)



Schematic Diagram of ECCS

The PAFS removes the RCS heat by natural circulation in emergency situations where normal steam extraction and feedwater supply are unavailable. The PECCS is composed of the reactor vessel, containment vessel, EDVs, Emergency Recirculation Valves (ERVs). Two (2) ERVs and two (2) EDVs provide integrity of the Reactor Coolant Pressure Boundary (RCPB) during normal operation, and open when the pressure difference between the RV and CV reaches the predetermined setpoint which is much lower than the pressure difference during the normal operation. Emergency cooling of the core is initiated by relieving the RV's hot and high-pressure steam and water into the CV via the EDVs. The hot steam and water explosively expand and builds up the CV pressure. The steam is soon condensed by the PCCS heat exchanger and the heat is dissipated mainly to the Emergency Cooling Tank (ECT). The condensed water accumulates in the annulus between RV and CV. Once the pressures of the RV and CV get balanced close enough, the condensed water in the CV annulus is recirculated back to the RV via the ERVs. As such, continuous natural circulation is maintained and thus the continuous core cooling.

(b) Safety Approach and Configuration to Manage DBC

The safety approach for design and operation in i-SMR is based on the defence in-depth philosophy. Multiple barriers such as fuel pellet, cladding, RV, and CV prevent radioactive release to the environment and those barriers are protected by passive safety systems. Multiple, independent, and various systems are designed to remove heat for protection of those barriers. The following five (5) levels of defence are provided to the i-SMR based on IAEA SSR-2/1 and WENRA RHWG report, which is implemented in such a way that when one level fails it makes the subsequent level come into play:

Level 1: Prevention of abnormal operation and failures by means of conservative design and high quality in construction and operation, control of main plant parameters inside defined limits

Level 2: Detect and control of abnormal operation and failures by means of control and limiting systems and other surveillance features

Level 3: Control of accident to limit radiological releases and prevent escalation to core melt conditions by means of:

Level 3a: Control of design basis accidents by reactor protection system, safety systems and accident procedure

Level 3b: Control of complex sequences and prevention of core melt by means of, if required, additional safety measures and accident procedures

Level 4: Control of accident with core melt to limit off-site releases by means of complementary safety features to mitigate core melt and management of accident with core melt (severe accidents)

Level 5: Mitigation of radiological consequences of significant releases of radioactive material by means of off-site emergency response and intervention levels

A set of DBAs is postulated and accommodated by the design of the i-SMR. The consequences of the DBAs fall within the acceptance limits established to protect safety of the public and the plant staff, as well as to provide an appropriate level of plant protection

(c) Safety Approach and Configuration to Manage DEC

In the i-SMR design, severe accidents are addressed as follows:

- For phenomena likely to cause early containment failure, for instance, within 24 hours after accidents, mitigation systems shall be provided or design should address the phenomena although the probability for such accidents its low
- For phenomena which potentially lead to late containment failure if not properly mitigated, the mitigation system or design measures should be considered in conjunction with the probabilistic safety goal and cost for incorporating such features to address the phenomena

This approach is to enhance the effectiveness of investment on safety by avoiding undue over-investment on highly improbable accidents. Also, a realistic assessment is recommended for severe accident analyses.

The severe accident management plan encompasses a range of equipment, including safety and non-safety graded items, available both inside and outside containment, along with the necessary procedures for responding to accidents.

(d) Containment System

The Containment Vessel (CV) of i-SMR is a vertical cylindrical structure surrounding the RV, made of metal. Its primary function is to prevent the release of radioactive materials into the environment during normal operation and accidents. The outer surface of the CV is exposed to the atmosphere, and since there are multiple penetrations for piping and valve connections, isolation valves are installed on the safe end nozzles of the containment vessel to prevent the release of radioactive materials into the environment. By maintaining a vacuum inside, steam released in to the CV during the operation of the ECCS is condensed by the PCCS installed within the CV. The condensed cooling water is then injected into the RV through the ERV once the pressure between the RV and CV is equalized, helping to cool the reactor core.

(e) Spent Fuel Cooling Safety Approach / System

The Spent Fuel Pool Cooling and Clean-up System (SFPCCS) is designed to remove decay heat from the spent fuel assemblies, maintain the temperature of the Spent Fuel Pool (SFP), control water inventory, and maintain the water purity and clarity of the pools. The safety related function of cooling and shielding the fuel in the SFP is performed by the water in the pool.

6. Plant Safety and Operational Performances

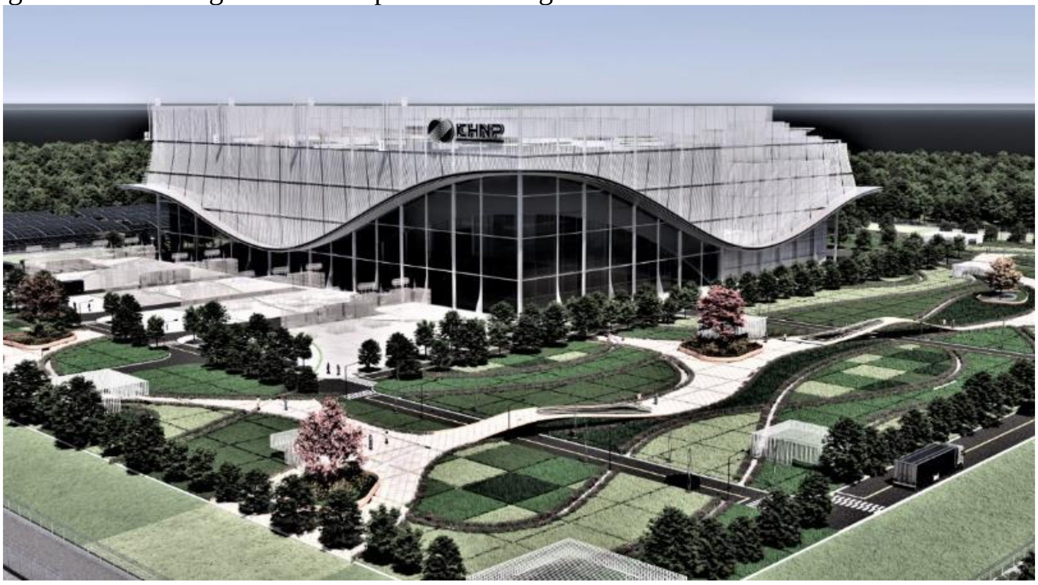
The safety goals of the i-SMR is to achieve a core damage frequency less than 10^{-9} per module-year and a large release frequency less than 10^{-10} per module-year for internal event and full power. The operating experience of KHNP has been applied to the i-SMR with the expected available factor no less than 95%. Load following operation of i-SMR is simpler than that of conventional PWR because boron-free operation of i-SMR with a large reactivity feedback effect minimizes the movement of control banks and boron concentration control is also not necessary. The daily load following performance simulation of the i-SMR core shows that radial peaking factor, 3D peaking factor and the axial offset were satisfied within the design limit.

7. Instrumentation and Control Systems

The Human-Machine Interface System (HMIS) features an advanced control room and the digital control systems. HMIS combined with human factor engineering for ensuring safety by minimizing the likelihood of human error provides the capability to protect, control, and monitor the nuclear power plants. The Integrated Main Control Room (IMCR) is provided with Human-Machine Interface (HMI) devices from which operator action can be taken simultaneously to operate for four integrated reactors safely under all operating conditions (except MCR fire) and maintain it in a safe condition under all operating conditions. The HMIS is developing to satisfy all regulatory requirements such as independence, redundancy, defense-in-depth and diversity and to improve the economics and operability.

8. Plant Layout Arrangement

The i-SMR is designed to accommodate four (4) integrated reactors and containment vessels, structures, systems and components. The power block of the i-SMR consists of the reactor building, the control building, the turbine generator building and the compound building.



i-SMR plant site overview

9. Testing Conducted for Design Verification and Validation

The i-SMR is based on the proven design and extensive development experience obtained from the operation and construction of the conventional PWR. The verification test of key passive safety features have been conducted and the safety validation tests for the i-SMR is ongoing for licensing.

10. Design and Licensing Status

The i-SMR design is developed in conformity with Korean law, codes and standards for nuclear power plants and safety principles.

11. Fuel Cycle Approach

The i-SMR adopts the same open fuel cycle as operating PWRs. The fuel cycle of i-SMR is 24 months.

12. Waste Management and Disposal Plan

The i-SMR has several design solutions to minimize radioactive waste generation. All radioactive waste will be processed with the conventional PWR waste treatment.

13. Development Milestones

2019 – 2020	Conceptual Design Phase	Complete
2021 – 2023	Basic Design Phase	Complete
2024 – 2025	Standard Design Phase (and preparation for pre-licensing)	On track
2026 – 2028	Licensing	Planned

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

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NuScale Power LLC., USA
Reactor type	Integral PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	250 / 77 (gross)
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	13.8/4.3
Core Inlet/Outlet Coolant Temperature (°C)	249/316
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)	50 - 55 with cycle optimization
Refuelling Cycle (months)	Nominal 18
Reactivity control mechanism	Control rod drive, boron
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	Dependent on plant design options
RPV height/diameter (m)	17.7 / 2.7
RPV weight (metric ton)	TBC
Seismic Design (SSE)	0.5g
Fuel cycle requirements / Approach	Nominal Three stage in-out refuelling scheme
Distinguishing features	Unlimited time for core cooling without AC or DC power, water addition, or operator action
Design status	Equipment Manufacturing in Progress

1. Introduction

The NuScale Power Module™ (NPM) is a small, light-water-cooled pressurized-water reactor (PWR). Small Modular Reactor (SMR) plants with NuScale Power Modules™ (NPMs) are scalable and can be built to accommodate a varying number of NPMs to meet customer's energy demands. NuScale standard configurations include the 4-NPM at 308 MW(e), the 6-NPM at 462 MW(e), and the 12-NPM at 924 MW(e). A six-module configuration is the reference plant size for the Standard Plant Design Approval design and licensing activities. Plant configurations can include air-cooled and/or water-cooled condensers. 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. The U.S. Nuclear Regulatory Commission (NRC) has approved the control room conduct of operations for 3 operators controlling up to 12 NuScale reactors and has eliminated the requirement for a Shift Technical Advisor. Significant plant design features include factory fabricated, compact NPMs that include the containment, 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

The NuScale design is a modular reactor for electricity production, with the capability for flexible operations to load follow, and for non-electrical process heat applications, including the cogeneration of heat and electricity.

3. Design Philosophy

The NuScale SMR design approach 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 reactor cooling time after a beyond design basis accident without AC or DC power, operator action, or makeup water. There are no design basis accidents that uncover the core or require operator action. The NPM is designed to operate efficiently at full-power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps.

4. Main Design Features

(a) 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.

(b) Reactor Core

The core configuration for the NPM consists of 37 fuel assemblies and 16 control rod assemblies. The fuel assembly design is an approved, commercially available, 17 x 17 PWR fuel assembly with 24 guide tube locations for control rod fingers and a central instrument tube. The assembly is nominally half the height of standard plant fuel and is supported by five spacer grids. The fuel is UO_2 with Gd_2O_3 as a burnable absorber homogeneously mixed within the fuel for select rod locations. The ^{235}U enrichment is up to the current U.S. manufacturer limit of 4.95 percent enrichment.

(c) 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. The absorber material in these control rods is B_4C and their length is 2 m.

(d) 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 a flange 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 and the reactor vent valves.

(e) Reactor Coolant System and Steam Generator

The reactor coolant system (RCS) provides for the circulation of the primary coolant using natural circulation. Hence, the RCS does not require reactor coolant pumps or an external piping system to generate flow during power operations. 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.

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 that is routed to the turbine-generator.

(f) Pressurizer

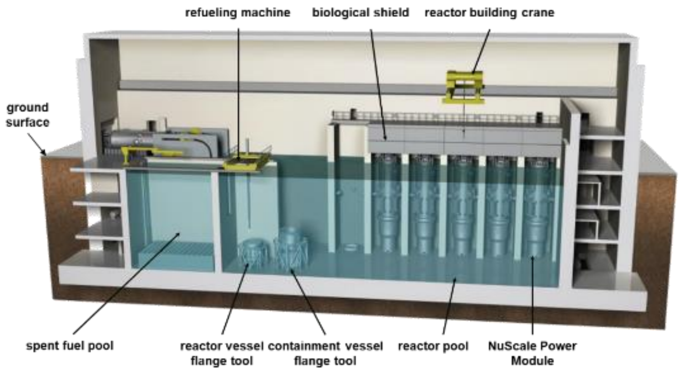
The internal pressurizer provides the primary means for controlling reactor coolant system pressure. It is designed to maintain a constant reactor coolant pressure during operation. Reactor coolant pressure is increased by applying power to a bank of heaters installed above the pressurizer baffle plate.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

5. Safety Features

The NuScale NPM 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. This fully passive safety design is rigorously proven by the NuScale Triple Crown for Nuclear Plant Safety™, which ensures that reactors will safely shut down and self-cool, indefinitely, and do so with no need for operator or computer action, AC or DC power, or the addition of water—a first for light water reactor technology.

(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 SMR power plant with NuScale NPMs

(b) Safety Approach and Configuration to Manage DBE

The decay heat removal system (DHRS) provides secondary side reactor cooling 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. 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) Safety Approach and Configuration to Manage DEC

Emergency core cooling system (ECCS) consists of two independent reactor vent valves (RVVs) and two independent reactor recirculation valves (RRVs). For LOCAs inside containment, the ECCS returns coolant from the CNV to the reactor vessel. This ensures that the core remains covered, and that decay heat is removed. The ECCS also 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 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.

(e) Spent Fuel Cooling Safety Approach

The used fuel pool provides storage for up to 10 years of used fuel storage, plus temporary storage for new fuel assemblies. The pool is connected to the ultimate heat sink, and hence, protected by the reactor building. The pool water volume provides approximately 150 days of passive cooling of the used fuel assemblies following a loss of all electrical power without the need for additional water.

After removal from the reactor core, used fuel assemblies are placed in dedicated used fuel storage racks in the below ground used fuel pool. The used fuel pool is a below-grade, stainless steel lined concrete pool adjacent to the reactor pool. A clean-up system reduces the build-up of contaminants. Within approximately 5 years, the thermal load of the used fuel assemblies is reduced significantly, and can be moved to a secure dry storage area. The plant site layout includes space allocation adequate for the dry storage of all the used fuel for the 60-year life of the plant.

6. Plant Safety and Operational Performances

Each NPM is operated independent of other modules. A module is refuelled by disconnecting it from its operations bay and moving it to a common refuelling area within the shared reactor pool. The module is disassembled into 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.

The nominal plant capacity factor is 95% with a refuelling outage time of 10 days,

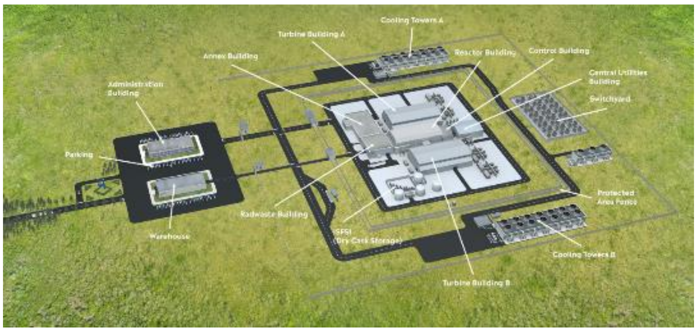
7. Instrumentation and Control System

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 NRC, 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.

8. Plant Layout Arrangement

(a) Reactor Building

NuScale's approach to an SMR plant consists primarily of a reactor building, a control room building, turbine-generator building(s), a radwaste treatment building, wet-cooled condensers (optional air cooling), a switchyard, and a dry-cast storage area for discharged fuel. The reactor building, shown in the Figure, includes up to 12 NPMs, 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 I standards.



(b) Control Building

The main control room is housed in the control building located adjacent to the reactor building. All NPMs are controlled from a single control room. The NRC has approved the control room conduct of operations for 3 operators controlling up to 12 NuScale reactors and has eliminated the requirement for a Shift Technical Advisor. 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

An SMR plant with 12 NPMs has up to 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.

9. Testing Conducted for Design Verification and Validation

NuScale designed, built, and operated a one-third scale prototype testing facility that replicated the NPM and its safety systems including the reactor pool. It provided an electrically-heated core to bring the system up to operating temperature and pressure. This facility was used to test the NPM and gather data for thermal hydraulic codes, safety analysis code, and reactor design validation.

NuScale designed and performed comprehensive testing to validate the operation of the helical coil steam generators, the safety valves, the control rod assembly and drive shafts, fuel and various other systems. NuScale's baseline nuclear energy system has an average Technology Readiness Level (TRL) of 8, meaning that the vast majority of subsystems (107 out of 114) are ready for construction or manufacturing with a TRL of 6 or higher. NuScale testing programs have been audited by the NRC.

10. Design and Licensing Status

In December 2016, NuScale submitted the Design Certification Application (DCA) to the NRC for a 160MWt (50MWe) design. In September 2020, the NRC issued the Standard Design Approval for the NuScale DCA, making it the first ever SMR to receive NRC design approval. The NRC then issued its final rule fully certifying the design effective February 2023, making it the 7th reactor design certification the NRC has ever issued.

In October 2022 U.S. NRC approved NuScale's methodology for determining the appropriate size of the Emergency Planning Zone (EPZ) surrounding the power plant allowing a wide range of potential plant sites to achieve site boundary EPZ.

In December 2022, NuScale submitted a standard design approval (SDA) application for an uprated 6-module, 250MWt (77 MWe) per module plant design. This design is currently under review by the NRC. The review is currently over 50% complete and is expected to be approved in July 2025. In 2023, manufacturing began on the first 6 NPM reactor pressure vessels and steam generator tube bundles.

11. Fuel Cycle Approach

The three-batch refuelling is conducted on a nominal 18-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. Actual batch size, loading pattern, and cycle length will be established by customer-driven optimization requirements.

12. 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.

13. Staffing

An SMR plant with 12 NuScale Power Modules will require a minimum of 3 licensed operators per shift in the control room, and is estimated to require 270 plant employees for normal operation and maintenance. No Shift Technical Advisor is required in the U.S. for this design.

14. Development Milestones

2003	Initial concept developed and integral test facility operational
2007	NuScale Power, LLC was formed
2008	Initiation of NRC Pre-Application
2011	NuScale Power is acquired by Fluor
2012	Twelve-reactor simulated control room was commissioned
2013	U.S. DOE SMR Cooperative Agreement signed
2016	Design certification application submitted to the U.S. NRC for 160 MWt design
2020	Standard Design Approval (SDA) received from the US NRC in September 2020
2022	NPM production began with the production of forging dies for the Upper Reactor Pressure Vessel
2022	U.S. NRC approved NuScale's methodology for determining the appropriate size of the Emergency Planning Zone (EPZ) allowing potential plant sites to achieve site boundary EPZ.
2022	Uprated Standard Design Approval submitted to the NRC to raise nominal power to 250 MWt
2023	NRC issued final rule fully certifying the 160MWt design effective February 2023, making it the 7th reactor design certification the NRC has ever issued.
2023	Fabrication of 6 Reactor Pressure Vessels and Steam Generator Tube Bundles begins
2023	Uprated Standard Design Approval docketed by US NRC for uprated NuScale Power Module with 250 MWt (77 MWe); US NRC provided a 24-month review schedule for approval.
2025	Expected Standard Design Approval for 250 MWt (77 MWe) NPM design from US NRC.

NUWARD™ (EDF Consortium, France)

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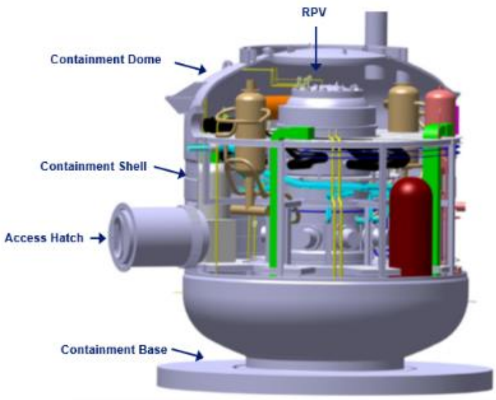
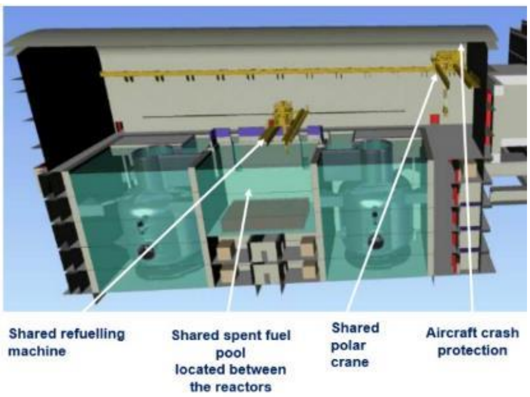




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	EDF, France with major contributions from CEA, Naval Group, Framatome, TechnicAtome, and Tractebel-Engie
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	2 x 540 / 2 x 170
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15 / 4.5
Core Inlet/Outlet Coolant Temperature (^{o}C)	280 / 307
Fuel type/assembly array	UO_2 / 17x17 square pitch arrangement
Number of fuel assemblies in the core	76
Fuel enrichment (%)	<5
Refuelling Cycle (months)	24 (half core)
Reactivity control	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)	15 / 5
RPV weight (metric ton)	310
Seismic Design (SSE)	0.3g
Distinguishing features	Integrated NSSS with pool submerged containment, boron-free in normal operation and in all Design Basis Conditions (DBC), semi-buried nuclear island
Design status	Basic Design

1. Introduction

NUWARD™ is an integrated PWR design to generate 340 MWe from two independent reactor units, offering flexible operation. The design incorporates the main components of the Nuclear Steam Supply System (NSSS) including Control Rod Drive Mechanism (CRDM), Compact plate Steam Generators (CSGs) and pressuriser, all contained within the Reactor Pressure Vessel (RPV). The RPV is then installed within a further layer of steel containment and this structure is immersed in a pool filled with water (the reactor pool). The NUWARD™ design includes the capability to cope with Design Basis Conditions (DBC) using passive systems without the need for any external electrical power supply. The reactor is self-reliant, connected to an internal ultimate heat sink (the reactor pool) which offers a coping time of more than 3 days without the need for intervention.

2. Target Application

The NUWARD™ technology is being developed to replace fossil-fired power plants in the 300-400 MWe range; to supply power to remote municipalities and energy-intensive industrial sites; and to power grids with limited capacity. By design, it is a multipurpose SMR that can be used for cogeneration of heat and electricity, hydrogen production, district heating, and water desalination. The design offers baseload and load-following capability to enable integration with renewable energy sources.

3. Design Philosophy

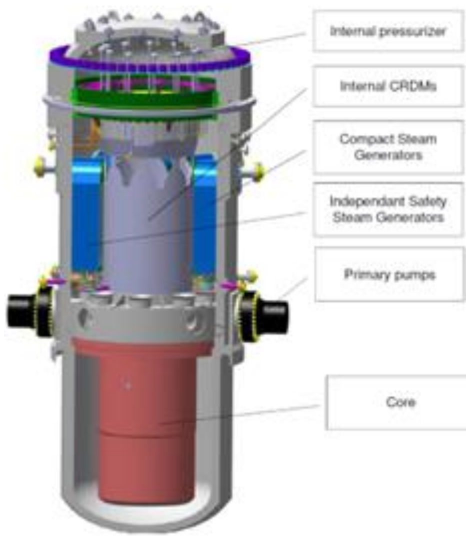
NUWARD™ design is based on the proven PWR-technology that incorporates significant experience acquired in the fields of medium and high-power generation, alongside several key technological innovations, to achieve the following design objectives:

i. Acceptability: robustness of the design will maximise safety and minimise environmental impact;
ii. Simplicity: simple architecture, enhanced manufacturability; and
iii. Schedule optimisation and constructability.

4. Main Design Features

(a) Nuclear Steam Supply System

The NUWARD™ reactor is a fully integrated PWR reactor, housing a single unique vessel containing all the main reactor coolant system components, including the CSGs, the pressuriser and the CRDM.



Reactor Cross-section View

(b) Reactor Core

The reference core is based on proven 17x17 (76) fuel assemblies used in the operating PWR fleet with a shortened height and UO_2 rods (enrichment < 5wt% ^{235}U). Due to the boron-free design, various ^{235}U enrichments and burnable poisons are used. The refuelling interval is 24 months for half the core.

(c) 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 operation in both normal conditions and DBCs, as well as a drastic reduction in effluents produced from operation.

(d) 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 Loss Of Coolant Accident (LOCA) size to a 30 mm diameter break.

(e) Reactor Coolant System and Steam Generator

The NUWARD™ reactor coolant system adopts an innovative steam generator technology using the plate heat exchanger concept. The CSGs are in direct connection to the reactor thus eliminate the need of external primary loops. This makes the design highly efficient with a high thermal power per volume ratio. The overall size of the reactor coolant system is therefore significantly reduced given the reactor thermal power.

(f) Pressuriser

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

(g) Primary pumps

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The NUWARD™ reactor and associated safety systems are designed for: (i) Passive management of DBC scenarios with no need for an external ultimate heat sink or any external electrical power supply (normal and emergency) for more than 3 days; (ii) Active management of Design Extension Condition (DEC)-A accidents, with simple diagnosis and implementation of diversified systems; and (iii) In-Vessel Retention of the corium (IVR) strategy for DEC-B management.

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: (i) Large reactor coolant inventory (kg/MWth) providing inertia versus power transients; (ii) Integrated reactor coolant system architecture reducing the maximum LOCA break size thus providing more coping time in the event of the design basis LOCA; (iii) Internal CRDMs preventing rod-ejection accidents; (iv) Boron-free in normal operation (including all DBCs) providing large and constant moderator counter-reaction and preventing boron dilution; (v) A metallic submerged containment providing passive cooling for several days, and (vi) A relatively small core in a large vessel enabling the efficient implementation of the IVR concept for DEC-B accident scenario.

(b) Safety Approach and Configuration to Manage DBC

NUWARD™ incorporates 2 trains of passive heat removal, via a natural circulation transfer system of the decay heat from the core to the water surrounding the unit containment (the reactor pool), through two dedicated Safety CSGs (S-CSG) independent from the operational or the operational six CSGs. Each train can be actuated by 2 diversified channels (diversified sensors and I&C) and redundant (backup spare) actuators. The water surrounding the unit 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 is considered a D-passive system according to IAEA classification. A set of 2 redundant low-pressure safety injection accumulators provide the make-up of reactor coolant water inventory in case of LOCA. NUWARD™ includes safety features to prevent criticality risks. The use of an internal CRDM eliminates the occurrence of a rod ejection accident. Dedicated safety systems to manage DBC are provided for each of the two reactor units.

(c) Safety Approach and Configuration to Manage DEC

DEC-A systems include Low flowrate depressurisation system and active 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 is available to cope with Anticipated Transients Without Scram (ATWS) accidents. DEC-B systems include the low flowrate depressurisation system; Flooding of the vessel pit in order to provide IVR of corium; and nitrogen injection to manage the risk of hydrogen combustion.

(d) Containment System

The NUWARD™ design includes steel containment as the 3rd barrier, which is immersed in water. The minimised 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. The containment is protected against hydrogen accumulation risk in DBCs by passive recombiners.

(e) Spent Fuel Cooling Safety Approach / System

Located between the 2 units that form the NUWARD™ design, is a shared spent fuel pool. The fuel assemblies are moved through a transfer chute located at the top of the steel containment; this feature is available on each unit. Along with the shared spent fuel pool is a shared refuelling machine used to achieve the fuel transfer.

6. Plant Safety and Operational Performances

The design target value for the lifetime capacity factor is above 90%, with the major planned refuelling only outages scheduled for 20 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.

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement

The Nuclear Island (NI) building is located below ground (semi-buried) for protection against external hazards and certain malicious acts; and increased ease of construction. The NI houses the 2 independent units and shared fuel storage pool. NUWARD™ plants are suitable for sea/lake-onshore and/or river-side sites, with open-loop conventional condenser cooling, as well as inland sites with aero condensers. The basic grid interface will be compliant with ENTSO-E and EUR requirements (typically 225kV/400kV and 50Hz).

NUWARD™ example plant layout

9. Testing Conducted for Design Verification and Validation

Various tests and studies are in progress to validate aspects of the design and its function, particularly around the residual heat removal passive system. Design feedback and evolution as a result of the outcomes from these studies and all other aspects of design and verification will follow. Certain aspects of the design are drawn from operational experience and design experience related to existing medium and high power PWRs.

10. Design and Licensing Status

NUWARD™ is close to completing its conceptual design phase at the time of writing and is preparing for pre-licensing. A safety options file (Dossier d'Options de Sûreté or DOS) will be completed by the end of the conceptual design to be formally reviewed by the French safety authority (ASN). Following EDF's initiative, NUWARD™ is the case study for an on-going Joint Early Review, led by ASN with the participation of STUK and SUJB, respectively the Finnish and the Czech safety authorities, from which early insights into the design development and safety approach are anticipated. Site permit for First Of A Kind (FOAK) is being pursued. Agreement has been reached with the Government of France that a FOAK NUWARD™ will be built in France. a number of potential sites are being considered for this.

11. Fuel Cycle Approach

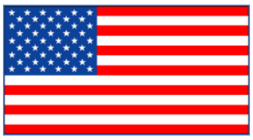
The reference plant refuelling cycle is for half a core every 2 years. The plant provides a storage of spent fuel assembly for 10 years after operation before decommissioning.

12. Waste Management and Disposal Plan

Options regarding waste disposal are currently under assessment, taking note of industry best practice.

13. 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 (and preparation for pre-licensing)
2023 – 2026	Basic Design Phase
06/2024	**EDF Group has decided to shift its SMR product strategy towards the development of a design based on proven technology bricks only.**



PWR-20 (Last Energy Inc., United States of America)

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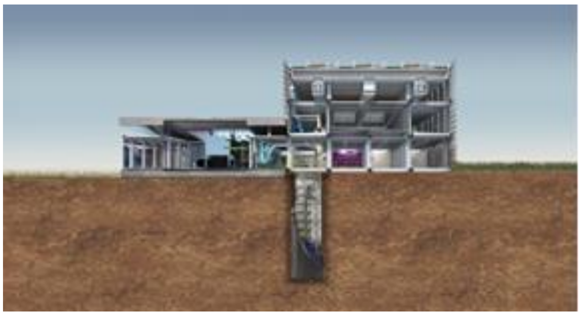


Isolated PWR-20 Nuclear Steam Supply System

Last Energy PWR-20 Cross-section

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Last Energy, United States of America
Reactor type	PWR
Coolant/moderator	Light water/light water
Thermal/electrical capacity, MW(t)/MW(e)	80 MWth / 20 MWe (nominal)
Primary circulation	Forced (4 pumps)
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 2.8
Core Inlet/Outlet Coolant Temperature (^{o}C)	270 / 331
Fuel type/assembly array	UO_2 pellet / 17x17 grid
Number of fuel assemblies in the core	12 assemblies
Fuel enrichment (%)	<4.95
Core Discharge Burnup (GWd/ton)	30
Refuelling Cycle (months)	72
Reactivity control	Control rods and negative moderator coefficient
Approach to safety systems	Défense-in-depth, Redundancy, Diversity, and Passive Safety
Design life (years)	42
Plant footprint (m^2)	968
RPV height/diameter (m)	8.9/1.8
RPV weight (metric ton)	39
Seismic Design (SSE)	0.5g horizontal and 0.4g vertical peak ground accelerations
Distinguishing features	Modular Construction, Factory-Fabricated, Air-Cooled, Zero Water Consumption
Design status	Detailed Design

1. Introduction

Last Energy PWR-20 is a production-ready, modular Pressurized Water Reactor (PWR) designed for distributed baseload applications. With a nominal electric output of 20 MWe, the PWR-20 aims to provide reliable, modular energy solutions with a focus on standardization and ease of deployment. The design leverages proven PWR technology and avoids new innovations in fuel or reactor physics.

2. Target Application

The PWR-20 is intended for distributed baseload power applications, including industrial siting and combined heat and power scenarios. Its compact size and modular nature allow for flexible deployment and integration with energy consumers.

3. Design Philosophy

The PWR-20 emphasizes modularization, standardization, and ease of construction. By using off-the-shelf components and proven technology, the design minimizes risks and costs associated with new reactor technology.

4. Main Design Features

(a) Nuclear Steam Supply System

The Nuclear Steam Supply System is engineered with a single-loop configuration, optimizing thermal management and reducing complexity in the heat transfer process. Its modular components are designed to streamline assembly and maintenance, ensuring efficient operation throughout the reactor's life. This design facilitates rapid installation and minimizes operational disruptions.

(b) Reactor Core

The reactor core is organized using a standard 17x17 fuel assembly grid, which is a well-established configuration that supports reliable and uniform neutron flux distribution. The core utilizes low-enrichment uranium fuel, providing a stable and controlled nuclear reaction while maintaining high safety standards. This arrangement ensures consistent reactor performance and reduces the risk of operational issues.

(c) Reactivity Control

The reactivity control system integrates both control rods and burnable poisons to precisely manage the reactor's reactivity levels. Control rods are used to absorb excess neutrons, while burnable poisons gradually absorb neutrons over time, balancing the core's reactivity throughout the fuel cycle. This dual approach enhances core stability and operational safety.

(d) Reactor Pressure Vessel and Internals

The reactor pressure vessel and its internals are designed as a compact, integrated unit that can be prefabricated in a factory setting. This design minimizes the time required for on-site construction and reduces the potential for errors during assembly. The compact design also supports efficient use of space and resources.

(e) Reactor Coolant System and Steam Generator

The reactor coolant system employs forced convection to circulate coolant through the reactor core, which is critical for maintaining optimal temperatures and efficient heat transfer. It is complemented by an air-cooled tertiary loop that enhances overall cooling efficiency and helps manage excess heat. This configuration ensures reliable operation and effective heat dissipation.

(f) Pressuriser

The pressurizer is designed as a separate vessel to maintain the necessary pressure within the reactor's coolant system. This design helps regulate the coolant pressure, preventing fluctuations that could impact reactor performance and safety. By isolating the pressurizer from other system components, the design simplifies pressure management and enhances operational stability.

(g) Primary pumps

Standard primary pumps are used in the PWR-20 to maintain forced circulation of coolant through the reactor core and heat exchangers. These pumps are integral to ensuring a steady flow of coolant, which is crucial for maintaining reactor temperature and operational stability. Their use of proven, reliable technology contributes to the reactor’s overall efficiency and ease of maintenance.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The PWR-20's safety system is fully passive, requiring no active controls or operator actions during emergencies. It automatically manages accidents without external power, enhancing reliability and reducing risk. This design ensures core safety through automatic, self-sufficient mechanisms.

(b) Safety Approach and Configuration to Manage DBC and DEC

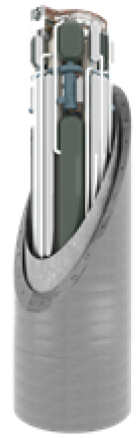
The PWR-20 manages Design Basis Conditions (DBC) and Design Extension Conditions (DEC) with fully passive safety systems, requiring no operator intervention during emergencies. It features multiple redundant safety layers to handle all credible accident scenarios, ensuring reactor stability and safety even in extreme conditions.

(c) Core Damage Frequency (CDF) and Large Release Frequency (LRF) Controls

The PWR-20 is designed with a core damage frequency (CDF) target of 10^{-7} per reactor-year, emphasizing its high safety standards. Its design ensures no core melt scenarios, thanks to fully passive safety systems. The large release frequency (LRF) is managed through robust containment provisions, which include a thick iron cask and multiple layers of defense, preventing significant radioactive releases.

(d) Containment System

The PWR-20's containment system consists of a 500-metric-ton iron cask, mostly situated underground, with thick metal walls and an interior liner to prevent leaks. This design ensures containment of radioactive materials and effectively manages Loss of Coolant Accidents (LOCAs) and external hazards.



(e) Spent Fuel Cooling Safety Approach / System

The PWR-20 follows a six-year refuelling cycle where the reactor is taken offline, and the spent fuel is passively cooled. The system is designed so that after each cycle, the reactor's spent fuel is left to cool naturally, without active mechanical systems, by relying on passive methods such as natural convection and thermal radiation.

6. Plant Safety and Operational Performances

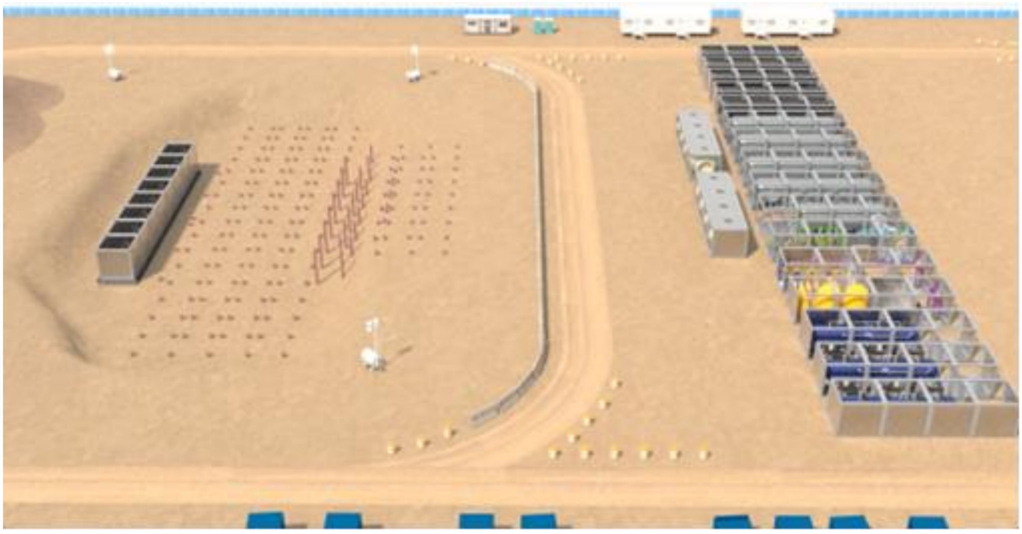
The PWR-20 is designed with substantial safety margins and fully passive safety systems, ensuring that all credible accident scenarios are managed without operator intervention. This design philosophy minimizes operator involvement and maximizes reliability by incorporating extensive safety features that do not rely on active controls or electronic systems.

7. Instrumentation and Control Systems

The reactor utilizes advanced instrumentation and control systems, including Programmable Logic Controllers (PLC) and Distributed Control Systems (DCS), to automate plant operations and enhance performance monitoring. These systems ensure efficient operation, maintain safety standards, and provide real-time data for operational adjustments, all while simplifying the management of the plant's numerous components.

8. Plant Layout Arrangement

The PWR-20 features a modular design with factory-fabricated components, allowing for rapid on-site assembly. This modular approach not only accelerates the construction process but also optimizes spatial efficiency and reduces the overall footprint required for installation, making it adaptable to various site conditions.



PWR-20 On Site Assembly Layout

9. Testing Conducted for Design Verification and Validation

The reactor design has undergone extensive testing and validation processes to ensure compliance with regulatory standards and operational reliability. This includes thorough operational and safety assessments that verify the design's ability to meet safety requirements and performance expectations in real-world scenarios.

10. Design and Licensing Status

The PWR-20 is currently in various stages of licensing across multiple countries, with commercial operation expected to commence in 2025 following successful regulatory approvals. The design has progressed through detailed planning and pre-licensing reviews, positioning it for timely deployment and integration into the global energy market.

11. Fuel Cycle Approach

The reactor operates on a six-year refuelling cycle, during which it requires minimal on-site fuel handling. This design simplifies plant operations and maintenance, reducing the need for complex fuel management processes and enhancing overall safety by minimizing worker exposure to radioactive materials.

12. Waste Management and Disposal Plan

Spent fuel management for the PWR-20 includes repurposing the reactor vessel as a spent fuel cask. This approach aligns with best practices for environmental protection and regulatory compliance, ensuring safe and effective disposal and management of radioactive waste over the reactor's lifetime.

13. Development Milestones

2018	Utility and government need-finding through Titans of Nuclear podcast	Complete
2020	Launch of OPEN100 (open source PWR) for technical and economic validation	Complete
2021	Preliminary studies and pre-conceptual design for a 20 MWe scale reference plant	Complete
2022	Detail design complete and pre-licensing overviews with four (4) nuclear regulators	Complete
2024	Agreements signed for deployment of 57 units across four countries	On-Track
2025	Commercial operation of reference plant	Planned
2026	10x 20MWe plants online	Planned
2027	Assembly line manufacturing begins	Planned

RITM-200N (JSC “Afrikantov OKBM”, Russian Federation)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Afrikantov OKBM , Rosatom, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Water / Water
Thermal/electrical capacity, MW(t)/MW(e)	190 / 55
Primary coolant system circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.7 / 3.83 (steam pressure)
Core inlet/outlet coolant temperature (°C)	283 / 321
Fuel type/assembly array	UO_2 pellet/hexagonal assembly
Number of fuel assemblies in the core	199
Fuel enrichment (%)	< 20
Refueling cycle (months)	60 – 72
Core discharge burnup (GW·d/ton)	–
Reactivity control mechanism	CRD mechanisms of CPS
Approach to safety system	Combined active and passive safety systems
Design service life (years)	60
NPP footprint (m^2) (rectangular building envelope)	120 000
RPV height/diameter (m)	7.5 / 3.4
RPV weight (metric ton)	164
Seismic design (SSE)	9 points on the MSK-64 scale
Requirements for or approach to the operating cycle	Refueling every 5 – 6 years
Distinguishing features	Modular design; integral reactor; in-vessel corium retention in severe accidents
Design status	Detailed design

1. Introduction

The RITM-200 series reactors are the Afrikantov OKBM latest development in the Generation III+ SMR lineup. While possessing all the best characteristics of their predecessors, these reactors are based upon the time-tested PWR technology and upon the Rosatom’s 400-reactor-year operating experience with small-sized reactors on board icebreakers. Six RITM-200 reactors have been successfully installed on icebreakers *Arktika*, *Sibir* and *Ural*. The lead icebreaker *Arktika* with two reactors on board was commissioned in October of 2020. The first in the series, the icebreaker *Sibir*, was commissioned in January of 2022. For late 2022, it is planned to commence operations of the icebreaker *Ural*. It is planned to construct 2 more icebreakers of this class – *Yakutia* and *Chukotka*. The concept of and engineering solutions for the Afrikantov OKBM -developed RITM-200 reactor plant are used in the RITM-200N reactor plant design for a pilot land-based small-sized nuclear power plant.

The standardized design-schematic solutions and construction-layout solutions adopted in the RITM-200N design and in the small-sized NPP design make it possible to implement a power lineup of energy-generating sources – 55 MW (1 reactor plant + 1 STP in the main building); 110 MW (2 reactor plants + 2 STPs in the main building); 220 MW, 330 MW by a modular principle through constructing additional units with their own main buildings and cooling towers and with common plant-shared systems.

2. Target Application

The RITM-200N reactor plants may be used to generate electricity, cogenerate electricity and heat, and to desalinate sea water.

3. Design Philosophy

The reactor plant systems and equipment and nuclear fuel handling equipment were designed using the proven engineering solutions adopted by the nuclear propulsion plants in operation. The design is developed in compliance with the requirements of federal standards and regulations in nuclear energy use for NPPs. IAEA requirements are incorporated. Enhanced safety requirements for Generation III+ reactors are met. Modularity is adopted for design and transport. An industrial large-unit installation process is used for construction.

4. Main Design Features

(a) Nuclear Steam Supply System

The NSSS adopts an integral PWR. The reactor consists of a reactor core, four steam generators built in the RPV, four canned-motor RCPs, control rod drive mechanisms of the CPS. The primary coolant system uses forced circulation during normal operation and ensures natural circulation under accident conditions.

(b) Reactor Core

Low-enrichment fuel assemblies are used as in KLT-40S. This ensures long-time operation without refueling and meets the international non-proliferation requirements. The core height is 1650 mm. The core comprises an array of 199 fuel assemblies.

(c) Reactivity Control

Control rods are used to control reactivity. A group of CPS control rod drive mechanisms is provided to compensate for the excessive reactivity during a startup, power operation and a reactor scram.

(d) Reactor Coolant System

The primary coolant heat is transferred in steam generators sitting in the reactor pressure vessel. The steam generators generate superheated steam from feed water and transfer the steam to the steam turbine plant (STP). The secondary coolant system consists of four loops that include a steam generator, steam and feed water piping, valves, automatic safety devices and measuring instruments. The steam generators provide steam with a temperature of 295 °C, pressure of 3.83 MPa and an output of 305 t/h.

(e) Steam Generator

The RITM-200N reactor plant design uses once-thorough straight-tube SG. The configuration of the steam generating cassettes makes it possible to compactly place them in the reactor pressure vessel. The SGs are divided into four loops of the primary coolant system. Each loop consists of three cassettes (making up the total of 12) with shared feed water and steam headers. Each cassette contains 7 modules.

(f) Pressurizer

The RITM-200N reactor plant uses an external gas pressurizer system that is well proven in the Russian marine power engineering. The system is known for its design simplicity, which enhances reliability, ensures compactness and does not require any electricity. The pressurizer system is divided into two independent groups to reduce the pipeline diameter in the reactor nozzles and to reduce the coolant leakage in large break LOCAs in the system piping. This solution allows one of the pressurizers to be used as a hydraulic accumulator, which significantly enhances the reactor plant reliability in possible loss-of-coolant accidents.

5. Safety Features

The safety concept of the RITM-200N reactor plant is based upon the defense-in-depth principle in combination with the inherent safety features and the use of active and passive systems. The inherent safety features are intended to limit the core power output as a function of the primary coolant pressure and temperature, the heat generation rate, the primary coolant system leak volume and outflow rate, the failed fuel fraction, the maintained RPV integrity in severe accidents. The RITM-200N reactor plant optimally combines passive and active safety systems, which cope with the events associated with abnormal operation, design-basis accidents and beyond-design-basis accidents.

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

The high safety level of the RITM-series reactors is achieved through inherent safety features and through a combination of passive and active safety systems. In addition to that, it is provided that equipment and channels in the safety systems be redundant and functionally and/or physically separated to ensure high-level reliability.

The control rods of the CPS fall into the core by gravity or driven by a spring when the power supply is disconnected from the electromagnetic couplings, which is followed by a reactor shutdown even in the case of a complete blackout.

(b) Residual Heat Removal System

The RHRS consists of four safety channels: an active safety loop with forced circulation through the steam generator; an active safety loop with forced circulation through primary-to-third coolant system heat exchanger of the primary coolant purification system; two passive safety loops with natural coolant circulation through steam generators from water tanks. The water evaporated in the SGs condenses in the air-cooled heat exchangers and comes back to the tanks with water-cooled heat exchangers. When the water is completely evaporated from the tanks, the air-cooled heat exchangers continue the cooling for an unlimited time. The combination of air-cooled and water-cooled heat exchangers allows the overall dimensions of the heat exchangers and water tanks to be minimized. All safety channels are connected to different SGs and ensure that residual heat be removed in compliance with the single failure criterion.

(c) Emergency Core Cooling System (ECCS)

The ECCS consists of a safety injection system (SIS) to inject water into the primary coolant system in order to mitigate the consequences of a LOCA resulting from a pipeline break. The system is based on active and passive safety principles with redundant active elements in each channel. The ECCS consists of: Two pressurized passive hydraulic accumulators; Two active channels with water tanks and two makeup pumps in either channel. In combination with the residual heat removal system, the passive safety channels provide a 72-hour grace period without any actions by personnel or upon a loss of power in a LOCA plus a complete blackout.

(d) Containment System

The reactor plant is placed in a leak-tight enclosure in the form of a steel containment. The containment includes three levels: (i) the first containment is shaped as a cylinder 8.7 m in diameter and 22 m tall. It is designed for internal overpressure of up to 0.9 MPa, and is placed around the reactor pressure vessel to isolate a possible leak of radioactive products; (ii) the second containment is a strong containment of the building. It is made of thick reinforced concrete walls (800 mm thick) to protect the first containment from external events; (iii) the third level of the containment is a building structure of reinforced concrete walls to dissipate most of external impact energy and to minimize the impact to the second containment. The strong containment design and destructible elements take into account the maximum potential external events including a crash of large commercial aircraft.

6. Plant Safety and Operational Characteristics

In the development of the RITM-200 plants and nuclear energy generating sources equipped with the RITM-200-type plants, the priority area is preventing abnormal operation and accidents with account of the developing and operating experience in marine plants and nuclear generating stations. The RITM-200N design is developed in conformity with Russian laws, standards and rules for nuclear power plants; in conformity with the safety principles developed by the world community; and in conformity with IAEA recommendations.

7. Instrumentation and Control System

To monitor and control plant processes, an automated control system is provided in the small-sized NPP fitted with the RITM-200N. In terms of safety functions, this system has the necessary redundancy and ensures both automated and remote control of the power plant.

8. General Layout of the Plant

The basic option of the small-sized NPP includes two RITM-200N reactors with a total installed electric power of 110 MW. The basis for the placement of buildings and structures on the site is based on the principle of zoning, which ensures maximum separation of buildings and structures according to their functional purpose in compliance with sanitary and fire gaps between buildings. Below is a general view of the small-sized NPP.

The reactor building and turbine building constructed on additional areas allow a gradual growth of energy generation in 110 MW increments. In this case, the site area for a 100 MW small-sized NPP is 27 acres (0.11 km^2); for 220 MW, 38 acres (0.15 km^2); for 330 MW, 49 acres (0.19 km^2).

9. Testing to Check and Validate the Design

The engineering solutions used in the reactor plant are traditional for marine power engineering. The solutions have been tested in the course of many operating years and ensured the required reactor plant reliability and safety performance. The RITM-200N reactor pertains to integral-type reactors. Integral-type reactors are used in a series of Project 22220 multipurpose nuclear-powered icebreakers *Arktika*, *Sibir* and *Ural*.

10. Design and Licensing Status

The small-sized NPP fitted with the RITM-200N reactor plant is at a design stage. A decision has been made to construct a pilot small-sized NPP in the Arctic zone near the town of Ust-Kuiga, Ust-Yansky ulus, the Republic of Sakha (Yakutia), Russia. The declaration of intent to invest into the construction of the small-sized NPP based on the 55+ MW RITM-200N reactor plant in Ust-Yansky ulus, the Republic of Sakha (Yakutia), Russia was approved by Rosatom Director General A.Ye. Likhachev. It is planned to obtain the site license in March of 2023.

11. Approach to the Operating Cycle

In the small-sized NPP, the nuclear fuel handling system uses a refueling complex analogous to that developed for the multipurpose nuclear-powered icebreakers fitted with the RITM-200 reactor plant. The refueling complex is used to load the core into the reactor and to unload the core to the spent fuel pool (SFP), to load spent fuel assemblies (SFAs) into shipping containers to transport them for reprocessing.

12. Waste Management System and Waste Disposal Plan

The design provides for the appropriate systems and equipment for radwaste handling (to convert liquid radwaste into a solid phase, to shred solid radwaste, to pack it in special containers) and temporary storage at the small-sized NPP.

13. Development Milestones

2021	Declaration of Intent approved by the Republic of Sakha (Yakutia)
2022	RITM-200N detailed design developed
2023	Siting License obtained
03/2024	SNPP design documentation developed
08/2024	Construction license to be obtained
02/2027	Operation license to be obtained
10/2027	SNPP to generate electricity for a first time

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

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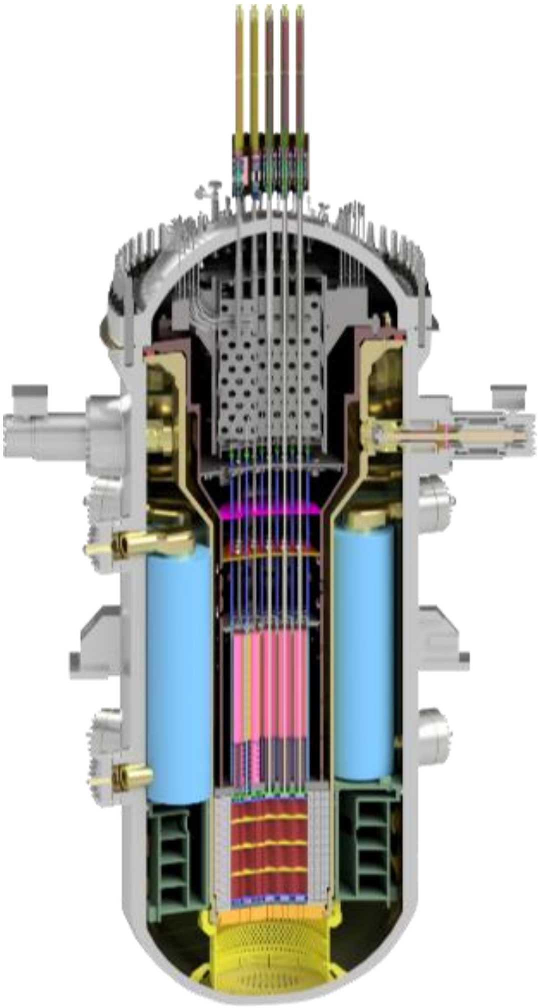


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	KAERI, Republic of Korea and K.A.CARE, Kingdom of Saudi Arabia
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	365 / 107
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15 / 5.8
Core inlet/outlet coolant temperature (°C)	296 / 322
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	57
Fuel enrichment (%)	< 5
Refueling cycle (months)	30
Core discharge burnup (GWd/ton)	< 54
Reactivity control mechanism	Control rod drive mechanisms and soluble boron
Approach to safety	Passive
Design life (years)	60
Plant footprint (m^2)	90,000
RPV height/diameter (m)	16.8 / 6.0
RPV weight (metric ton)	1070 (including coolant)
Seismic design (SSE)	> 0.3 g with 0.18 g of automatic shutdown
Fuel cycle requirements/approach	Conventional LWR requirements applied (spent fuel storage capacity: 30 years)
Distinguishing features	Coupling with desalination and process heat application, integrated primary system
Design status	Detailed design

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 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

The SMART is a multi-purpose reactor for electric power generation, desalination, district heating, and process heat for industries. SMART has been developed to be 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. 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 period and high plant availability.

4. Main Design Features

(a) Nuclear Steam Supply System

The SMART has an integral reactor coolant system (RCS) configuration that enables the elimination of a large break loss of coolant accident (LBLOCA) from the design basis events. The nuclear steam supply system (NSSS) consists of the RCS forming a reactor coolant pressure boundary, secondary system, chemical and volume control system (CVCS), component cooling water system (CCWS), passive residual heat removal system (PRHRS), passive safety injection system (PSIS), automatic depressurization system (ADS), containment pressure and radioactivity suppression system (CPRSS), etc.

(b) Reactor Core

The low power density design with a slightly enriched UO_2 fueled core ensures a thermal margin of greater than 15%. In the core, there are 57 fuel assemblies 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 refueling scheme without reprocessing provides a cycle of 870 effective full power days for operation.

(c) 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 (CRDM) which has been widely used in the commercial nuclear power plants (NPPs).

(d) Reactor Pressure Vessel and Internals

The RPV houses the reactor core, 8 steam generators (SGs), 4 canned motor reactor coolant pumps (RCPs), 25 control rod drive mechanisms (CRDMs) and reactor internals such as the core support barrel assembly and the upper guide structure assembly.

(e) Reactor Coolant System

The RCS transfers core heat to the secondary system through the SGs and plays a role of a barrier that prevents the release of reactor coolant and radioactive materials to the reactor containment. The major components of the RCS are a reactor vessel assembly containing the core, pressurizer space, SGs, RCPs, CRDMs, and related pipes, valves, and instrumentations. The forced circulation flow of the reactor coolant is formed along the flow path by the RCPs during normal operation. The RCS and its supporting systems are designed with sufficient core cooling margin for protecting the reactor core from damage during all normal operation and anticipated operational occurrences (AOOs).

(f) Steam Generator

The SMART has 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 in emergency conditions. The small inventory of the secondary side (tube side) water in each SG prohibits a return to power following a main steam line break accident. In case of an accident, the SG can be used as the heat exchanger for the PRHRS.

(g) Pressurizer

The in-vessel pressurizer uses the free volume in the upper part of the RPV. The primary system pressure during normal operation 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 maneuvering operation. The reactor over-pressure at the postulated design basis accidents can be reduced through the actuation of pressurizer safety valves.

(h) Primary Pumps

The RCPs are installed horizontally on the external wall of the reactor pressure vessel. It is a mixed flow pump adopting a canned motor. It consists of the pressure retainer, the impeller and diffuser, the shaft assembly, and the motor. No coupling is needed to connect the impeller shaft and the motor shaft. All of the pump parts are enclosed by the pressure retainer. Therefore, there is no mechanical seal device to prevent the reactor coolant from leaking through the pressure retainer. The component cooling water flowing in the helical tubes removes the heat on the motor. The reactor coolant pump sucks the reactor coolant through the annulus between the core support barrel and upper guide structure, and then it discharges the reactor coolant to the space above the steam generator.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

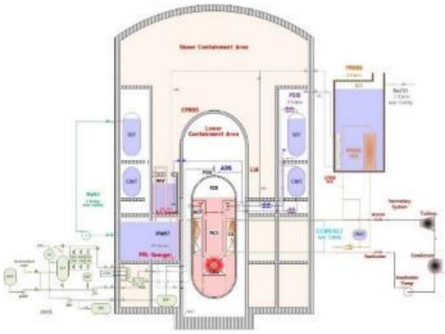
Safety systems of SMART are designed to function automatically on demand. These consist of the PRHRS, PSIS, and CPRSS. Additional safety systems include the ADS and pressurizer safety valves, and a severe accident mitigation system.

(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

The PSIS provides emergency core cooling following postulated design basis accidents. Emergency core cooling is performed using 4 core make-up tanks (CMTs) and 4 safety injection tanks (SITs). Core cooling inventory is maintained through passive safety injection from CMTs and SITs. The CMTs that are full of borated water provide makeup and borating functions to the RCS during the early stage of an SBLOCA or non-LOCA. 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.



Emergency Core Cooling System

(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 CPRSS as a passive safety system. The containment system is composed of a lower containment area (LCA), an upper containment area (UCA), an in-containment refueling 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.

6. Plant Safety and Operational Performances

The core damage frequency of SMART is estimated to be less than 2E-7 per reactor year for internal events. Power maneuvering 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.

7. Instrumentation and Control System

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.

8. Plant Layout Arrangement

SMART NPP has been designed to have a seawater intake structure and other buildings including chlorination building in the yard. Power block accommodates reactor containment and auxiliary buildings (RCAB), turbine generator buildings and one compound building shared by two units of SMART. The RCAB houses reactor containment, auxiliary and fuel handling areas to adapt the small and modular plant concept. Reactor containment area consists of the LCA and UCA. The LCA houses the RPV, CMTs, and SITs. Auxiliary area houses emergency cooldown tanks, main control room (MCR), electrical and control facilities, and safety-related equipment required to provide safe shutdown capability. The balance of plant (BOP) design consists of turbine generator buildings and electric power systems.

Plant Layout Arrangement

9. Testing Conducted for Design Verification and Validation

The advanced design features of SMART were verified by a comprehensive technology validation program that includes safety tests and performance tests. The safety tests consist of core critical heat flux tests, separate effect and integral effect tests of the safety systems, thermal-hydraulic experiments, and digital man-machine interface system (MMIS) tests. The performance tests cover fuel assembly out-of-pile tests, performance tests of the major components including RPV dynamic tests, RCP mockup test and SG irradiation test, and MMIS control room tests.

10. Design and Licensing Status

Korea Atomic Energy Research Institute (KAERI) received the standard design approval for SMART from the 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 (PPE) 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. As the results of SMART PPE project, the SMART is equipped the fully passive safety systems and the improved reactor designs in detail. The SMART with passive safety systems and improved design is named SMART100. The SDA application and licensing documents including the standard safety analysis report (SSAR) for SMART100 were submitted to the NSSC in December 2019.

11. 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.

12. Waste Management and Disposal Plan

The 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.

13. Development Milestones

1997	Conceptual design development
2002	Basic design development
2009	SMART-PPS (Pre-Project Service)
2012	Technology verification, Standard Design Approval (SDA)
2012	SMART safety enhancement project
2015	SMART pre-project engineering agreement signed between KAERI and K.A.CARE for the global commercialization of SMART
2019	SMART pre-project engineering completed
2019	SMART100 Standard Design Approval applied
2024	SMART100 Standard Design Approval issued

STAR (STAR ENERGY SA, Switzerland)

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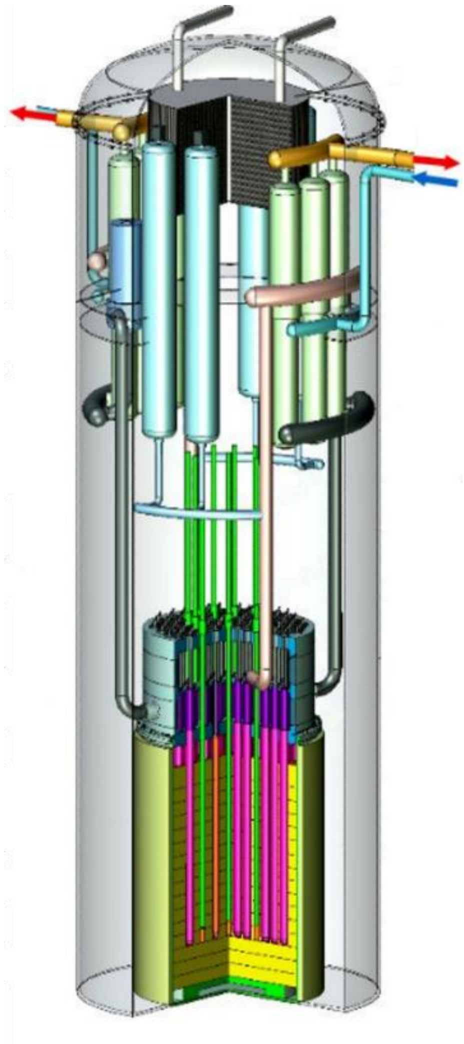


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	STAR ENERGY SA Switzerland
Reactor type	Pressure tube light water reactor
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	30 / 10
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	12.5 / 4.4
Core inlet/outlet coolant temperature (°C)	270 / 300
Fuel type/assembly array	UO_2 pellet assemblies in individual pressure channels
Number of fuel assemblies in the core	138
Fuel enrichment (%)	19
Refuelling cycle (months)	120
Core discharge burnup (GWd/ton)	60
Reactivity control mechanism	Control rods and Er / Gd_2O_3 burnable absorber
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	36 000
RPV height/diameter (m)	7.6 / 2.3
RPV weight (metric ton)	20
Seismic design (SSE)	Degree VIII (MSK-64)
Fuel cycle requirements/approach	Once-through HALEU; no on-site storage
Distinguishing features	Compact, low-weight design without a pressure vessel; flexibility of operation; intrinsic passive safety
Design status	Basic design

1. Introduction

“STAR” is a channel-type two-circuit light-water nuclear reactor with a thermal rate of 30 MW(t) and a nominal electric power output of 10 MW(e). Based on time-proven solutions utilized in multiple contemporary light-water reactor designs, “STAR” features a distinctive layout, wherein the coolant of a double-loop primary circuit circulates through individual pressurized channels instead of an outer pressure vessel, which is the main factor contributing to the mass and dimensions of traditional PWRs. The channels’ design in the form of bayonet heat exchangers and the use of UO_2 fuel enriched to 19 percent further decrease the reactor unit’s size, which effectively enables it to be fully assembled in factory and delivered to the installation site by land, sea, or air, substantially decreasing its production and maintenance costs.

2. Target Application

The “STAR” reactor is designed to serve as a versatile tool to fulfil the energy needs of remote or isolated localities, such as small settlements or power-intensive industrial areas. Depending on the configuration of the secondary island, it is capable of functioning as a district heating or electric power plant or operate in cogeneration mode. Additionally, it can be utilized as a source of heat or steam for water desalination or other industrial purposes.

3. Design Philosophy

Since SMRs generally do not enjoy the economy-of-scale benefits offered by larger nuclear power plants, the main goal of the design is to provide a financially viable alternative to small-scale fossil fuel power stations while maintaining the required high safety standards. This goal is mainly achieved by combining well-tested technical solutions used in existing light-water reactors with a unique layout tailored for ease of manufacturing, assembly, transportation, and installation of the reactor. The design assumes that the fabrication process shall only require commercially available parts and materials to reduce the initial investment required for research and licensing. It also includes multiple redundant passive safety features following the "defence in depth" principle with a further goal to reduce the power plant's total footprint, limit the required controlled zone to its boundaries, and enable deploying it near the end user.

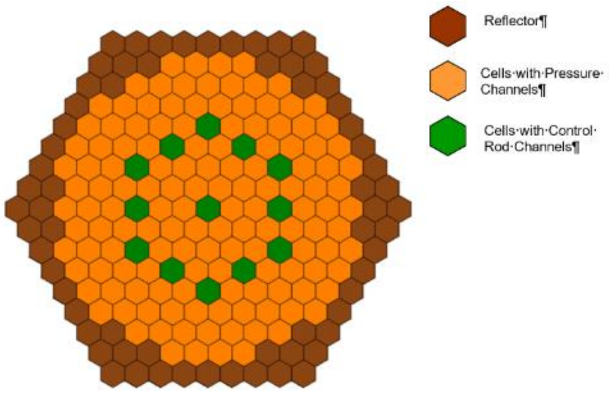
4. Main Design Features

(a) Power Conversion

There are three primary possible variants of nuclear energy conversion at a "STAR" power plant: a nuclear electric plant referred to as "STAR-E", a nuclear heating plant ("STAR-T"), and a combined nuclear power plant ("STAR-C"). The design of the secondary circuits for all three variants is based on the parameters of the reactor unit's two steam generators with a combined output of 30 MW_t. The admission pressure of steam produced in the steam generator is 4.4 MPa with a steam dryness of 99%. The feed water temperature is 220°C.

(b) Reactor Core

The reactor core consists of 138 pressure channels, each designed as a bayonet heat exchanger with its outer tube welded into the lower plate of the primary coolant inlet chamber and the inner tube welded into the lower plate of the primary coolant outlet chamber. The coolant is fed to the core through the primary coolant inlet chamber, from which it flows along the outer section of a pressure channel and rises along the inner tube, which houses a fuel assembly. The tubes are manufactured from a zirconium-niobium alloy, and the space between the tubes is filled with a neutron-transparent material. The reactor core is encased into a neutron reflector consisting of stacked Al-Be alloy rings with a thickness of 100 millimetres.



Reactor core layout

(c) Fuel Characteristics

A "STAR" nuclear fuel assembly consists of 18 nuclear fuel rods placed in a triangular grid in two concentric circles around a central support rod, fixed in place with spacing meshes. The central support rod is used to attach the fuel assembly to the upper plate of the primary coolant outlet chamber and the drive mechanism. The fuel rods contain fuel pellets made of uranium dioxide with a maximum enrichment of 19%. Additionally, 6 of the 18 fuel rods contain Er or Gd_2O_3 burnable absorber, which compensates for loss of reactivity during normal operation of the reactor and extends its fuel cycle, which is set to 10 years. The nuclear fuel stays inside the core throughout the entire fuel cycle, including the scheduled maintenance periods.

The inner volume of the nuclear fuel rods is filled with high-purity helium to improve thermal conductivity of the gas mixture inside the fuel rods and to reduce the average volumetric fuel temperature.

(d) Reactivity Control

Reactivity control is performed with burnable erbium and gadolinium oxide absorbers in a number of the fuel assemblies and ^{10}B reactivity control rods evenly distributed inside the core. Boric acid is not intended for reactivity control during normal operation; however, it can be used for ensuring safe shutdown of the core during accident conditions. Additional reactivity control is also provided by the negative thermal feedback of the moderator.

(e) Reactor Pressure Vessel and Internals

The "STAR" reactor does not feature a pressure vessel. Instead, the pressure of the coolant is contained by the elements of the primary circuit, which, together with the I&C systems, are bundled into a "monoblock" layout and encased into a sealed cylindrical thin-walled (1 cm thick) steel reactor vessel that ensures convenient transportation and installation of the reactor unit and serves as an additional containment barrier. The vacant space between the reactor vessel and the internals is filled with high purity helium. The total height of the reactor module is 7.6 meters with a diameter of 2.3 meters.

(f) Reactor Coolant System

The primary circuit coolant system includes 138 pressure channels containing the reactor core's fuel assemblies, coolant inlet and outlet chambers, and two primary circuit loops, each featuring a pressurizer, a main circulation pump, and a triple-sectioned steam generator. The double-loop layout has been selected to reduce the overall size of the reactor module. The primary circuit is designed to hold an operating pressure of 12.5 MPa and maintain an outlet coolant temperature of 300°C.

(g) Secondary System

The secondary circuit connects to the reactor vessel from the outside through openings for feed water and steam output pipelines, interfaces with the primary coolant circuit through two steam generator arrays inside the reactor vessel and provides the final output of the reactor module. Like the primary circuit, the secondary circuit has forced circulation, so the secondary island must be equipped with a high-pressure feed water pump. The circuit's output has a temperature of 256°C with a feed water temperature of 220°C.

(h) Steam Generator

Unlike other reactors currently in operation, the reactor contains two steam generators, each consisting of three separate sections connected through inlet and outlet pipelines. Each steam generator is designed as a once-through unit with an inlet of the primary coolant in its upper part and the outlet in the lower part. The upper part of a steam generator contains centrifugal and louvered separators. The combined thermal power rate of all the steam generators is 30 MW.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The design of the "STAR" reactor follows the principle of inherent safety and provides for both long-term heat removal after reactor shutdown and emergency cooling during design basis accidents, including mechanically triggered safety measures requiring no power source or human input.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

During normal operation, sufficient radiation and thermal protection is achieved with several layers of shielding, including the reactor core's reflector, a cushion of pure inert gas, an airtight reactor vessel, and a reactor shaft shielding. In the event of an emergency reactor shutdown, the design of "STAR" offers systems for both short-term emergency cooling and long-term residual heat removal. The short-term emergency cooling system is triggered in a loss of coolant accident by several mutually independent mechanical sensors. The system injects pressurized water from an emergency tank into the reactor vessel to provide cooling to the core and prevent its exposure. The long-term cooling system is designed to passively remove decay heat from the first circuit through steam generators; it has been estimated that the first circuit shall maintain natural circulation of coolant even in the event of failure of both main circulation pumps. Finally, in the occurrence of a guillotine rupture of a primary circuit pipeline, additional cooling shall be provided by a steam condenser installed at the top of the reactor vessel.

(c) Spent Fuel Cooling Safety Approach / System

Initial calculations and modelling show that, given proper handling and fractioning, spent fuel from a "STAR" reactor cools passively and does not require temporary on-site storage in a spent fuel pool. Requirements to equipment and procedures for waste management shall be elaborated at further stages of the design.

(d) Containment System

The design's containment system is based on the "defence in depth" philosophy and includes a set of technical and administrative measures aimed at preventing environmental exposure to radiation and radioactive substances from the reactor.

The set of barriers includes:

- Fuel pellet matrix;
- Fuel rod casings;
- Pipelines end equipment of primary coolant circuit;
- Reactor core pressure channels;
- Reactor core shielding;
- Airtight reactor vessel;
- Ferrocement reactor shaft.

(e) Chemical Control

The functions of and requirements for the chemical control systems shall be elaborated on further stages of the design.

6. Plant Safety and Operational Performances

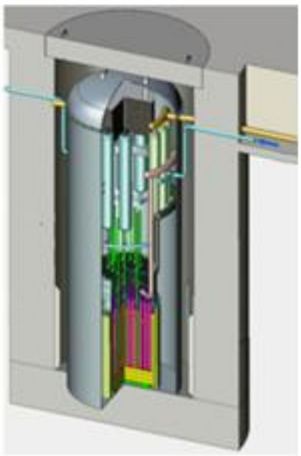
From the beginning of development of "STAR", reasonable safety requirements to be imposed for the reactor's design features are as follows:

- Core damage frequency – less than 10^{-6}/reactor year
- Large early radioactivity release frequency – less than 10^{-7}/reactor year

7. Instrumentation and Control System

On the conceptual level, the I&C system is divided into the following main blocks:

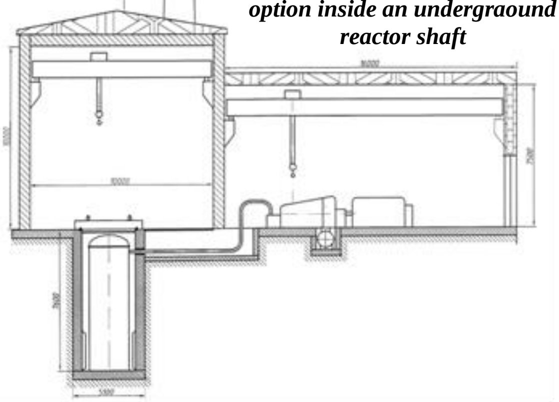
- Primary and reserve control rooms;
- Reactor control and protection system, including remote controls and two independent emergency shutdown systems;
- Process parameters control system;
- Internal reactor control system;
- Normal operation control systems;
- Miscellaneous automatic systems.



"STAR" module placement option inside an undergraound reactor shaft

8. Plant Layout Arrangement

The default design suggests deploying the reactor unit in an underground ferrocement reactor shaft to improve its containment capabilities, enhance resilience to external threats, and increase convenience of emergency water tank placement. However, other reactor building arrangements are also available. Secondary island and civil part buildings should be designed with consideration for the given country's legislation and environment.



Example layout of a reactor building with an adjacent secondary island

9. Testing Conducted for Design Verification and Validation

Most parts and materials used for the "STAR" design are tested, commercially available and have been nationally or internationally certified. During the conceptual design stage, extensive physical calculations and modelling for the reactor unit were completed using MCNP, WIMSD4, and SSL DYNCO codes. Further means of validation, including probabilistic and deterministic safety analysis tools and live test facilities for various components, shall be developed on the later design stages.

10. Design and Licensing Status

Interaction with Nuclear Safety Authority – Started; Site permit for FOK plant – To be developed.

11. Fuel Cycle Approach

The design assumes six fuel loading throughout the reactor's lifetime: the initial loading and 5 reloads every 10 years. Apart from description and requirements for refuelling equipment and procedures to be developed on later design stages, current regulations and best practices concerning handling of HALEU fuel are applicable to this design.

12. Waste Management and Disposal Plan

By design, the "STAR" reactor facility does not release liquid or solid radioactive waste during normal operation. Any solid or liquid radioactive waste collected during refuelling and maintenance shall be disposed of in accordance with applicable regulations and practices.

13. Development Milestones

2015	Initial concept developed; patent priority date secured
2019	Concept submitted for peer review

2021	Conceptual design developed; preliminary feasibility study for a FOAK unit completed
2022	Conduction of feasibility study for FOAK unit and initiation of licensing procedures with local regulator and basic design
Current (2024)	Development of basic design and PSAR, conduction of site survey
2025	Launch of FOAK plant project, development of detailed design and extended SAR
2026-2027	Pilot unit installation and launch

Rolls-Royce SMR (Rolls-Royce, United Kingdom)

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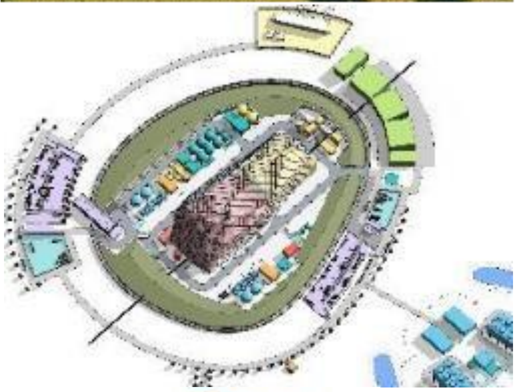


KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Rolls-Royce SMR Limited, United Kingdom
Reactor type	3-loop PWR
Coolant/moderator	Light water
Thermal/electrical capacity, MW(t)/MW(e)	1358/470
Primary circulation	Pumped at power Natural circulation for backup decay heat removal
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 7.4
Core Inlet/Outlet Coolant Temperature (°C)	295 / 322
Fuel type/assembly array	Industry standard uranium oxide fuel in 17x17 array
Number of fuel assemblies in the core	121
Fuel enrichment (%)	<4.95
Core Discharge Burnup (GWd/tHM)	~65
Refuelling Cycle (months)	18
Reactivity control	Control rods
Approach to safety systems	Passive and active
Design life (years)	60
Plant footprint (m^2)	54,500
RPV height/diameter (m)	7.82/4.20
RPV weight (metric ton)	169
Seismic Design (DBE)	0.3 g
Distinguishing features	Whole plant modular design facilitating rapid and cost-effective build.
Design status	Mature conceptual design

1. Introduction

The Rolls-Royce 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. Rapid, certain and repeatable build is enhanced through site layout optimisation and maximising modular build, standardisation and commoditisation.

2. Target Application

The Rolls-Royce SMR is primarily intended to supply baseload electricity for both coastal 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. Design Philosophy

The design philosophy for the Rolls-Royce SMR is to optimise levelised cost of electricity with a 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.

4. Main Design Features

(a) Nuclear Steam Supply System

The Rolls-Royce SMR is a three-loop Pressurised Water Reactor employing an indirect Rankine cycle. Coolant is circulated via three centrifugal Reactor Coolant Pumps to three corresponding vertical u-tube Steam Generators.

(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 1358 MW(t). Through cycle reactivity compensation is provided using gadolinia neutron poison. Different fuel assemblies contain different poison loadings, through inclusion of different numbers of poison pins and different gadolinia weight loading.

(c) Reactivity Control

Duty reactivity control is provided through movement of control rods and use of the negative moderator temperature coefficient inherent to Pressurised Water Reactors. 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 boric acid as well as the environmental impact of boron discharge. The moderator temperature coefficient is consistently high throughout the fuel cycle due to the lack of soluble boron.

(d) Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel comprises the upper shell containing the six Reactor Coolant Loop nozzles, three Direct Vessel Injection nozzles, the lower shell consisting of a plain cylindrical forging with no circumferential welds present within the high flux region of the core and the lower head. The Integrated Head Package forms an assembly of the reactor head area components. It comprises the Reactor Pressure Vessel Closure Head and outer bolting flange, Control Rod Drive Mechanisms and their cooling system, In-Core Instrumentation, integral missile shield and lifting assembly.

The lower Reactor Pressure Vessel Internals comprise the core barrel which holds the fuel and diverts coolant flow from the coolant inlets to the flow distribution device, the flow distribution device which diverts, straightens and distributes the coolant flow prior to entry into the inlet plenum and the radial neutron reflector. The upper Reactor Pressure Vessel Internals take coolant from the outlet of the fuel assemblies and transfer it to the outlets. The upper internals comprise the upper support barrel which holds the fuel in place and the control rod housing columns which align and support the control rods and control rod drive shafts.

(e) Reactor Coolant System and Steam Generator

The Reactor Coolant System contains three-loops, each with a vertical u-tube Steam Generator located around the circumference of the Reactor Pressure Vessel. The pressuriser is connected to the pipework hot leg. A centrifugal Reactor Coolant Pump is mounted directly to the outlet nozzle of each Steam Generator. Each Steam Generator is elevated above the Reactor Pressure Vessel to ensure that a sufficient thermal driving head is available for natural circulation flow for situations where pumped flow from the Reactor Coolant Pumps is unavailable. The Steam Generators include an integral crossflow preheater which works by preferentially directing feedwater to the cold side of the tube bundle, resulting in an increased thermal efficiency and smaller component compared to a non-preheater design for the same duty.

(f) 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 rapid and significant cooldown or heat-up accommodated.

(g) Primary pumps

Forced Reactor Coolant System flow is provided via three Reactor Coolant Pumps utilising a seal-less design. Pumped flow supports power operation with Reactor Coolant Pumps also configured to enable natural circulation flow during conditions where pumped flow is unavailable, ensuring the availability of passive heat rejection.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The Rolls-Royce 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. Defence in depth is provided through 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 at least 72 hours, minimising the demand on human actions and electrical power.

(b) Safety Approach and Configuration to Manage DBC

The Rolls-Royce SMR employs both active and passive decay heat removal systems. High Temperature Decay Heat Removal utilises the Steam Generators and either the normal duty steam condenser or Atmospheric Steam Dumping to cool the primary plant. It utilises much of the same equipment used for steam condensing during power operation.

Passive Decay Heat Removal is a dedicated safety measure that utilises the Steam Generators and the Local Ultimate Heatsink System elevated water store to cool the reactor. Heat rejection is closed loop between the Reactor Pressure Vessel and the Steam Generators and decay heat can be passed from the core to the Steam Generator through either pumped reactor circuit flow or natural circulation flow.

Emergency Core Cooling provides decay heat removal through depressurisation of the Reactor Coolant System and sustained supply of injected coolant by gravity feed. The function includes three accumulators connected directly to the Reactor Pressure Vessel. The Emergency Core Cooling function also places requirements on the Reactor Coolant Pressure Relief System for emergency blowdown and Local Ultimate Heatsink System to transfer heat from the Passive Containment Cooling heat exchangers to the environment.

The Rolls-Royce SMR employs both active and passive automatically initiated control of reactivity safety measures. The scram safety measure inserts solid neutron absorbers into the core and the diverse emergency boron injection system uses a pumped configuration to rapidly inject soluble potassium tetraborate into the reactor. Both systems are capable of independently providing full shutdown margin to cold zero power.

(c) Safety Approach and Configuration to Manage DEC

The RR SMR is designed to provide at least two independent and diverse safety measures for frequent initiating events (>1E-03 per year) causing fault conditions within the design basis. Diversification is provided in the mechanical, electrical and I&C systems through use of diverse equipment delivering safety functions using fundamentally different principles and phenomena. This is in keeping with UK practice and goes beyond the international practice whereby diversification is required as a design extension condition. Such conditions include complete loss of main feedwater to the steam generators and failure of alternative methods of feeding the steam generators, station blackout, anticipated transient without scram due to mechanical failure of the control rods, anticipated transient without scram due to failure of the reactor protection system, total loss of spent fuel pool cooling chain. Design extension conditions without significant fuel degradation arising from very low frequency events or complex conditions are also addressed such as main steam line break with consequential steam generator tube ruptures.

Design extension conditions with core melt are addressed through an in-vessel retention approach. The reactor pressure vessel lower head is cooled externally by gravity flooding the reactor cavity pit with water before the corium relocates to the lower head. After relocation, decay heat is then transferred from the corium mass, through the reactor pressure vessel walls, to the water. Water boils off and is vented to containment. Steam in containment is then condensed by the containment cooler and falls under gravity back to the sump. Diverse active and passive heatsinks are provided that are each independently capable of supporting in-vessel retention.

(d) Containment System

The reactor circuit and other key systems are located within a steel containment vessel to confine release of materials during faulted and accident conditions. The Rolls-Royce SMR also adopts in-vessel retention to confine the postulated melt in severe accidents.

(e) Spent Fuel Cooling Safety Approach / System

Active and passive cooling systems provide robust decay heat removal defence in depth during fault conditions. Reactivity control is provided in a soluble boron-free environment using geometric spacing and solid neutron poison racks.

6. Plant Safety and Operational Performances

Plant conditions have been analysed using industry validated codes to demonstrate significant safety margin across the levels of defence in depth. The Probabilistic Safety Assessment calculates a core damage frequency from plant faults of <1E-07 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 to optimise segregation. Key equipment is protected by the hazard shield which is resilient against external hazards including aircraft impact and tsunami.

The Rolls-Royce SMR is designed for full compliance with the U.K. Grid Code, provides load following between 50% and 100% power at a rate of 3-5% per minute, and can deliver stable operation for at least two hours supplying only the power station's house load.

An 18-day outage period is targeted to support high availability, at least 92.5% availability over the 60-year life of the Rolls-Royce SMR.

7. Instrumentation and Control Systems

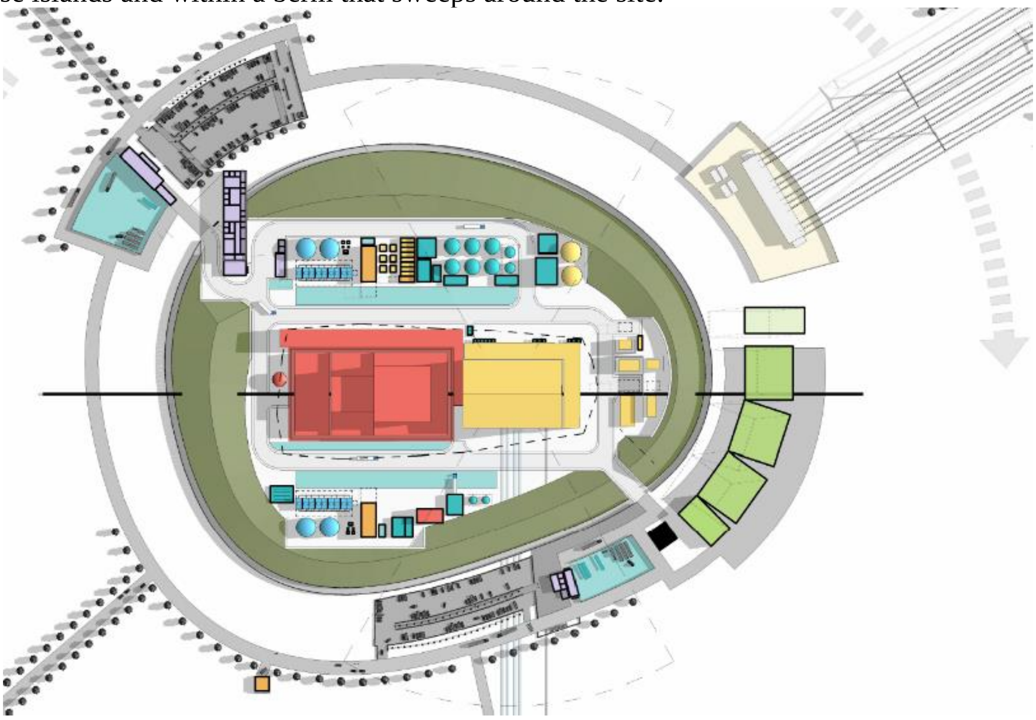
The Rolls-Royce SMR reactor plant is controlled and protected by a number of control and instrumentation systems. The Reactor Plant Control System manages duty operations and uses an available-in-industry programmable logic controller or distributed control system. It uses mixed analogue and non-programmable 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 provides shutdown, decay heat removal and containment in response to a fault. The Reactor Protection System contains priority logic, which from the range of input signals received from lower classified systems, determines the safe operation of actuators. The Diverse Protection System uses non-programmable electronics and as such provides a diverse means to provide shutdown, decay heat removal and containment in response to fault conditions. The Diverse Protection System contains priority logic, independent of Reactor Protection Systems which, from the range of input signals received from lower classified systems, determines the safe operation of actuators.

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.

8. Plant Layout Arrangement

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. This flexibility is enabled through design features such as seismic isolation for key safety areas. The nuclear reactor is located in the Reactor Island (red) adjacent to Turbine Island (yellow). Support buildings and auxiliary services are situated around these islands and within a berm that sweeps around the site.



9. Testing Conducted for Design Verification and Validation

The Rolls-Royce SMR design is based on optimised and enhanced use of proven technologies. Test rigs have been designed for validation of safety claims of the key safety systems, utilising established test facilities reflecting that safety systems are based on readily provable phenomena.

10. Design and Licensing Status

The Rolls-Royce SMR entered formal design assessment by UK regulators in 2022 and have successfully completed two steps of the three-step regulatory process. Completion is targeted in time for construction of the first of a kind power station to commence in 2027.

11. Fuel Cycle Approach

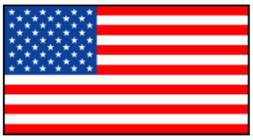
Approaching equilibrium, the Rolls-Royce SMR operates on an 18-month fuel cycle, with a three-batch equilibrium core. Used fuel is subsequently transferred to a spent fuel pool adjacent to the containment building for storage prior to transfer to long term dry cask storage.

12. Waste Management and Disposal Plan

The Rolls-Royce 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.

13. Development Milestones

2015	Development of initial reference design	Complete
2017	Development of whole power station conceptual design	Complete
2022	Formal regulation entered in the UK	Complete
2027	Construction of first plant begins	Planned
Early 2030s	Commercial operation of first plant	Planned



SMR-300 (Holtec International, United States of America)

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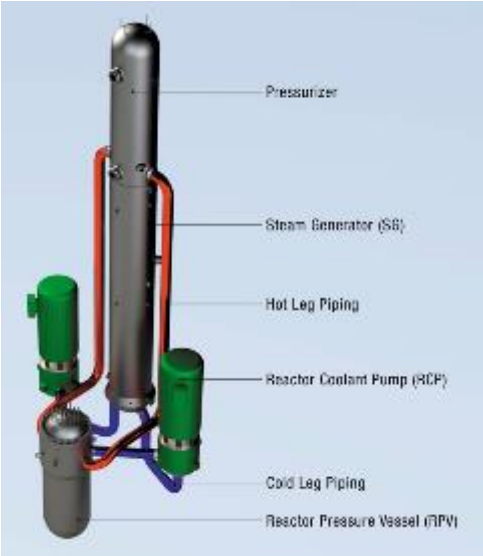
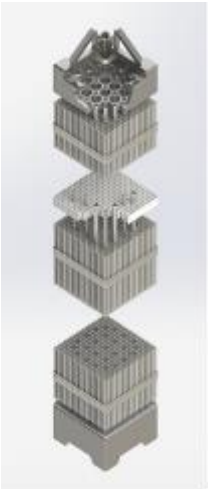


SMR-300 Primary Circuit



SMR-300 Fuel Assembly

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Holtec International, United States of America
Reactor type	PWR
Coolant/moderator	Light water/light water
Thermal/electrical capacity, MW(t)/MW(e)	1050 MWth (nominal) 366 MWe (gross) /320 MWe (net)
Primary circulation	Forced (2 pumps)
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 6.2
Core Inlet/Outlet Coolant Temperature (°C)	292 / 321
Fuel type/assembly array	UO_2 pellet / square array
Number of fuel assemblies in the core	69
Fuel enrichment (%)	4.9% (average) / 5% (max)
Core Discharge Burnup (GWd/ton)	43 (initial design)
Refuelling Cycle (months)	18
Reactivity control	Soluble Boron and Rods (RCCAs)
Approach to safety systems	Incorporates robust passive safety systems, requiring no operator action for accident mitigation.
Design life (years)	80
Plant footprint (m^2)	160,000 (dual unit, fenced area)
RPV height/diameter (m)	TBD
RPV weight (metric ton)	TBD
Seismic Design (SSE)	0.4
Distinguishing features	Passive safety cooling systems and active non-safety systems; flexible; spent fuel pool within containment (no transfer channel); integrated dry spent fuel storage and transportation system; smart modular design concept
Design status	Detailed Design

1. Introduction

The SMR-300, developed by Holtec International, is a Small Modular Reactor (SMR) designed to deliver efficient and reliable power generation. It features advanced Pressurized Water Reactor (PWR) technology in a compact and scalable design. This reactor aims to provide a flexible solution for various applications, from remote locations to integrated power grids. Its innovative safety and operational features set it apart from traditional large-scale reactors.

2. Target Application

The primary application of SMR-300 is electricity production with optional cogeneration equipment (i.e., hydrogen generation, thermal energy storage, district heating, seawater desalination).

The SMR-300 is ideal for deployment in areas where large nuclear reactors are impractical due to space or logistical constraints. Its modular design allows for easy installation in remote or industrial sites and integration into existing power infrastructure. The reactor's flexibility makes it suitable for both standalone operations and as part of larger energy systems. This adaptability supports diverse energy needs and locations with varying requirements.

3. Design Philosophy

The design philosophy of the SMR-300 prioritizes safety and simplicity by incorporating advanced passive safety systems that require no operator intervention. The reactor's streamlined design reduces complexity, maintenance, and operational costs while enhancing reliability. It integrates proven technologies and materials to ensure robust performance under various conditions. This approach ensures that the reactor operates efficiently and securely with minimal operational challenges.

4. Main Design Features

(a) Nuclear Steam Supply System

The SMR-300 utilizes a two-loop Pressurized Water Reactor (PWR) system with a single once-through steam generator. This configuration enhances thermal efficiency and power generation. The steam generator efficiently transfers heat from the reactor coolant to produce high-pressure steam. Integrated design simplifies system operations and improves overall reliability.

(b) Reactor Core

The reactor core of the SMR-300 features a 17x17 array of low-enriched uranium dioxide fuel assemblies. Designed for an 18-month refueling cycle, it maximizes fuel efficiency and operational lifespan. The core's configuration supports robust performance and safety margins. Available and proven PWR technology and fuel ensures reliable and effective core operation.

(c) Reactivity Control

Reactivity control in the SMR-300 is achieved through the use of control rods and/or boron in the reactor coolant. These mechanisms allow precise adjustment of the reactor's power output and maintain stable operation. The control system is designed to handle various operational and safety scenarios. This approach ensures effective management of reactor reactivity throughout its operating cycle.

(d) Reactor Pressure Vessel and Internals

The reactor pressure vessel is designed to be compact and robust, housing the core and primary coolant system. It features durable materials to withstand high pressures and temperatures. The internal configuration minimizes complexity and supports efficient maintenance. The design enhances the overall safety and longevity of the reactor.

(e) Reactor Coolant System and Steam Generator

The reactor coolant system employs forced circulation during normal operations with two cold-leg pumps and a single once-through steam generator. This system ensures efficient heat removal and transfer to the steam generator. The integrated design of the steam generator and pressurizer improves thermal management. The system's simplicity contributes to reduced maintenance and increased operational efficiency. The SMR-300 safety systems are passive and are driven by natural forces (e.g., gravity, conductive, and convective heat transfer), with no reliance on pumps, external water, or offsite power.

(f) Pressuriser

The pressurizer is integrated into the steam generator, maintaining the pressure of the reactor coolant system. This design reduces system complexity and enhances reliability. It regulates the coolant pressure to prevent boiling and ensure stable reactor conditions. The integration supports efficient thermal and pressure management within the reactor system.

(g) Primary pumps

The SMR-300 uses two vertically mounted reactor coolant pumps to ensure forced circulation within the primary loop. These pumps are designed for high reliability and efficiency in circulating coolant through the reactor core and steam generator. Their configuration supports consistent heat removal and system stability. The design minimizes the need for additional components and maintenance.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The SMR-300 employs advanced passive safety systems designed to function without operator intervention or external power sources. The safety systems include redundant and diverse pathways for heat removal and accident mitigation. These systems are integrated into the reactor's robust containment structure, enhancing overall safety. The design minimizes the likelihood of accidents and ensures effective response in emergency scenarios.

(b) Safety Approach and Configuration to Manage DBC

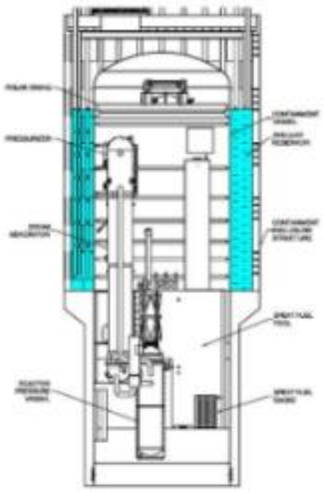
For Design Basis Events (DBCs), the SMR-300 incorporates redundant safety features that automatically manage and mitigate potential accidents. The reactor's passive safety systems provide continuous cooling and containment without requiring operator action. This approach ensures that the reactor remains safe under a wide range of anticipated operational occurrences. Enhanced safety margins are maintained to protect the plant, the public health and safety, the environment, and the health and safety of plant workers.

(c) Safety Approach and Configuration to Manage DEC

The reactor's safety design addresses Design Extension Conditions (DECs) by providing long-term, passive cooling solutions and containment integrity. The SMR-300 includes multiple safety barriers and systems to handle severe accident conditions. These systems are designed to maintain core cooling and prevent radiological release. The reactor's ability to self-cool and manage heat post-accident enhances its resilience to extreme scenarios.

(d) Containment System

The SMR-300 features a robust containment system that integrates a large water reservoir for heat dissipation and missile protection. This containment structure is designed to withstand external impacts and prevent the release of radioactive materials. The water reservoir aids in cooling and managing decay heat during accidents. The containment system enhances overall plant safety and environmental protection.



SMR-300 Configuration

(e) Spent Fuel Cooling Safety Approach / System

Spent fuel cooling is managed through a small spent fuel pool located within the reactor containment, providing initial cooling and radiation shielding. Long-term cooling is achieved using dry cask storage, which secures spent fuel in welded stainless-steel canisters. This approach ensures safe handling and minimal environmental impact. The spent fuel management system supports efficient and secure waste storage throughout the reactor's operational life.

6. Plant Safety and Operational Performances

The SMR-300 is designed for high safety and operational performance, featuring robust passive safety systems and simplified maintenance procedures. The reactor's design ensures minimal operator intervention while maintaining high reliability and efficiency. Safety margins are maximized through the use of proven materials and advanced technologies. This approach enhances the reactor's ability to operate safely and effectively under a range of conditions.

7. Instrumentation and Control Systems

The SMR-300 utilizes the MELTAC Nplus S digital safety I&C platform to enhance plant control and safety. This system provides advanced monitoring and control capabilities, optimizing operator interface and defense-in-depth. The platform ensures robust performance and reliability in managing plant operations. It integrates with the reactor's safety features to support secure and efficient operation.

8. Plant Layout Arrangement

The SMR-300 features a compact and efficient plant layout designed to maximize space utilization and simplify construction. Major components, including the reactor core, steam generator, and safety systems, are arranged to facilitate easy maintenance and operational efficiency. The layout supports streamlined operations and reduces the need for extensive infrastructure. Diagrams and layouts illustrate the reactor's efficient use of space and integration of key systems.

9. Testing Conducted for Design Verification and Validation

Extensive testing and simulations are conducted to verify and validate the SMR-300's design, ensuring it meets safety and performance criteria. This includes evaluating safety systems, core performance, and operational reliability under various conditions. The testing process confirms the reactor's ability to operate safely and efficiently. Results from these tests support the reactor's readiness for deployment and operation.

10. Design and Licensing Status

The SMR-300 is currently in the Detailed Design phase, with ongoing efforts to finalize the regulatory submissions. Holtec International is preparing for site characterization and regulatory reviews, with construction planned to start by mid-2027. The reactor is expected to enter commercial operation in 2030. This timeline reflects progress towards a fully licensed and operational reactor.

11. Fuel Cycle Approach

The SMR-300 employs a low-enriched oxide fuel cycle with a traditional 18-month reload shuffle. This approach optimizes fuel usage and extends the reactor's operational life. The fuel cycle is designed to maintain high performance and efficiency throughout the reactor's lifespan. The use of proven fuel technology supports reliable and effective operation.

12. Waste Management and Disposal Plan

The SMR-300's waste management strategy includes a small spent fuel pool within containment for initial cooling and long-term dry cask storage for spent fuel. This plan ensures secure handling and minimal environmental impact. The system incorporates rigorous security measures and monitoring to maintain safe waste management. The approach supports effective disposal and long-term safety.

13. Development Milestones

February 2024	SMR-300 FOAK site selection (Palisades site in MI, USA)	Completed
October 2025	Integral and Separate Effects Testing (ISET) Program completed at Idaho National Labs (INL), USA	Planned
June 2026	Submit Construction Permit Application for Palisades site	Planned
June 2027	Commence Nuclear Island Construction Activities at the Palisades site	Planned
June 2030	Completion of Construction	Planned
December 2030	Commercial Operation Date (COD)	Planned

WATER COOLED SMALL MODULAR REACTORS (MARINE BASED)

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

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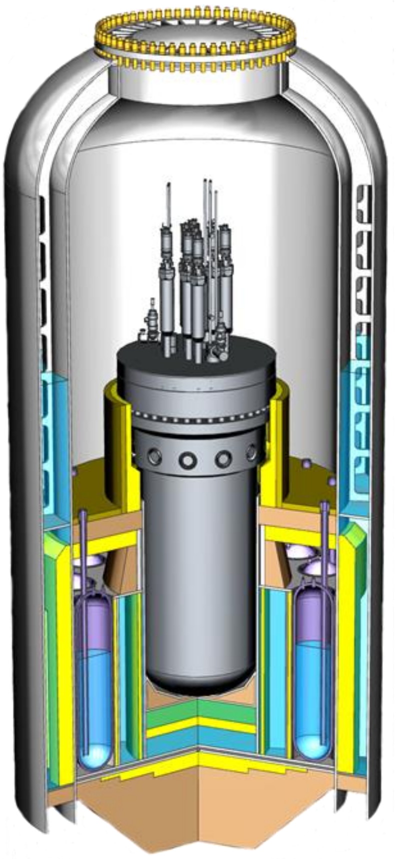


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, Rosatom, Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	38 / 6 – 9
Primary circulation	Natural circulation
NSSS Operating 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.

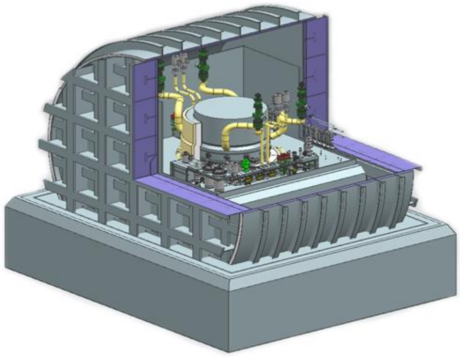
2. Target Application

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. Design Philosophy

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

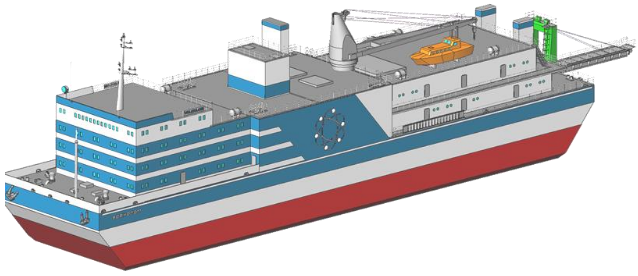
- Integral primary circuit layout with natural circulation of the primary coolant;
- Negative feedbacks and enhanced thermal inertia;
- Passive and self-actuated safety systems;
- Increased resistance to extreme external events and personnel errors;
- Use of nuclear fuel with the enrichment of less than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

4. Main Design Features

(a) 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 ^{235}U. Special stainless steel is used as fuel cladding.

(b) 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).

(c) 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.

(d) Reactor Coolant System

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

5. Safety Features

(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.

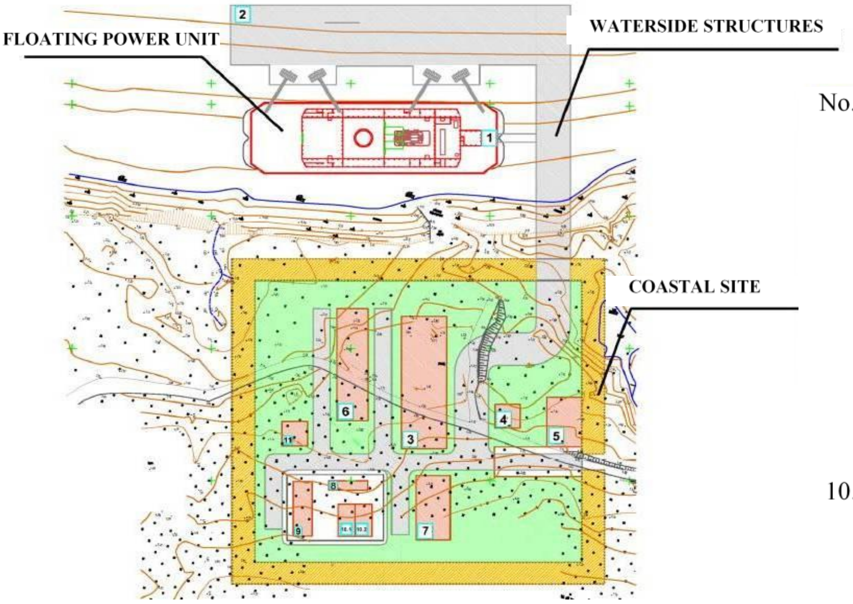
(d) Containment System

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.

6. Plant Safety and Operational Performances

The NPP with ABV-6E generates electricity and heat in the power range of 20–100%N_{nom} with the continuous operation time of 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.

7. Plant Layout Arrangement



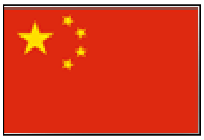
No. in plan	Title
1	Floating power unit
2	Waterside structures
3	Integrated process building
4	Treatment facilities
5	Access control point
6	Guard room
7	Administrative building/shelter
8	Indoor switchgear
9	Substation-shared control room
10.1, 10.2	Diesel-generator sets
11	Garage

8. Design and Licensing Status

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

9. 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)



ACP100S (CNNC, China)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNNC/NPIC, 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 (°C)	286.5 / 319.5
Fuel type/assembly array	UO_2 / 17×17 square pitch arrangement
Number of fuel assemblies in the core	57
Fuel enrichment (%)	< 4.95
Refuelling cycle (months)	24
Core discharge burnup (GWd/ton)	< 52
Reactivity control mechanism	Control rod drive mechanism (CRDM), Gd_2O_3 solid burnable poison and soluble boron acid
Approach to safety systems	Passive + Active
Design life (years)	40
Plant footprint (m^2)	8 000 (Floating NPP)
RPV height/diameter (m)	10 / 3.35
RPV weight (metric ton)	300
Seismic design (SSE)	Not applicable
Fuel cycle requirements/approach	Shore refuelling, temporally stored in the spent fuel pool
Distinguishing features	Integrated reactor with once through steam generator, towed to sites along coast
Design status	Basic design

1. Introduction

ACP100S was developed based on its land-based reactor ACP100 whose pilot project has been under construction in Hainan province in China to provide capacity of 125 MW(e) per module. It inherits most of the main features of ACP100 with integrated design and could be used for power generation, desalination, heating, for offshore and open sea area, and isolated island, etc. It mainly adopts passive safety system with active passive measure as an implement to cope with the marine environment. It could be constructed in the shipyard and delivered to the sites, and satisfy the safety requirement of third generation PWR.

2. Target Application

The ACP100S is a multipurpose power reactor designed for coastal and open sea areas, or isolated island where central electric grid is hard to reach. It could be used for electricity production, heating, steam production or seawater desalination.

3. Design Philosophy

The ACP100S realizes design simplification by integrating the primary cooling system and enhanced safety by using the cooling water from the sea. It adopts mature design philosophy with land-based ACP100 being constructed. ACP100S could be constructed in the shipyard and factory, which greatly improve the economic competitiveness.

4. Main Design Features

(a) 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. All these technologies provide high inherent safety and avoid large LOCA in large-size nuclear power plant.

(b) Reactor Core

The 57 fuel assemblies (FAs) of ACP100S core with total length of 2.15 m core have a squared 17×17 configuration. The fuel ^{235}U enrichment is about 1.9 – 4.95%. The reactor will be able to operate 24 months at balance fuel cycle.

(c) 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).

(d) 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.

(e) Reactor Coolant System

The ACP100S 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.

(f) Steam Generator

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

(g) Pressurizer

The pressurizer of ACP100S is located outside of the reactor vessel. The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads.

(h) Primary pumps

The ACP100S uses canned-motor pumps as reactor coolant pumps which are directly mounted on the RPV nozzle. The shaft of the impeller and rotor of the canned-motor pump is contained in the pressure boundary, eliminating the seal LOCA of reactor coolant pump.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The ACP100S is designed with inherent features, eliminating large primary coolant piping which in turns eliminates large break LOCA. It mainly adopts passive safety system with active passive measure as a supplement to cope with the marine environment. The safety system mainly consists of the passive decay heat removal system (PDHRS), passive emergency core cooling system (ECCS), passive containment suppression system (PCS) and active containment spray system.

(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 ACP100S consists of two residual remove exchangers and associated valves, piping, and instrumentation. The residual heat remove exchanger is located in the passive residual heat remove water storage tank *mounted* on left and right side of shipboard, 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 72 hours without operator intervention.

(c) Emergency Core Cooling System

The emergency core cooling system (ECCS) consists of two safety injection system and recirculation system, which mainly include two core make-up tanks (CMT), two low head pressure injection pumps and associated injection lines and valves. In the DBA, cooling water from the CMTs would be injected to the reactor core by gravity force to cool down the reactor core, and recirculation would be generated for long term after the DBA accident.

(d) Spent Fuel Cooling Safety Approach / System

ACP100S adopts the mature shore refuelling strategy and would be towed to the dock when refuelling is needed after one refuelling period of 24 month.

(e) Containment System

The ACP100S use small steel containment containing one module of the NSSS. It's a vertical, cylindrical vessel with hemispherical top and bottom heads, a personnel gate is mounted on the cylindrical side and a hemishperical top gate is set as a refuelling gate. The containment is figured in the reactor compartment which is a reinforced structure of the boat. The containment could withstand the maximum pressure during DBA of NSSS, and also withstand the external water pressure in the sink accident.

6. 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 ACP100S 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 ACP100S incorporates operational experience of the state-of-the-art design. Proven technology and equipment are adopted as much as reasonably possible. It's estimated that ACP100S has the core damage frequency (CDF) of less than 1×10^{-5} per reactor year and the large release frequency (LRF) of 1×10^{-6} per reactor year. The load factor is no less than 0.9.

7. Instrumentation and Control System

The Instrumentation and Control (I&C) system designed for ACP100S 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.

8. Plant Layout Arrangement

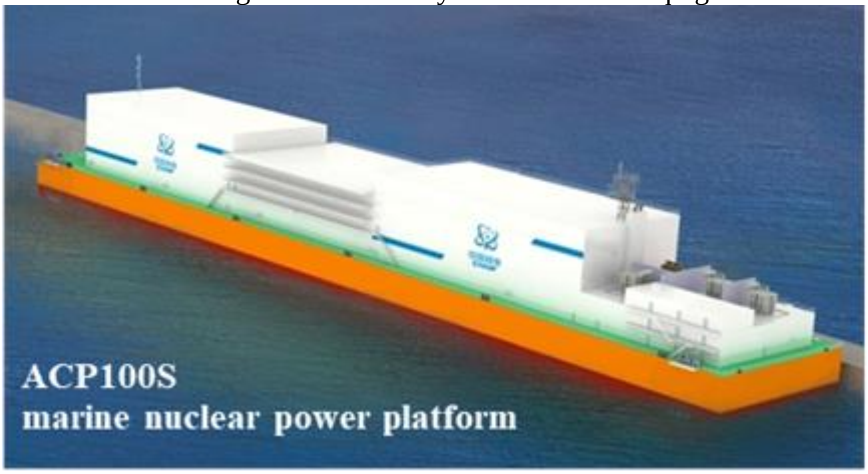
The ACP100S is a non-propelled FNPP mounted on the boat, and could be towed to locations where needed with the ability of operating independently. Actually, one or two NSSS modules could be installed on the boat according to the requirement, with the length of the boat about 200meters or 150 meters respectively. The containment figured in the reactor compartment is located in the middle of the boat to lower the effect caused by the ocean. The turbogenerator compartment is located next to the reactor compartment, and accommodations are arranged in astern. The ACP100S has enough space to figure reactor auxiliary compartment and I&C room, the cogeneration facility could be installed on the board or at the coastline according to the operation requirement.

9. Testing Conducted for Design Verification and Validation

Since ACP100S is developed based on mature ACP100， most of verification tests have been finished, such as CHF, test relevant with SG, flow induced vibration of internals, etc. Experiments are mainly planned to verify the new design brought by the ocean environment. For example, the control rod drive line cold and hot test, containment scale structure test, and containment suppression test are in progress. ACP100S has a mature NSSS with only small part of verification needed.

10. Design and Licensing Status

Presently, the licensing of ACP100S is yet to be launched. The preliminary feasibility study of ACP100S has been finished aiming at the site in city of Yantai and Fuqing in the East of China.



11. Fuel Cycle Approach

Spent fuel processing is conducted on the shore, using the service facility prepared at the base harbour, which is similar to the land-based reactor.

12. Waste Management and Disposal Plan

After waste is transported to the base harbour, the waste management approach and disposal plan is similar to other nuclear power plants.

13. Development Milestones

2012	The concept design of ACP100S finished.
2018	The basic design of ACP100S finished.
2019	Preliminary feasibility study for one site finished.
2021	Preliminary design of ACP100S finished.
2023	The demonstration of marine nuclear power finished.
2024	The feasibility study of demonstration project finished
2026	Target to start construction

BANDI (KEPCO E&C, Republic of Korea)

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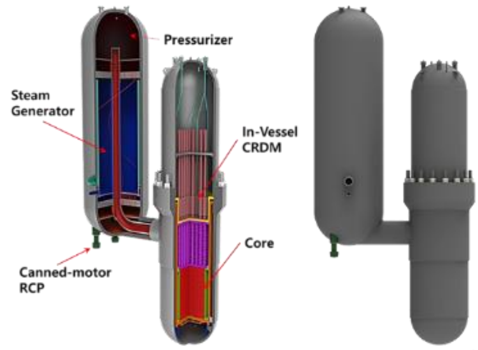


RCS Configuration and Cross-Sectional View

A Typical Plant Layout of BANDI

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	KEPCO E&C, Republic of Korea
Reactor type	PWR
Coolant/moderator	Light Water (H_2O)
Thermal/electrical capacity, MW(t)/MW(e)	200 MW(t) / 60 MW(e)
Primary circulation	Forced (2 pumps)
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 5.5
Core Inlet/Outlet Coolant Temperature (°C)	293 / 322
Fuel type/assembly array	UO_2 ceramic fuel, 17x17 square array
Number of fuel assemblies in the core	52
Fuel enrichment (%)	4.95
Core Discharge Burnup (GWd/ton)	29.4
Refuelling Cycle (months)	48 - 60
Reactivity control	Control rods, Temperature
Approach to safety systems	Passive safety systems including PECCS, PRHRS, and PCCS
Design life (years)	60
Plant footprint (m^2)	TBD (To be determined)
RPV height/diameter (m)	13.6 / 2.8 – 3.4
RPV weight (metric ton)	110
Seismic Design (SSE)	N/A
Distinguishing features	Semi-integral design, block-type RCS, and passive safety systems
Design status	Conceptual Design

1. Introduction

BANDI, developed by KEPCO E&C, is named after a Korean firefly known for its clean image of emitting bright green light in a deep forest in a summer night. Its reactor system features a semi-integral or a 'block-type' design, with a steam generator (SG) block and a reactor vessel (RV) block directly connected without large pipes. It does not use soluble boron for its core reactivity control. Its Control Rod Drive Mechanism (CRDM) is installed inside the RV block, called In-Vessel CRDM or IV-CRDM.

2. Target Application

The primary target of BANDI is floating nuclear power plants that provide clean, safe, reliable and affordable energy for remote isolated communities like a firefly brightens in a deep, dark forest. It can also be designed for carbon-free propulsion power of big merchant ships like bulky containers. KEPCO E&C is now working closely with a Korean global leading shipbuilding company on engineering and business planning.

3. Design Philosophy

The design of BANDI is based on KEPCO E&C's technology and experience proven through conventional nuclear power engineering over 40 years. Nuclear safety functions of reactor shutdown, decay heat removal and radiation protection are further enhanced by inherent and passive safety features. New technologies like advanced nuclear fuel, compact and simple I&C systems, automatic load following, autonomous operation, etc. will also be added.

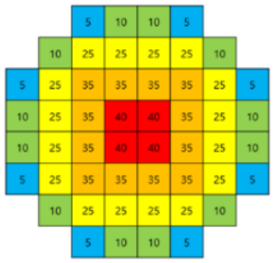
4. Main Design Features

(a) Nuclear Steam Supply System

BANDI has a single-loop design of Reactor Coolant System, featuring a semi-integral or a block-type arrangement where the Reactor Vessel (RV) block and Steam Generator (SG) block are directly connected by nozzles. The nozzle-to-nozzle connection is designed in accordance with ASME BPVC Sec. III, Class 1 NB-3300. With it, a large pipe break LOCA, one of the most serious accident scenarios, can be eliminated from the design basis. Compared to fully integral systems, the block-type offers easier maintenance and better scalability. Soluble boron-free design eliminates troublesome concerns caused by the corrosive boron as experienced in conventional PWR plants. This significantly simplifies the Chemical and Volume Control System (CVCS) and reduces the footprint of plant. It also alleviates corrosive environments and reduces liquid radioactive wastes.

(b) Reactor Core

The BANDI reactor core uses conventional UO_2 ceramic fuel enriched up to 4.95% in a 17×17 array and the active core height is 2m. It consists of 52 fuel assemblies designed to produce 200 MW thermal power over 4~5 years. The core incorporates burnable absorbers, specifically Pyrex (B_2O_3) rods, which replace 24 fuel rods per assembly and vary in boron concentration from 5% to 40%. The core's configuration allows for a 4.79-year operation cycle at full power.



Reactor Core with Burnable Poison

(c) Reactivity Control

BANDI is soluble boron-free for its core reactivity control. The negative moderator temperature coefficient (MTC) is increased in magnitude and thus enhances the inherent safety and stability of reactivity control. Its reactor core employs 40 control rod assemblies, each containing 24 B_4C control rods, grouped into two types: 18 regulating and 22 shutdown rods. These are further organized into four banks for regulation and two banks for shutdown. The use of burnable absorbers, specifically Pyrex (B_2O_3) rods, helps manage excess reactivity throughout the core's life. This setup ensures adequate negative reactivity to maintain shutdown capability without soluble boron, and supports the reactor's operational efficiency and safety.

(d) Reactor Vessel Block

BANDI reactor vessel (RV) block consists of a reactor pressure vessel (RPV) and the internals. BANDI RPV is a vertically mounted cylindrical structure with a hemi-spherical lower head and a removable top cover, designed to avoid low head penetrations by using Top-Mounted In-Core Instrumentation (TM-ICI). This design minimizes the risk of lower vessel failure even in a severe core meltdown accident.

Control Rod Drive Mechanism (CRDM) is installed inside the RV block and called In-vessel CRDM. It eliminates the reactivity accident of rod ejection from the design basis, which otherwise could be more serious for small boron-free reactors like BANDI than bigger conventional ones since the reactivity worth of individual control rod of small reactors is stronger.

(e) Steam Generator Block

The SG block contains one Steam Generator, one Pressurizer and two canned-motor Reactor Coolant Pumps in it. The SG is once-through type with helical heat transfer tubes and produces superheated steam. The reactor coolant flows through the tube side, and the secondary steam and feedwater through the shell side.

The pressurizer is integrated into the SG block's upper section, where it manages system pressure through heaters and sprays. It is designed with a large capacity to control reactor coolant system volume across the full range of reactor power levels. Safety Depressurization Valves (SDVs) are included on the top of the pressurizer to handle pressure equilibrium during small break loss of coolant accidents. BANDI reactor uses two leak-tight canned-motor pumps (RCPs) located in the SG block's bottom plenum. These pumps are designed for reliability and efficient coolant circulation, ensuring stable heat transfer and system operation. Their sealed design minimizes maintenance needs and enhances operational safety.

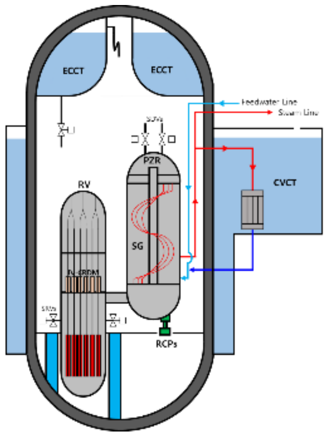
5. Safety Features

(a) Engineered Safety System Approach and Configuration

BANDI reactor uses a Passive Emergency Core Cooling System (PECCS) with Emergency Core Cooling Tank (ECCT), Safety Depressurization Valves (SDVs), and Safety Recirculation Valves (SRVs) for emergency coolant injection, without reliance on active safety systems. It integrates passive safety features like gravity-driven safety injection and natural circulation cooling to ensure reactor safety under any accident conditions.

(b) Safety Approach and Configuration to Manage DBC

BANDI's safety approach for Design Basis Conditions (DBC) includes a block-type reactor design with a single heat transfer loop to minimize LOCA risks. It features a robust Reactor Pressure Vessel and a helical tube Steam Generator, along with Safety Depressurization Valves (SDVs), and Safety Recirculation Valves (SRVs) for emergency cooling. The reactor's design includes redundant cooling and pressure control systems, along with advanced instrumentation for early warning and response during extreme conditions.



Safety Systems of BANDI

(c) Safety Approach and Configuration to Manage DEC

For Design Extension Conditions (DEC), BANDI uses passive safety systems like the Passive Emergency Core Cooling System (PECCS) and Containment Vessel Cooling Tank (CVCT) to handle severe accidents without external power.

(d) Containment System

The reactor containment system is composed of two main structures - Containment Vessel (CV) and Reactor Compartment. The CV of BANDI is a steel cylindrical type and its principal function is to confine radioactive materials that can be released during both normal and accident conditions including very unlikely severe accident. The reactor compartment encloses the whole primary systems and protects them from external hazards such as airplane crash.

(e) Spent Fuel Cooling Safety Approach / System

The spent fuel cooling system of BANDI includes a spent fuel pool designed to store spent fuel assemblies for up to 10 years, with a dedicated heat removal system and chemistry control. The pool is maintained by a heat removal system to ensure safe decay heat removal management and is used for both new and spent fuel during refuelling outages. Fuel assemblies are moved to and from the pool using a fuel transfer system, with the process completed in about 10 days.

6. Plant Safety and Operational Performances

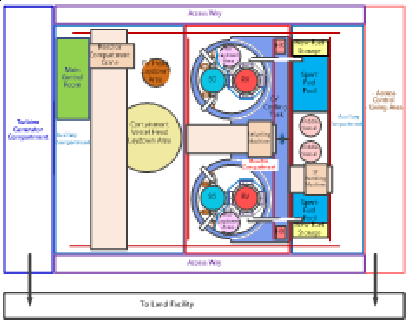
BANDI reactor emphasizes safety with passive systems for emergency cooling and decay heat removal. Operational efficiency is enhanced by the born-free reactivity control design fully utilizing the enhanced MTC feedback effect and the Reactor Trip Prevention System (RTPS) to prevent unintended shutdowns.

7. Instrumentation and Control Systems

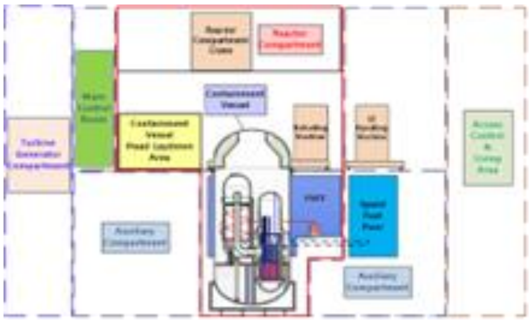
The Instrumentation and Control (I&C) system is a fully digital system with data communication network and has a simple and compact structure suitable for BANDI. I&C design of BANDI prevents software Common Cause Failure (CCF) by implementing its own diversity design in the safety system.

8. Plant Layout Arrangement

The plant layout is designed for optimal safety and efficiency, with clear zoning for reactor operations, support systems, and emergency facilities. In a dual-unit setup of reactor systems in the floating nuclear power plant, it features two separate Containment Vessels (CVs) with individual reactor systems to reduce accident impacts. The auxiliary compartment safeguards safety-related systems from natural hazards with protective measures, while the turbine compartment houses Turbine Generators, condensers, and heaters.



Plane View of Plant Layout



Section View of Plant Layout

9. Testing Conducted for Design Verification and Validation

BANDI has verified its advanced design feature of the In-Vessel Control Rod Drive Mechanism (IV-CRDM) through full-scale testing to ensure its reliability under the normal operating conditions of high pressures and temperatures.

In-Vessel CRDM Demonstration Test Facility

10. Design and Licensing Status

BANDI is in the second phase of its conceptual design, incorporating 45 years of PWR experience and recent innovations. Major design updates were made in 2023, with further refinements planned for 2024. Licensing reviews are planned for the late 2020s, with construction of the first unit expected to begin after 2030.

11. Fuel Cycle Approach

BANDI supports a single batch refuelling every 4-5 years to minimize downtime, or multi-batch cycles for cost optimization. New fuels can be transported with modified casks, while spent fuel can be stored on-site for up to 5~10 years before being shipped to a dedicated facility. Fuel cycle options can vary based on owner preferences.

12. Waste Management and Disposal Plan

BANDI's waste management strategy involves temporary on-site storage of radioactive waste before it is transferred to long-term disposal facilities. Low- and intermediate-level wastes are safely contained, while high-level waste is managed with enhanced safety measures. Spent fuel is stored on-site for approximately 5~10 years before being shipped in dry casks to dedicated storage facilities, with options for future disposal or reprocessing depending on owner preferences.

13. Development Milestones

2012 – 2015	Preliminary and basic studies for SMRs	Complete
2013 – 2018	Government-supported R&D projects for key innovative element technologies of SMR	Complete

2016 – 2022	Nuclear Steam Supply System (NSSS) conceptual design for maritime SMRs	Complete
2023	Overall design change	Complete
2023 – 2024	NSSS, Balance of Plant (BOP), and floating system conceptual design for FNPPs	On track
2025 – 2026	Basic design for FNPPs	Planned
2026 - 2030	Standard design and licensing review for FNPPs	Planned
2031 ~	Projected construction	

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

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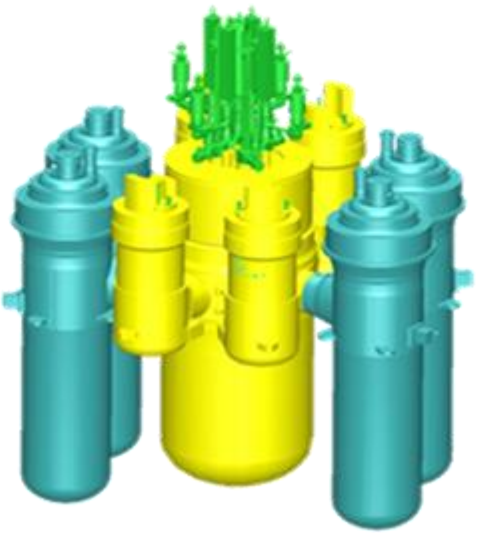


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, Rosatom, Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	150 / 35
Primary circulation	Forced circulation
NSSS Operating 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. 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.

4. Main Design Features

(a) 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.

(b) 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.

(c) 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.

(d) 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.

(e) 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).

5. Safety Features

The KLT-40S is designed with proven safety aspects such as a compact structure of the SG unit with short nozzles connecting the main equipment, primary circuit pipelines with smaller diameter, and with proven reactor emergency shutdown actuators based on different operation principles, emergency heat removal systems connected to the primary and secondary circuits. Additional barriers are provided to prevent the release of radioactivity from the FNPP caused by severe accidents. Among them are passive and active physically separated and independent safety systems, I&C systems, diagnostic systems, active cooling train through primary circuit purification system's heat exchanger thermally coupled with a 'third' independent circuit exchanging heat energy with ambient sea or lake water, active cooling train through the SGs heat exchangers with decay heat removal accomplished through the condenser which in turn is cooled down by ambient sea or lake water, 2 passive cooling trains through the SGs with decay heat removal via emergency water tank heat exchangers, and venting to atmosphere by evaporation from said tanks. Both active and passive safety systems are to perform the reactor emergency shutdown, emergency heat removal from the primary circuit, emergency core cooling and radioactive products confinement. The KLT-40S safety concept encompasses accident prevention and mitigation system, a physical barriers system, and a system of technical and organizational measures on protection of the barriers and retaining their effectiveness, in conjunction with measures on protection of the personnel, population and environment. The KLT-40S safety systems installed on FNPPs are distinctive from those applied to land-based installations in security of the water areas surrounding the FNPP, anti-flooding features, anti-collision protection, 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

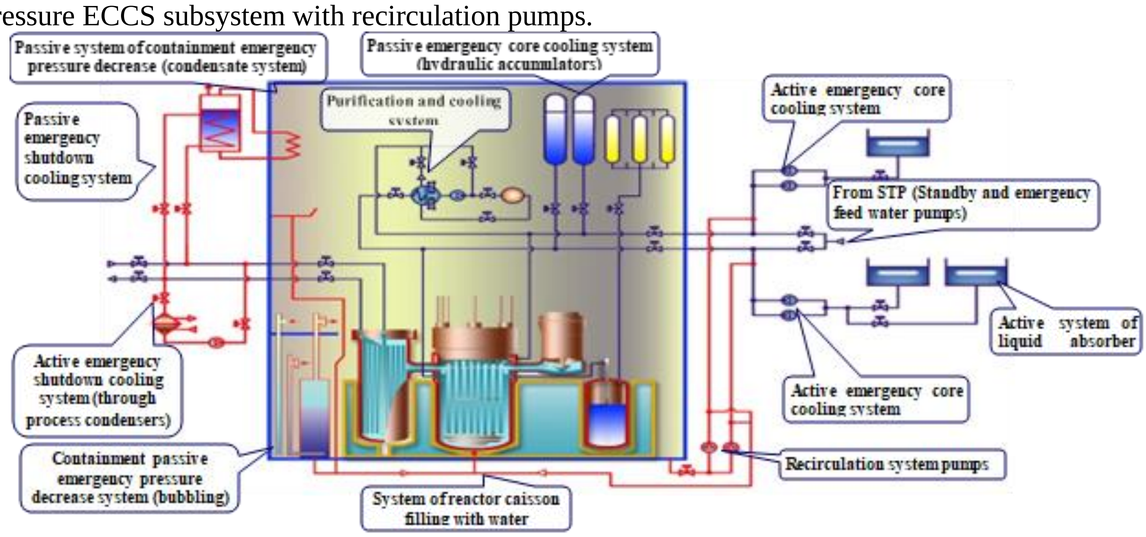
The active components of the protection system are scram actuators for six (6) groups of the control rods.

(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.

6. 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.

7. 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.

8. Plant Layout Arrangement

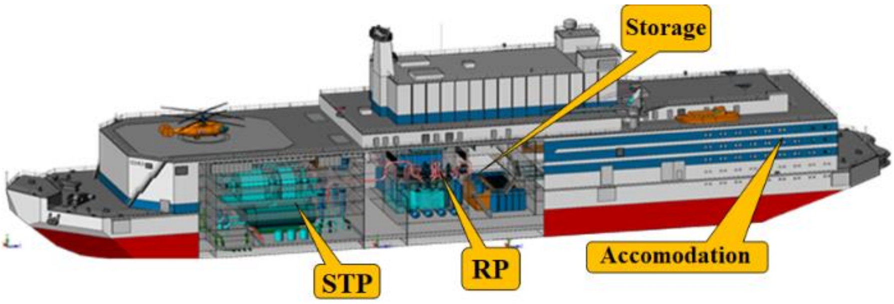
The coastline line of the FCNPP has the complex engineering building with equipment to distribute and transmit electricity to loads and to prepare and transfer heating water to loads and auxiliary buildings, including: two hot water storage tanks; partially in ground tank with slime water; wet storage 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.

9. Design and Licensing Status

KLT-40S is the closest to commercialization of all available FNPP designs, and expects deployment through the Akademik Lomonosov FNPP. 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.

10. 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 19^{th} of December in Pevek
May 2020	Fully commissioned in Pevek on 22^{nd} of May

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

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Afrikantov OKBM, Rosatom, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	198 / 50
Primary coolant system circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.7 / 3.83
Core inlet/outlet coolant temperature (°C)	284 / 321
Fuel type/assembly array	UO_2 (cermet fuel) pellets / hexagonal
Number of fuel assemblies in the core	241
Fuel enrichment (%)	<20
Refueling cycle (months)	Up to 120
Core discharge burnup (GW·d/ton)	–
Reactivity control mechanism	Control rod drive mechanism; within the control and protection system (CPS)
Approach to safety system	Combined active and passive system
Design service life (years)	60
Ship footprint (m^2)	3360
RPV height/diameter (m)	8.6 / 3.45
RPV weight (metric ton)	265
Seismic design (SSE)	0.3g
Requirements for or approach to the operating cycle	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 (2 reactors are in the process of testing)

1. Introduction

RITM series reactors RITM-200 and RITM-200M are the latest development in Generation III+ SMR line designed by the Afrikantov OKBM and has incorporated all the best features from its predecessors. Floating power units (FPUs) with the RITMs are commercially available for medium and long terms. The RITM-200M adopts refueling cycle up to 10 years.

2. Target Application

The RITM-200M design was developed for the use on the Optimized Floating Power Unit (OFPU). The OFPU is a power generating facility in the form of a compact non-self-propelled vessel having two RITM-200M reactor plants. The FPUs based on RITM-200M may satisfy the needs of small residentials or industrial facilities. The OFPU can provide electricity to domestic and industrial consumers. The

OFPU can also be used for heat supply and water desalination purposes after installing additional equipment.

3. 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 60 years (more than 400 reactor-years). Incorporation the steam generators into the reactor pressure vessel, the reactor system and containment is very compact compared to the KLT-40. RITM design makes it possible to increase electric output (40% more) and reduces the dimensions (45% less) and the mass (35% less) in compare with KLT-40S.

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Cooling System

The reactor pressure vessel (RPV) is thick-walled cylindrical pressure vessel. The reactor is designed as an integral vessel with the main circulation pumps (MCP) located in separate external hydraulic chambers with side horizontal sockets for steam generator (SG) cassette nozzles. Each of the four SGs have 3 rectangular cassettes, while the four main circulation pumps are installed in the colds leg of the primary circulation path and separated into four independent loops. The SGs generate steam of 295°C at 3.83 MPa flowing at 261 (280) t/h. The conventional MCPs are used.

(e) Steam Generator

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

(f) Pressurizer

The design adopts pressure compensation gas system proved comprehensively in the Russian ship power engineering. It is characterized by a simple design, which increases reliability, compactness, and no electric power required. The design enables the use one pressurizer as a hydraulic accumulator, increasing reactor plant reliability considerably in potential loss-of-coolant accidents.

5. Safety Features

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

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

(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 channels and their functional and/or physical separation are provided to ensure high reliability. Safety systems are driven automatically by the control system, when controlled parameters achieve appropriate set points. In case of automated systems failure, self-actuating devices will actuate directly under the primary circuit pressure to ensure reactor trip and initiate the safety systems.

(b) Residual Heat Removal System

The residual heat removal system (RHRS) consists of four safety trains:
- Active safety loop with forced circulation through steam generator.
- Active safety loop with forced circulation through primary-third circuit heat exchanger of primary circuit coolant purification loop.
- Two passive safety loops with natural coolant circulation from water tanks through steam generators. Evaporated is steam generators water condenses in air cooled heat exchangers and flow back to tanks with water heat exchangers. After complete water evaporation from the tanks, the air cooled exchangers continue provide cooling for unlimited time.

(c) Emergency Core Cooling System

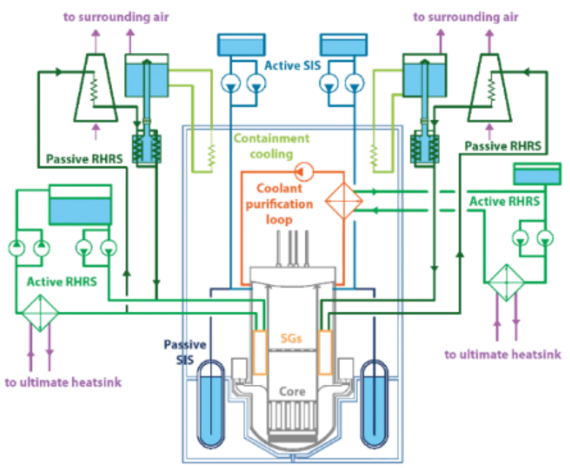
The ECCS consists of safety injection system (SIS) for water injection in primary circuit to mitigate the consequences of a break loss-of-coolant accident. The system is based on active and passive principles with redundancy of active elements in each channel and consists of (i) two passive pressurized hydraulic accumulators; (ii) two active channels with water tanks and two make-up pumps in each channel.

(d) Containment System

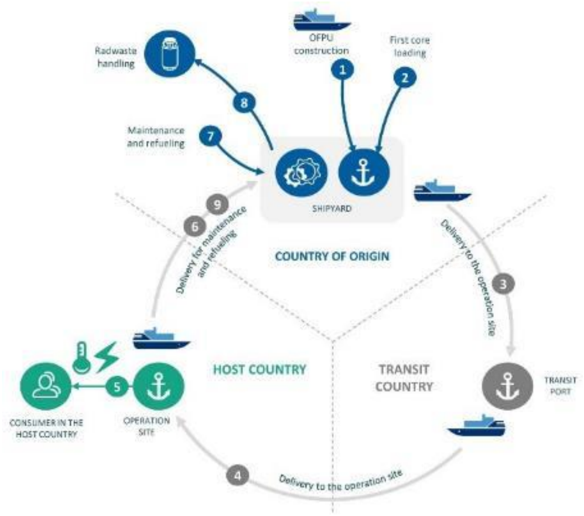
RITM is placed within the hermetically sealed envelope with dimensions of 6.6 m×6.4 m×16.2 m around the reactor vessel to localize possible radioactive releases. In case of severe accident thick wall of the reactor vessel keeps molten corium within the reactor. Water filled caisson under the reactor provides the reactor vessel cooling. The envelope integrity ensured by overpressure relief valve, containment cooling system, and passive autocatalytic recombiners.

6. Plant Safety and Operational Characteristics

The country of origin constructs the OFPU and conducts the first core loading. The transportation to the operation site is through the territorial sea of transit countries. Power and heat are generated at the operation site in the host country for up to 10 years without refueling. For maintenance, refueling and radwaste handing, the OFPU is returned to the country of origin. Afterwards, it is transported to the operation site in the customer's country.



RITM-200M reactor safety system



OFPU life cycle

7. Instrumentation and Control System

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

8. General Layout of the Plant

The OFPU with the RITM-200M reactor is designed to supply electricity, thermal power, and desalinated water to coastal or isolated territories, offshore installations, islands, and archipelagos. The OFPU can be rapidly delivered to the site by sea. The only needs to launch operation is docking pier and onshore power transmitting infrastructure

OFPU fitted with the RITM-200M reactor plant

9. Testing to Check and Validate the Design

The engineering solutions used in the design are traditional for marine power engineering. The solutions have been checked in the course of many operating years and ensured the required reactor plant reliability and safety performance. The RITM-200M reactor pertains to integral-type reactors. Integral-type reactors are used in a series of Project 22220 multipurpose nuclear-powered icebreakers *Arktika, Sibir* and *Ural.*

10. Design and Licensing Status

The RITM-series reactors have been developed in conformity with Russian laws, codes and standards in a peaceful use of atomic energy and in conformity with IAEA recommendations. At present, the reactors are manufactured and installed on nuclear-powered icebreakers; the OFPU design is under development.

11. Approach to the Operating Cycle

The OFPU is delivered to the site with fresh fuel in its reactors. After the cycle of operation is over, the OFPU comes back to the exporting country along with the spent fuel in its reactors. Spent nuclear fuel post-irradiation handling and reprocessing are performed in the exporting country.

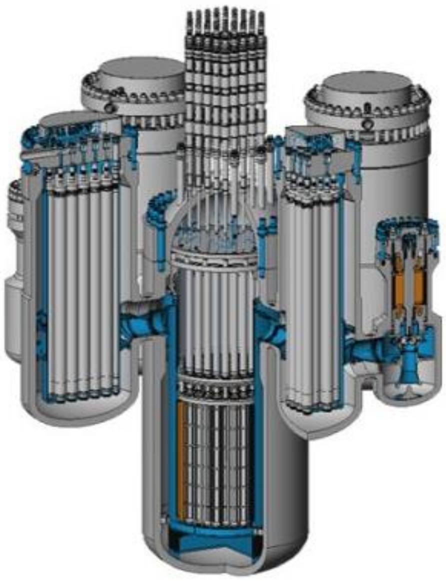
12. Waste Management System and Waste Disposal Plan

The waste is stored within the OFPU, not in the operation site water area. The waste ensuing from plant operation is compact, has a low activity level and is securely isolated from the biosphere. It has been verified that there is no effect to marine organisms in the deployment site water area.

13. Development Milestones

2016	The first RITM-200 installed on board the icebreaker *Arktika*
2020	The icebreaker *Arktika* under testing
2020	A conceptual design of the OFPU with RITM-200M
2024	Technical design of RITM-200M for floating nuclear power unit was completed
2024	Design of floating power unit is expected to be completed

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 (ºC)	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. 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.

4. Main Design Features

(a) 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.

(b) 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.

(c) 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.

(d) 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.

(e) 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.

(f) 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.

(g) Pressurizer

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

5. Safety Features

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

(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.

6. Plant Safety and Operational Performances

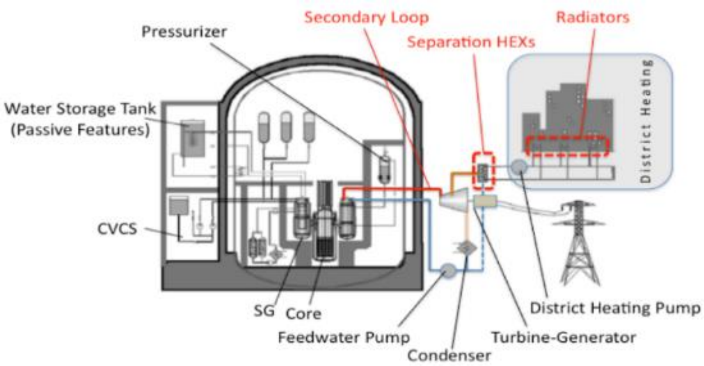
The VBER-300 safety concept is based on the defence-in-depth principles. With the modular configuration, it has increased resistance to impact loads in case of earthquake and aircraft crash. Results of strength analysis under seismic loads of up to 8 points according to the MSK-64 scale carried out for 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.

7. 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.

8. Design and Licensing Status

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

9. 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 3×2 years and 4×1.5 years. The number of FAs in the refuelling batch is either 15 or 30; maximal fuel burnup does not exceed 60 GWd/ton U for the cycle with 30 fresh FAs in the reloading batch and maximum initial uranium enrichment.

10. 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

HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS

Energy Multiplier Module (EM^2) (General Atomics)

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

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	General Atomics (GA), United States of America
Reactor type	Gas-cooled Fast Reactor (GFR)
Coolant/moderator	Helium/None
Thermal/electrical capacity, MW(t)/MW(e)	500 MW(t) / 265 MW(e) per module, 1060 MW(e) total
Primary circulation	Driven by the Power Conversion Unit (PCU), no separate pumps
NSSS Operating Pressure (primary/secondary), MPa	13.3 / NA
Core Inlet/Outlet Coolant Temperature (°C)	550 / 850
Fuel type/assembly array	Uranium Carbide (UC), Hexagonal Assemblies
Number of fuel assemblies in the core	85
Fuel enrichment (%)	6.5 (average), 17 (maximum)
Core Discharge Burnup (GWd/ton)	140
Refuelling Cycle (months)	360
Reactivity control	Rods
Approach to safety systems	Defense-in-depth, passive safety systems
Design life (years)	60
Plant footprint (m^2)	5000
RPV height/diameter (m)	12.5 / 4.6
RPV weight (metric ton)	280
Seismic Design (SSE)	0.3 g
Distinguishing features	Convert-and-burn/Silicon carbide composite cladding/ Fission gas collection
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 closed Brayton cycle gas turbine power conversion unit 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 pellets enclosed in silicon carbide (SiC) composite cladding (i.e., SiGA®).

2. Target Application

The EM^2 reactor targets baseload electricity generation with high efficiency and modular design. It is suitable for both grid-connected and remote locations. Its long refueling cycle and low emissions make it ideal for sustainable energy solutions.

3. Design Philosophy

The EM² reactor embraces a "convert and burn" design philosophy, optimizing fuel use by separating fissile and fertile materials within the core. This approach enhances fuel efficiency and extends fuel burnup, thereby reducing waste and operational costs. The reactor's design integrates passive safety features and high thermal efficiency to ensure robust performance and long-term sustainability.

4. Main Design Features

(a) Nuclear Steam Supply System

EM² employs a combined cycle power conversion system (PCS) with a direct helium Brayton cycle and a Rankine bottoming cycle to maximize thermal efficiency. There is no nuclear steam supply system.

(a) Reactor Core

The core is designed with a "convert and burn" approach, featuring a mix of fissile and fertile materials to enhance fuel efficiency. It uses uranium carbide fuel assemblies and incorporates zirconium-alloy and graphite reflectors to sustain a high level of neutron flux.

(b) Reactivity Control

Reactivity is managed by 18 control rods and 12 shutdown rods, which ensure the reactor can be safely controlled and shut down if necessary. The control rods use a ball-screw mechanism, while the shutdown rods are driven by linear motors.

(c) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) is constructed from SA533 material, providing robust containment for the reactor core and primary coolant system. The RPV is designed to accommodate the high pressures and temperatures of the reactor operation while ensuring structural integrity. Internally, the vessel contains the reactor core, control mechanisms, and coolant channels, all designed to maintain operational safety and efficiency.

(d) Reactor Coolant System and Steam Generator

Helium is used as the primary coolant in a closed loop system to transfer heat from the reactor to the PCS. The PCS includes high-efficiency heat exchangers to manage heat transfer to the ultimate heat sink. There is no steam generator.

(e) Pressuriser

The operating conditions of the primary coolant helium are maintained by the helium purification system. There is no specific separate pressurizer.

(f) Primary pumps

Primary coolant circulation is managed by the turbocompressor of the PCS that is driven by hot helium flow from the reactor core. This closed Brayton cycle ensures a continuous and controlled flow of helium through the reactor core and PCS. There is no separate primary pumps.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

EM² reactor employs a comprehensive engineered safety system which features multiple redundant and diverse safety systems, including passive and active safety measures that function independently of external power sources. This configuration includes automatic shutdown systems, emergency core cooling systems, and a robust system for detecting and responding to any anomalies, ensuring the reactor's safe operation even in extreme scenarios.

(b) Safety Approach and Configuration to Manage DBC

EM² reactor uses advanced safety features to manage Design Basis Conditions (DBC). The Direct Reactor Auxiliary Cooling System (DRACS) provides effective heat removal during all operational states, and reactivity control is maintained with control and shutdown rods that ensure the reactor can be safely brought to a cold subcritical state. The containment system is designed to handle high-pressure scenarios, preventing the release of radionuclides, and ensuring robust protection.

(c) Safety Approach and Configuration to Manage DEC

EM² reactor uses a combination of passive and active safety systems to handle Design Extension Conditions (DEC). Key features include the DRACS for core cooling and the Fission Product Vent System (FPVS) for maintaining the lowest source term level. These systems work together to ensure public safety even under severe conditions.

(d) Containment System

EM²'s containment system is a sealed, below-grade structure divided into two interconnected chambers. It features a hermetically sealed design to ensure robust protection against radioactive releases. The system includes structural ligaments with additional shielding and is designed to handle peak pressures up to 0.62 MPa with an argon atmosphere. The leakage rate is less than 0.2% per day.

(e) Spent Fuel Cooling Safety Approach / System

EM² employs a passive spent fuel cooling system that utilizes natural convection to remove heat from spent fuel. Spent fuel assemblies are stored in sealed, air-cooled containers located in a dedicated storage facility. This facility is designed to handle cooling for up to 60 years of operation without requiring active cooling systems or water, ensuring long-term safety and minimal operational complexity.

6. Plant Safety and Operational Performances

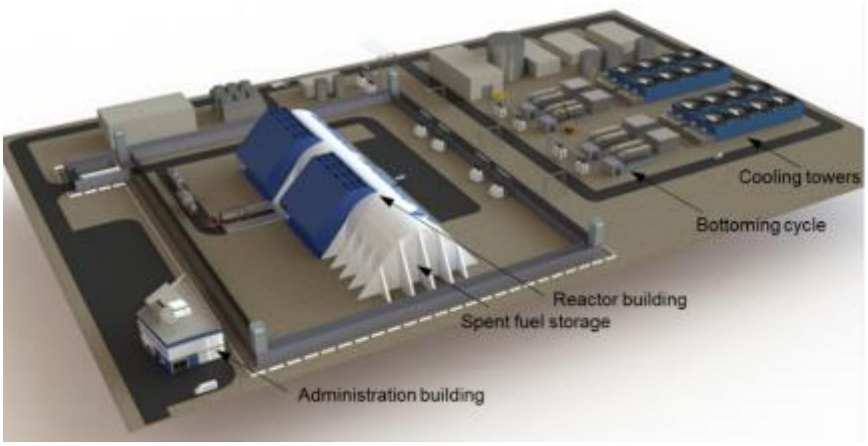
EM² reactor, plant safety and operational performance are ensured through a multi-layered safety approach and high operational efficiency. The plant employs advanced safety systems like the DRACS and helium purification systems to manage core heat and coolant quality. Operationally, the EM² is designed for high availability with a high capacity factor and modular construction to streamline maintenance and reduce downtime.

7. Instrumentation and Control Systems

EM² reactor utilizes advanced instrumentation and control systems to maintain safe and efficient operation. Advanced sensors, including solid-state and silicon carbide neutron flux monitors, provide precise data for reactor control. The integrated control system manages reactor power, coolant flow, and turbine operations, ensuring responsive adjustments to plant conditions and maintaining operational stability.

8. Plant Layout Arrangement

EM² plant layout is designed for efficiency and accessibility, covering a total footprint of 9.3 hectares (23 acres). It features a modular configuration with four 265 MW(e) reactors, each including a complete powertrain from reactor to heat rejection, allowing for sequential construction and independent operation. The maintenance hall is situated at grade level and serves all four reactors, with the roof designed as a protective shield and the facility organized to support effective maintenance and operational access.



EM² plant layout on 9 hectares of land

9. Testing Conducted for Design Verification and Validation

Design verification and validation for the EM^2 reactor involve extensive testing to ensure that all systems perform as expected. Experimental measurements have been partially conducted for: i) UC spherical kernel irradiation tests in High Flux Isotope Reactor (HFIR) in 2021 and measurements of fission gas release and swelling in 2022, ii) a series of thermal, mechanical, and safety measurements of the SiC composite tube, and iii) a low-fluence irradiation of zirconium silicide reflector material in 2016.

10. Design and Licensing Status

EM^2 reactor is currently in the conceptual design phase. The reactor design has undergone several reviews to meet regulatory and safety requirements, and it is progressing towards more design improvements and regulatory engagement.

11. Fuel Cycle Approach

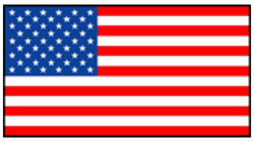
The EM^2 open fuel cycle with LEU/DU vented fuels exceeds 30 years without refuelling or shuffling, leading to a reduced cost of power, low proliferation risk, high fuel utilization, and low mass of waste streams. The core can burn used light water reactor (LWR), plutonium, and thorium fuels. Fissile self-sufficient fuel cycle is feasible by removing fission products from the EM^2 spent fuel.

12. Waste Management and Disposal Plan

EM^2 reactor minimizes nuclear waste through high burnup fuel and recycling strategies. It extracts more energy from fuel, reducing the volume and radiotoxicity of spent fuel. Recycled materials are safely stored and eventually disposed of in a geological repository, ensuring responsible waste management.

13. Development Milestones

2009	Start of design	Complete
2013	Pre-conceptual design complete	Complete
2020	Pre-application license review (in USA)	On track



Fast Modular Reactor (FMR) (General Atomics)

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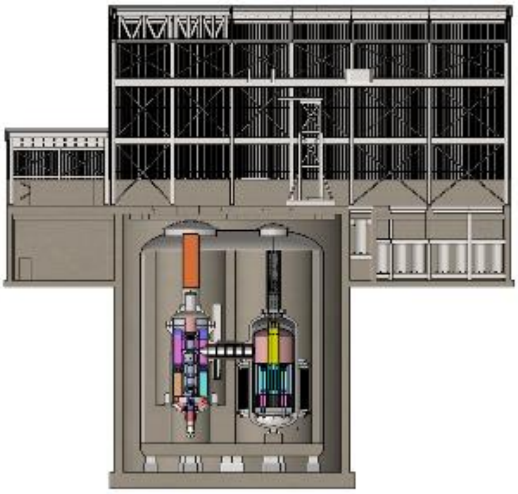
KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	General Atomics, United States of America
Reactor type	Gas-cooled Fast Reactor (GFR)
Coolant/moderator	Helium / None
Thermal/electrical capacity, MW(t)/MW(e)	100 MW(t) / 44 MW(e)
Primary circulation	Driven by the Power Conversion Unit (PCU), no separate pumps
NSSS Operating Pressure (primary/secondary), MPa	7 / N/A
Core Inlet/Outlet Coolant Temperature (°C)	506 / 800
Fuel type/assembly array	Uranium Dioxide (UO_2) / Triangular array
Number of fuel assemblies in the core	198
Fuel enrichment (%)	17.6% (average) / 19.75% (maximum)
Core Discharge Burnup (GWd/ton)	100
Refuelling Cycle (months)	180
Reactivity control	Control rods, Burnable absorbers (B_4C)
Approach to safety systems	Combination of active and passive systems, no core melt provisions, reactor vessel passive cooling system
Design life (years)	60
Plant footprint (m^2)	2,000 m^2 (reactor building) / 38,000 m^2 (site)
RPV height/diameter (m)	11 / 4.6
RPV weight (metric ton)	230
Seismic Design (SSE)	0.3 g
Distinguishing features	Helium coolant, SiC composite cladding, dry-cooling heat sink, closed Brayton cycle
Design status	Conceptual Design



Reactor building and major systems and components arrangement designed with defense-in-depth

1. Introduction

The Fast Modular Reactor (FMR) is a 100-MW thermal Gas-cooled Fast Reactor (GFR) developed by General Atomics Electromagnetic Systems (GA-EMS) for the US electricity market by the mid-2030s. Designed to be safe, maintainable, and cost-effective, the FMR features helium coolant for high-temperature operation, uranium dioxide fuel for long life and high burnup, and silicon carbide (SiC) composite cladding for radiation tolerance. Its simplified design includes air-cooling as an ultimate heat sink and passive heat removal systems, making it a flexible and dispatchable power source.

2. Target Application

Its primary application is to provide a flexible and reliable source of energy that can provide baseload power, support grid stability and complement renewable energy sources. With a focus on modularity, the FMR aims to meet the growing need for small, distributed nuclear power stations that can be deployed in diverse locations and offer a low-carbon alternative to conventional baseload generation.

3. Design Philosophy

The FMR design philosophy focuses on simplicity, safety, and efficiency. It uses an inert helium coolant and SiC composite materials to enhance safety and thermal efficiency. The reactor is modular with an air-cooling system, aiming for cost-effectiveness and adaptability to various grid needs.

4. Main Design Features

(a) Nuclear Steam Supply System

The equivalent Nuclear Steam Supply System (NSSS) of the FMR integrates the reactor core with the power conversion unit (PCU). It features a direct Brayton cycle where helium gas, heated in the reactor, drives a turbine to generate electricity. The system's design includes a reactor pressure vessel (RPV) connected to the PCU via a cross-duct, ensuring efficient heat transfer and power generation. There is no system that supplies steam.

(b) Reactor Core

The reactor core of the FMR is a compact, annular design surrounded by advanced neutron reflectors like zirconium silicide and graphite. Low core power density and use of uranium dioxide (UO_2) fuel and SiC composite matrix ceramic cladding secures the safety margin of the fuel.

(c) Reactivity Control

FMR utilizes a combination of reactivity control mechanisms to ensure stable and safe operation. The primary method involves control rods that manage reactivity changes and maintain the core's stability. Additionally, the reactor design incorporates a three-batch refuelling scheme to optimize fuel use and minimize the need for excess reactivity control.

(d) Reactor Pressure Vessel and Internals

FMR RPV is a cylindrical structure made from stainless steel, specifically 316H, designed to contain the reactor core and primary coolant. The internals include the high-temperature upper support structure, the lower core support structure, and the in-core instrumentation support structure.

(e) Reactor Coolant System and Steam Generator

The FMR utilizes a helium-based reactor coolant system that ensures efficient heat transfer and high thermal efficiency. The coolant, being inert and single-phase, facilitates operation at high temperatures while minimizing chemical reactions. The system integrates with a closed Brayton cycle PCU, where helium transfers heat from the reactor core to the turbine-generator of the PCU. There is no steam generator.

(f) Pressuriser

FMR design uses a helium coolant in a direct Brayton cycle system, which generally does not require a traditional pressurizer. The operating conditions of the helium coolant are maintained by the helium service system.

(g) Primary pumps

FMR utilizes a helium coolant in a direct Brayton cycle, which typically does not require conventional primary pumps. The coolant's circulation is achieved through the use of compressors and turbine integrated into the PCU. This design eliminates the need for separate primary pumps, as helium's properties facilitate efficient coolant flow and heat transfer within the reactor core and the PCU.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The engineered safety features (ESFs) of the FMR mitigate the consequences of postulated accidents. The ESFs include reactivity control system, reactor protection system, maintenance cooling system, reactor vessel cooling system (RVCS), and containment.

(b) Safety Approach and Configuration to Manage DBC

FMR employs robust design features to handle various Design Basis Conditions (DBCs). It integrates passive safety systems like the RVCS for heat removal, which ensures safety during design-basis accidents (DBAs) without relying on active systems.

(c) Safety Approach and Configuration to Manage DEC

The reactor features passive safety systems, such as the RVCS, which ensures heat removal even in the event of a loss of coolant accident (LOCA) or other extended conditions. The design includes redundant safety systems for core cooling and containment, which function without relying on external power.

(d) Containment System

The sealed containment vessel is a free-standing cylindrical pressure vessel constructed of stainless-steel plate that is located inside the concrete building structure, independent of the concrete mat foundation, walls, or slab above. The main function of the sealed containment is to prevent the uncontrolled release of radiological material into the surrounding environment under normal operating, abnormal accident, or extreme environmental conditions including severe accident temperature and pressure changes.

(e) Spent Fuel Cooling Safety Approach / System

FMR's spent fuel cooling safety approach involves passive heat removal and natural circulation to maintain safe temperatures. The spent fuels are stored in dry storage canisters designed for long-term safety and cooling through air convection. The system employs both passive and active safety features to ensure effective heat management and containment of spent fuel throughout its lifecycle.

6. Plant Safety and Operational Performances

The plant design incorporates measures to prevent unauthorized access and/or theft of nuclear material, complying with 10 CFR 73.55 for physical protection of licensed activities in nuclear power reactors against radiological sabotage and 10 CFR 73.67 for the physical protection of special nuclear material of moderate and low strategic significance.

7. Instrumentation and Control Systems

Instrumentation and Control (I&C) system consists of the safety-related reactor protection system (RPS) and the non-safety-related plant protection system (PPS), plant control, data, and instrumentation system (PCDIS), and plant monitoring system (PMS). The I&C employs methods such as prioritization and plant state dependency for alarms, to avoid operator information overload caused by untimely or unnecessary audible or visual information.

8. Plant Layout Arrangement

FMR plant contains four main buildings for the reactor, auxiliary system, cooling tower, and administration. The reactor building includes the underground containment structure, which contains the sealed containment vessel, reactor, power conversion system, and reactor cooling system, as well as the reactor vessel passive cooling system, control room, spent fuel storage, helium service system, and support equipment. The reference design allocates 9.5 acres for the FMR plant, excluding the switchyard and exclusion zones. The total space allocation including the switchyard is 16.5 acres.

Site-flexible FMR plant layout

9. Testing Conducted for Design Verification and Validation

FMR design validation involves several key testing and verification activities to ensure its performance and safety. Experimental measurements have been partially conducted for the high-burnup UO_2 fuel measurements of fission gas release and swelling, a series of thermal, mechanical, and safety measurements of the SiC composite tube, low-fluence irradiation of zirconium silicide reflector, and passive reactor vessel cooling.

10. Design and Licensing Status

FMR is in its pre-licensing phase, with key milestones including the completion of concept design by 2026, a pre-licensing review starting in 2027, and the start of construction in 2030. The engineering design will be finalized and licenses secured by 2029, with commercial operation expected by 2036. Examples of pre-licensing documents include the Principal Design Criteria (PDC) and Quality Assurance Program (QAP).

11. Fuel Cycle Approach

FMR baseline design uses a once-through fuel cycle approach, involving initial use of UO_2 fuel without recycling. The design includes interim storage and direct disposal of spent fuel. The fuel cycle is optimized for high burnup and long operational life, with a focus on simplicity and safety. The FMR cam also recycle and/or transmute used nuclear fuels with appropriate recycling technologies.

12. Waste Management and Disposal Plan

Radioactive wastes are segregated into liquid, gaseous, and solid categories. The liquid waste undergoes filtration and demineralization before being processed and combined with solid waste. Gaseous wastes are reduced in activity to safe levels before atmospheric release. Solid wastes are collected, solidified, packaged, and stored for offsite disposal. The waste management plan ensures minimal environmental impact and complies with regulatory requirements for waste handling and disposal.

13. Development Milestones

2022	Start of design	Complete
2026	Concept design complete	On track
2027	Start of pre-licensing vendor design review (in USA)	Planned
2029	Engineering design complete/ Secure necessary licenses (in USA)	Planned
2030	Start construction of a demonstration unit (in USA)	Planned
2036	Commercial operation	Planned

GTHTR300 (JAEA Consortium, Japan)

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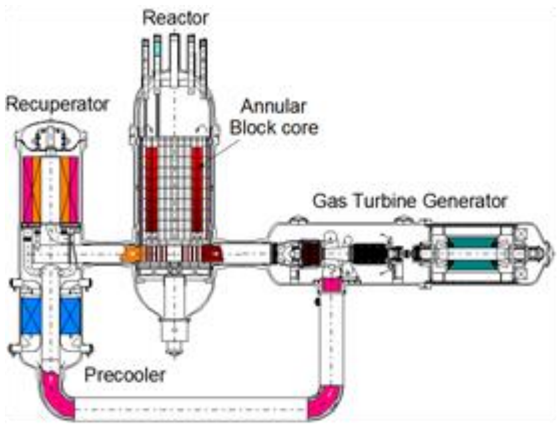
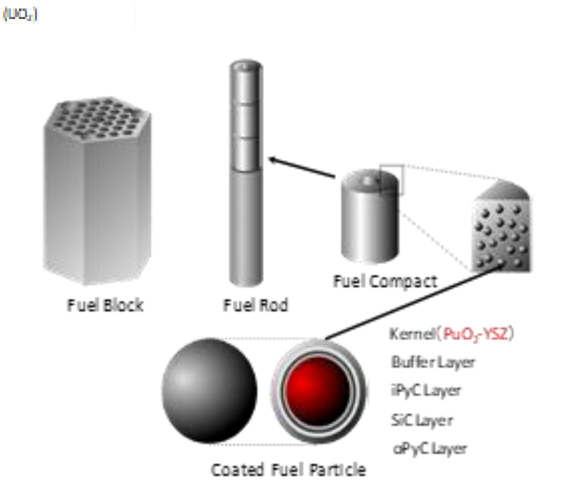




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JAEA, MHI, Toshiba/IHI, Fuji Electric, KHI, NFI, Japan
Reactor type	Prismatic HTGR
Coolant/moderator	Helium / graphite
Thermal/electrical capacity, MW(t)/MW(e)	<600 / 100~300
Primary circulation	Forced by gas turbine
NSSS Operating Pressure (primary/secondary), MPa	7 / 7
Core Inlet/Outlet Coolant temperature (°C)	587-633 / 850-950
Fuel type/assembly array	UO_2 TRISO ceramic coated particle
Number of fuel assemblies in the core	90
Fuel enrichment (%)	14
Refuelling Cycle (months)	48
Core Discharge Burnup (GWd/ton)	120
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Active and passive
Design life (years)	60
Plant footprint (m²)	~250 x 250 (4-reactor plant)
RPV height/diameter (m)	23 / 8
RPV weight (metric ton)	~1 000
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	Basic design

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) being developed by Japan Atomic Energy Agency (JAEA) for commercialization in 2030s. As a Generation-IV technology, the GTHTR300 offers important advances over current light water reactors. The coolant temperature is significantly higher in the range of 850-950°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%.

2. Target Applications

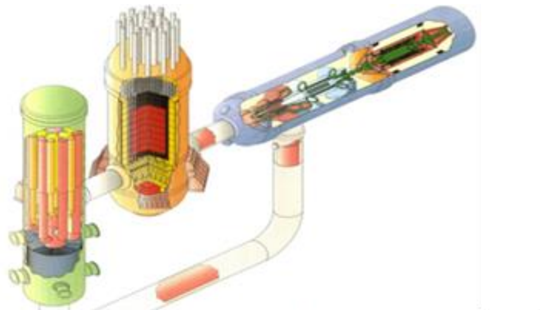
Typical applications include electric power generation, thermochemical hydrogen production, desalination cogeneration using waste heat, and steel production. The reactor thermal power may be rated up to 600 MW(t). The maximum hydrogen production per reactor is 120 t/d, enough to fuel about one million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

4. Main Design Features

(a) Power Conversion



Reactor Primary System of the GTHTR300

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.

(b) Reactor Core

The reactor core consists of 90 fuel columns arranged in an annular ring, 73 and 48 inner and outer removable reflector columns, and 18 outer fixed reactor sectors. The effective annular core is about 3.6-5.5m in inner to outer diameter and 8m in height. The fuel column is a hexagonal graphite block similar to the HTTR fuel but improved from it by an integral or sleeveless fuel rod for increased heat flux and by and enlarged coated particle buffer for improved fission product retention integrity under high burnup.

(c) Fuel Characteristics

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

(g) Reactor Coolant System

Helium is heated in the reactor core to the cycle top temperature at high pressure. It then enters the turbine for expansion to convert thermal energy into shaft power needed by the turbine to drive the compressor and electric generator on single shaft. The turbine exhaust helium enters the recuperator, wherein its residual heat is recovered in high effectiveness to preheat coolant to the reactor..

5. 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 1600^{o}C.
- 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 System / 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.

(c) Spent Fuel Cooling Safety Approach / System

The fuel cycle is of once-through-then-out design allowing for 1460 days (4 years) of in-core fuel residence with 120 GWd/ton average burnup. The spent fuel is stored in dry cooling facility with expenses included in fuel cycle cost estimate. Fuel reprocessing is technically feasible and preliminary evaluation for this option is being conducted.

(d) Containment System

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.

(e) Chemical Control

The helium purification systems are installed in the primary and secondary cooling systems in order to reduce the quantity of chemical impurities such as hydrogen, carbon monoxide, water vapor, carbon dioxide, methane, oxygen, and nitrogen. The primary helium purification system is mainly composed of a pre-charcoal trap, an inlet heater, two copper oxide fixed beds, coolers, two molecular sieve traps, two cold charcoal traps and helium compressors.

6. 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. 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.

7. Instrumentation and Control System

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.

8. 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.

9. Testing Conducted for Design Verification and Validation

The test results using the HTTR will be utilized for the development of GTHTR300. The test items cover fuel performance and radionuclide transport, core physics, reactor thermal hydraulics and plant dynamics, and reactor operations, maintenance, control, etc. The results of the system performance analysis showed that the reactor could be continuously operated with the above variable load conditions. However, an actual demonstration test is warranted for performance confirmation.

10. 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 demonstration plant operation in 2040s.

11. Fuel Cycle Approach

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

12. Waste Management and Disposal Plan

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

13. Development Milestones

2003	Basic design of GTHTR300 completed
2005	Design for cogeneration plant GTHTR300C
2015	Basic design for HTTR-connected gas turbine and H_2 plant (HTTR-GT/H_2) for system demonstration
2020s	HTTR-H_2 test plant construction and operation (planned)
2040s	Operation of demonstration plant (TBD)

GT-MHR (JSC "Afrikantov OKBM", Russian Federation)

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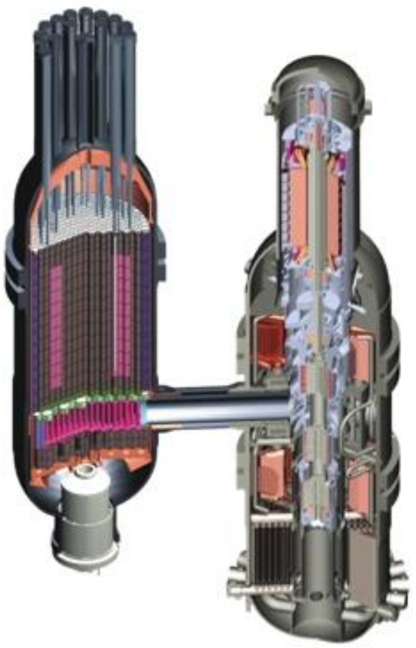


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", Russian Federation
Reactor type	Modular Helium Reactor
Coolant/moderator	Helium /graphite
Thermal/electrical capacity, MW(t)/MW(e)	600 / 288
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	7.2 / -
Core Inlet/Outlet Coolant Temperature (^{o}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. 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.

4. Main Design Features

(a) Power Conversion System Flow Diagram

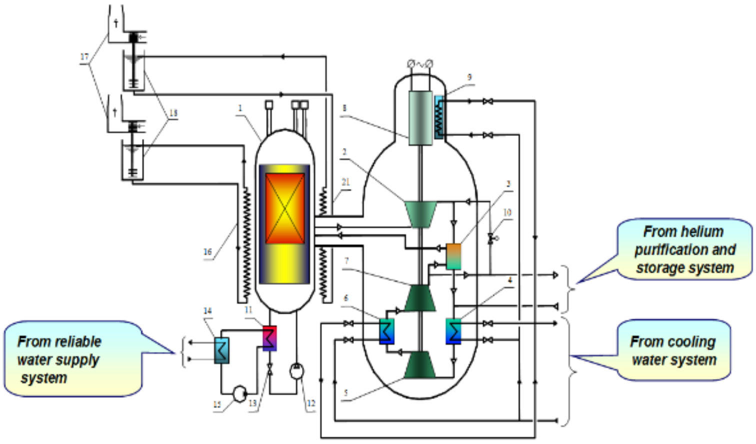
1. Reactor
2. Gas Turbine
3. Recuperator
4. Pre-Cooler
5. Low Pressure Compressor
6. Intercooler
7. High Pressure Compressor
8. Generator
9. Generator cooler
10. Bypass Valve

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.

(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 1600°C. About 1 billion fuel particles of the same type were manufactured and tested in Russia. The standard fuel cycle for the commercial GT–MHR utilizes low enriched uranium (LEU) but alternative cycles including the disposition of plutonium were also studied in detail.

(c) 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.

(d) 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.

5. 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.

6. Plant Safety and Operational Performances

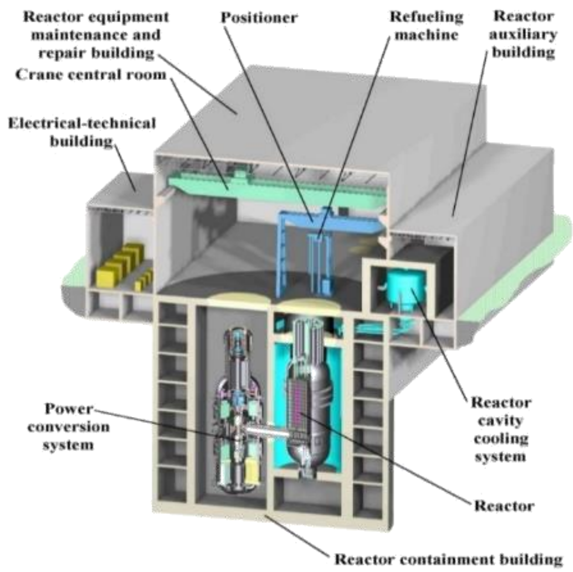
All safety systems are designed with two channels. Fulfilment of the regulatory requirements on safety, proven by a compliance with both deterministic and probabilistic criteria, is secured by an exclusion of the active elements in a channel or by applying the required redundancy of such active elements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

7. 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.

8. Plant Layout Arrangement

The plant layout is shown on the right.



9. Design and Licensing Status

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

10. 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.

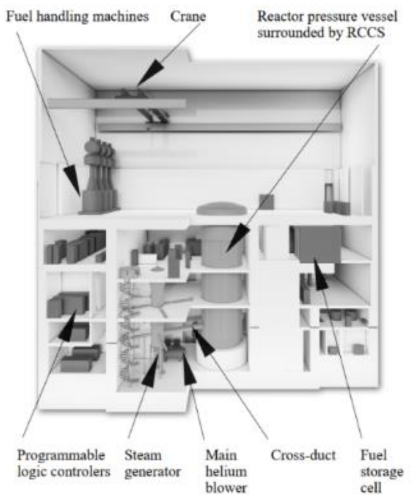
11. Waste Management and Disposal Plan

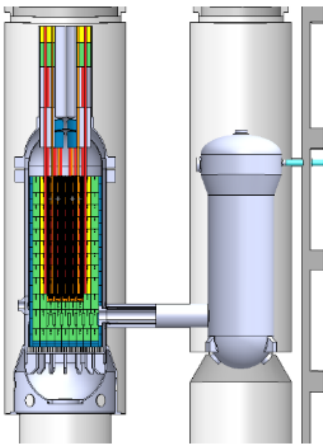
Facilities for long-term storage of spent nuclear fuel (SNF) and solid/solidified radioactive waste (RW) are included in the complex of a GT-MHR commercial 4-unit NPP. The capacity of the designed SNF storage is determined from the condition of capability to store fuel unloaded from the NPP for 10 years. 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.

12. 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)

HTGR-POLA (NCBJ, Poland)





KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	National Centre for Nuclear Research (NCBJ), Poland
Reactor type	HTGR
Coolant/moderator	Helium/Graphite
Thermal/electrical capacity, MW(t)/MW(e)	30/11.5
Primary circulation	Helium
NSSS Operating Pressure (primary/secondary), MPa	4.0/13.8
Core Inlet/Outlet Coolant Temperature (°C)	325/750
Fuel type/assembly array	UO_2 TRISO/ compact/ graphite blocks
Number of fuel assemblies in the core	114
Fuel enrichment (%)	10-12
Core Discharge Burnup (GWd/ton)	40
Refuelling Cycle (months)	20
Reactivity control	B_4C control rods, reserved B_4C pellets
Approach to safety systems	Inherent safety, passive and active
Design life (years)	60
Plant footprint (m^2)	25,000
RPV height/diameter (m)	16.2/4.08 (belt line)
RPV weight (metric ton)	200
Seismic Design (SSE)	0.3g
Distinguishing features	Prismatic fuel blocks
Design status	Basic Design

1. Introduction

The implementation of High-Temperature Gas Reactors (HTGRs) in Poland began in 2012, following the start of the Polish Nuclear Power Program in 2010. The Paris Agreement in 2015 aimed to reduce greenhouse gas emissions, Poland is investing in renewables and nuclear to decarbonise the energy mix around 2050.In 2016, the Minister of Energy appointed the "Committee for Analysis and Preparation of Conditions for Deployment of High-Temperature Nuclear Reactors", which concluded that Poland should build a fleet of industrial HTGRs with 180 MW_t each, providing steam at 540 C, 13.8 MPa, 230 t/h. However, due to the lack of commercial reactors in Poland, experts decided to construct a research HTGR of 30 MW thermal power as a technology demonstrator to convince regulator and the industry about its safety, economy, and practicality. The HTGR-POLA reactor is an updated design of the research-demo high temperature gas-cooled reactor based on the Japanese HTTR (High Temperature Engineering Test Reactor), designed to operate intermittently and produce electricity and heat for targeted practical applications.

2. Target Application

The HTGR-POLA is a demonstration for HTGR technology, supplying superheated steam to address Poland's industrial needs. The plant design incorporates solutions for a conventional island and secondary loop facilities. Test plans consider superheated steam for industrial applications and utility applications like electricity generation and district heating. Three primary cogeneration modes are identified: electricity production with a maximum output of 11.5 MW_e, superheated steam with a maximum output of 25 t/h, and communal heat with a maximum output of 16.5 MW_t.

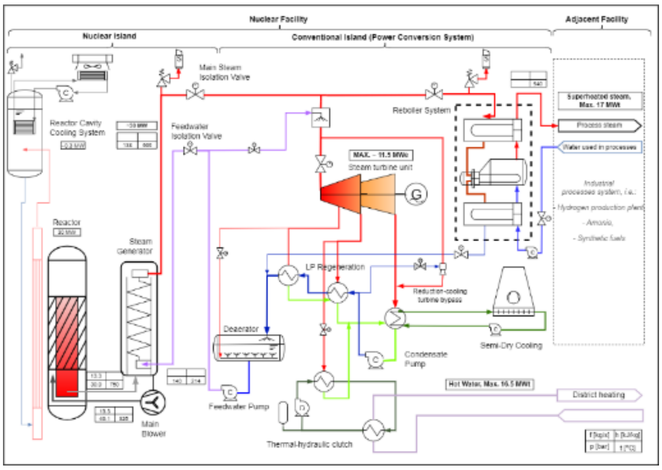
3. Design Philosophy

The research HTGR-POLA reactor aims to serve building competence (human resources, industry, regulator, etc.), research tasks, and as a small-scale demonstrator of HTGR technology for industrial applications. It combines features of the European GEMINI Plus reactor design with the proven properties of HTTR. The reactor design matches Polish requirements for research, demonstration, and applications, with the goal of maximizing similarity to an industrial type First-Of-A-Kind (FOAK) 180 MW_t reactor design.

4. Main Design Features

(a) Nuclear Steam Supply System

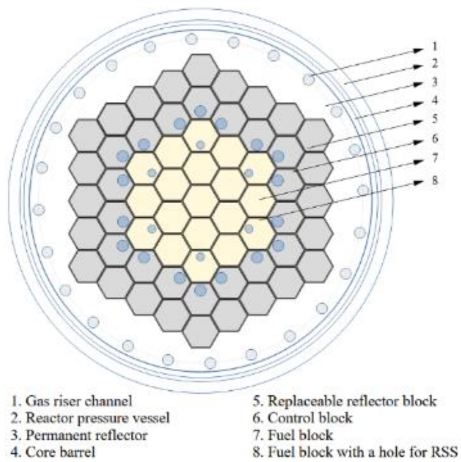
The NSSS consists of a reactor pressure vessel, steam generator, and helium blower. It operates at 30 MW_t, with helium coolant transporting heat from the reactor core to the steam generator. The core's coolant temperature increases to 750°C, with a 4.0 MPa coolant pressure. Heat is transferred to the secondary circuit, generating superheated steam (565°C, 13.8 MPa). The steam is split into two main flows, directed to the steam turbine and to the reboiler, which generates final process steam for various applications.



Simplified diagram of the HTGR- POLA NSSS

(b) Reactor Core

The active core of a fuel engine is composed of 19 fuel columns arranged in triangular pitch, forming two rings around a central fuel column. Each column contains a stack of six fuel blocks. A single fuel block is a hexagonal prism measuring 36 cm across the flats and 60 cm in height. The fuel block incorporates a grid of fuel holes and coolant channels. There are two types of fuel blocks: standard blocks and control blocks (with a channel for reserved shutdown system). The active core height is 3.6 m, and its equivalent diameter is 1.66 m. TRISO fuel with kernels enriched in ^{235}U (HALEU, 10-12 wt.%, UO_2) is used, with fuel compacts with packing factor of 15-30% inserted into dedicated holes. Between four to six of the fuel holes may be occupied by burnable poison rods instead of fuel compacts. Helium coolant flows through the coolant channels in fuel blocks. The active core is radially surrounded by two rings of replaceable reflector blocks. Selected reflector blocks are designed with channels for the control rods. There are two layers of replaceable reflector blocks in each location with the external dimensions being the same as the fuel blocks, except for height.



Reactor core configuration

(c) Reactivity Control

The reactor design uses two independent reactivity control systems: Control Rods (CR) and Reserve Shutdown System (RSS). 18 CR, using B_4C as neutron poison, are used for active operational control. If a CR becomes inoperable, the RSS is activated. The RSS uses cylindrical pellets (B_4C/C) released from hoppers into six channels within the active core. Burnable poison rods (B_4C/C) are used for long-term reactivity excess management. To adjust power, one needs to reduce heat uptake by reducing coolant flow, matching reactor power to the altered flow.

(d) Reactor Pressure Vessel and Internals

The core assembly, comprising fuel blocks and replaceable reflector blocks, is positioned on graphite core support structures (such as hot plenum blocks and support posts) and metallic core support structures (including support plates and grids). The core assembly and its support structures are encased by the permanent side reflector, consisting of graphite segments. This entire structure is contained within a metallic core barrel, which is ultimately embedded within the reactor pressure vessel.

(e) Reactor Coolant System and Steam Generator

The coolant from the upper mixing chamber is directed to the active core, then to the lower mixing chamber. It exits the RPV and is directed to the cross-duct. Hot helium is then sent to the steam generator (SG), where it is cooled and directed to the external main helium blower. The helium then flows back to the SG through the outer SG shell, and into RPV, where it enters the upper mixing chamber.

(f) Pressuriser

Pressure in the primary cooling circuit is controlled by helium purification and supply systems.

(g) Primary pumps

The reactor coolant pressure boundary includes a single-stage, magnetic-bearing main blower, and pipes connected to the steam generator. The cooled helium from the steam generator is directed to the main blower while high-pressure helium is routed back to the reactor pressure vessel.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

HTGR has unique safety properties due to its strong negative fuel, moderator, and reflector temperature coefficient of reactivity and large heat capacity of bulk graphite in the core. Increase fuel temperature decreases fission reactions and reactor power, leading to spontaneous shutdown in accident conditions. There is no safety injection of coolant to the reactor coolant system, and heat is dissipated through conduction, natural convection, and thermal radiation.

(b) Safety Approach and Configuration to Manage accident conditions

The HTGR-POLA design employs a defence-in -depth concept for accident management focusing on prevention and mitigation through active and passive systems. The inherent reactor features are emphasized, ensuring high-risk events are categorized as low probability. The first two layers of defence focus on preventing accidents using high quality materials like TRISO fuel and large safety margins (level 1). The I&C system detects and restores required parameters after departure from safe operation (level 2). Mitigation is achieved through activating level 3 defence-in-depth layer systems, such as passive safety system for core heat removal and maintaining physical barrier integrity. The Reactor Cavity Cooling System controls RPV wall and core temperature, reducing accident consequences. The SG drain system prevents chemical attack on core graphite structures. The two last defence lines (4 & 5) focus on accident management, the limited operator action requirements, RCCS secondary coolant addition and emergency preparedness with off-site response.

(c) Containment System

The HTGR-POLA containment system is a reactor functional containment comprising of protective barriers that limit radionuclides transport and release. These barriers include the fuel kernel, TRISO layers, graphite, graphite blocks, primary system boundary, and confinement in reactor building. The primary barrier for fission product retention is the TRISO fuel.

(d) Spent Fuel Cooling Safety Approach / System

The spent fuel storage system is designed to remove decay heat from spent fuel blocks and other materials by storing them on storage racks and temporarily storing them in the reactor building while being cooled by pool water. The hot transport water is a pipe placed in the upper part of the pool and flows into the water cooling and purification system. The cooled water flows into the pool through a pipe placed in the lower part of the pool. After primary cooling period in the spent nuclear fuel storage in the reactor building, fuel is planned to be transferred to the spent nuclear fuel building with dry storage system.

6. Plant Safety and Operational Performances

Since the HTGR-POLA has not been constructed yet, there is no empirical experience in operational.

7. Instrumentation and Control Systems

The HTGR-POLA uses I&C systems divided into process instrumentation, logic controllers and supervision and control systems. These systems are designed to withstand the environmental conditions and meet operational, design and safety licensing objectives. The visualisation and control level includes workstations, Main Control Room and the Reserved Shutdown Station. The I&C automation level consists of three subsystems: Reactor Protection System, Safety Automation System, and Process System. The Reactor Protection System ensures safe shutdown in case of exceeding DBE, DEC, or Limiting Conditions of Operation. The Safety Automation System safeguards pivotal components like the steam generator, residual heat removal system and confinement insulation. The Process System regulates technological infrastructure and maintains process according to plant technical parameters. The ultimate goal is to reduce the probability of operator error and safety impact through ergonomic design principles and adequate time to react.

8. Plant Layout Arrangement

The plant is divided into a nuclear and conventional island, with the reactor building being part of the nuclear island, this part will be composed of the following buildings: reactor building, auxiliary building and spent fuel building. The conventional island, designed to convert energy includes a turbine, reboiler, water treatment system, switch yard, power output and electrical building.

9. Testing Conducted for Design Verification and Validation

The HTGR-POLA is a new design based on a proven technology and lessons learned from past designs of prismatic HTGRs. Reactor core physics, thermal-hydraulic and PSA analysis are undergoing. Tests of HTGR-POLA construction materials, equipment and safety are planned.

10. Design and Licensing Status

The basic design phase and selected chapters of the Preliminary Safety Analysis Report (PSAR) were completed in 2024. Next step is the finalising of the PSAR with site investigations required for licensing application. In parallel the detailed design phase will be launched.

11. Fuel Cycle Approach

The HTGR-POLA is designed for open fuel cycle with TRISO fuel. The primary fuel enrichment is 10-12% of ^{235}U, the maximum burnup is currently limited to 40 GWd/t due to fuel qualification program.

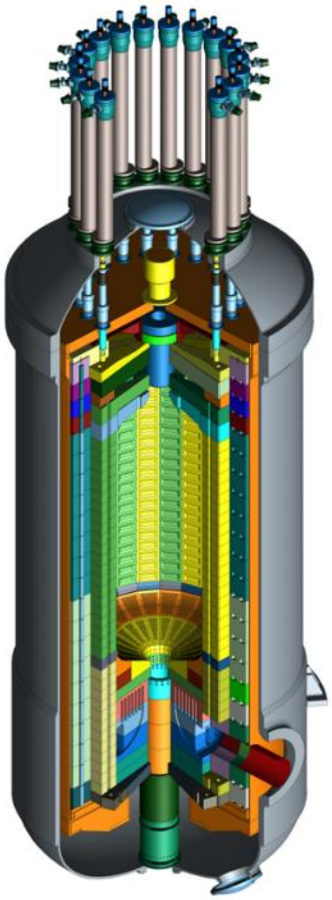
12. Waste Management and Disposal Plan

The plan for radioactive waste management includes solid, liquid, gas wastes, and spent nuclear fuel. Spent nuclear fuel will be temporarily stored on site, transferred to dry storage in the spent nuclear fuel building after ~2 years of primary cooling and irradiated reflector graphite blocks will be temporarily stored in the spent nuclear fuel building. Final disposal depends on the national plan.

13. Development Milestones HTGR-POLA

2012 – 2019	Preliminary studies and technological innovation (using available materials).	Complete
2019 – 2021	Pre-conceptual design phase and technology validation	Complete
2021 – 2022	Conceptual Design Phase	Complete
2022 – 2024	Basic Design Phase	Complete
2025 - 2028	Licensing Phase	Planned
2025 – 2028	Detailed Design Phase	Planned
2029 - 2032	Construction	Planned
2033	Commissioning	Planned

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



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	STL Nuclear (Pty) Ltd , 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 / 16
Core inlet/outlet coolant temperature (°C)	250 / 750
Fuel type/assembly array	TRISO particles in pebbles; LEU/Th
Number of fuel assemblies in the core	~150 000; around 125 to 150 pebbles/day throughput
Fuel enrichment (%)	10
Refuelling cycle (months)	Online fuel loading
Core Discharge Burnup (GWd/ton)	80-90
Reactivity control mechanism	Control Rods in the reflector
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m²)	5 000 (buildings only)
RPV height/diameter (m)	15.7 / 5.6 (flange outer diameter)
RPV weight (metric ton)	155
Seismic design (SSE)	0.3 g (generic site), 0.5 g under consideration)
Fuel cycle requirements/approach	Various options (see below)
Distinguishing features	No core meltdown, no active engineered safety systems
Design status	Basic 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.

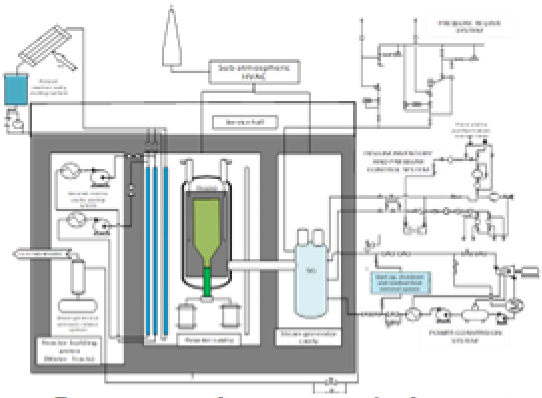
3. Design Philosophy

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.

4. Main Design Features

(a) Power Conversion

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.



Reactor core and power conversion layout

(b) Reactor Core

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.

(c) Fuel Characteristics

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 B4C absorber material, sandwiched between an inner and an outer tube segment. The inner tube segment allows cooling helium gas to flow form the top down in the circular channels. Each rod is equipped with a position indicator which measures the position of the rod over its entire positioning range and with position indicators for the upper and lower limit positions.

(f) Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel (RPV) is constructed 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.

(g) Reactor Coolant System

Helium enters the outer part of the co-axial duct and flows up the helium risers, it is then directed into the pebble bed core and removes heat as it flows downwards. The helium then collects in the lower plenum and is directed in the inner pipe of the co-axial duct back to the steam generator. Electric blowers are used to move the helium within the reactor pressure boundary. The secondary power conversion loop uses a conventional Rankine cycle and heat is transferred from the helium loop to the water loop to form steam which in turn drives the turbine.

5. 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) Decay Heat Removal System / 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) Spent Fuel Cooling Safety Approach / System

The spent fuel will be stored in special tanks of which the height/diameter is in excess of 4. Thermosiphons (heat pipes) are attached to the outside walls of these tanks which removes heat to the outside of the spent fuel area. The condenser ends of these heat pipes are then fitted with fins to dissipate the heat to atmosphere in an entirely passive way. There is also the possibility to use Stirling engine/generators on the condenser ends to generate electricity for charging batteries and in so doing provide a measure of energy conservation.

(d) Containment System

The primary fission product barrier is the SiC layer of the TRISO coated fuel particles, which keeps 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.

(e) Chemical Control

The probability of an in-leakage of water in case of a tube rupture in the steam generator is reduced by its vertical positioning relative to the hot parts of the reactor core. Water leaking from a ruptured tube will then accumulate in the shell of the steam generator and consequently will not be able to enter the core in quantities that will cause damage. In case of air ingress due to a break in the pressure boundary will be a slow diffusion process against the outflow rate of helium coolant. In addition, the construction of the coolant flow paths in and around the core structures is such that a naturally driven chimney type flow of oxygen is avoided inherently. Also, the grade of graphite used in the fuel and core structures is of such purity that self-propagating oxidation reactions are naturally not possible.

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control System

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.

8. Plant Layout Arrangement

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.

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.

9. Testing Conducted for Design Verification and Validation

Under design.

10. 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.

11. 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.

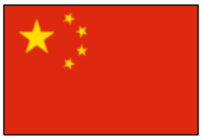
12. Waste Management and Disposal Plan

The HTMR100 fuel elements can be stored and disposed of a fuel sphere 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.

13. Development Milestones

2012	Project Started
2020	Preparation for pre-licensing application
2021	Concept design completed
2022	Basic design started

HTR-10 (Tsinghua University, China)



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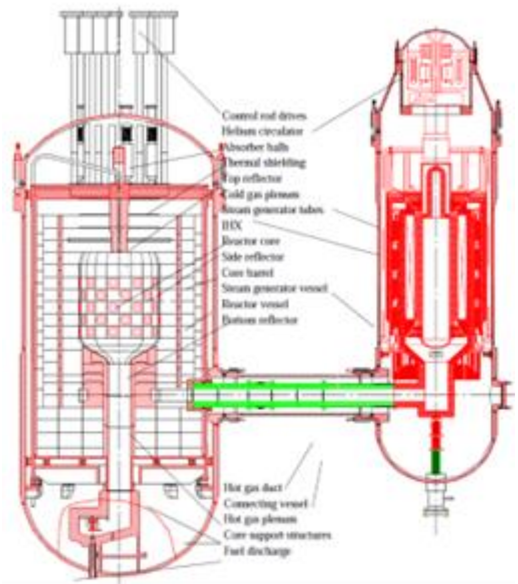


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	INET, Tsinghua University, People's Republic of China
Reactor type	Pebble bed modular high temperature gas-cooled test reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 / 2.5
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	3 / 4
Core Inlet/Outlet Coolant Temperature (°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 (m²)	100x130
RPV height/diameter (m)	11.1 / 4.2
RPV weight (metric ton)	167
Seismic Design (SSE)	3.3 m/s²
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.

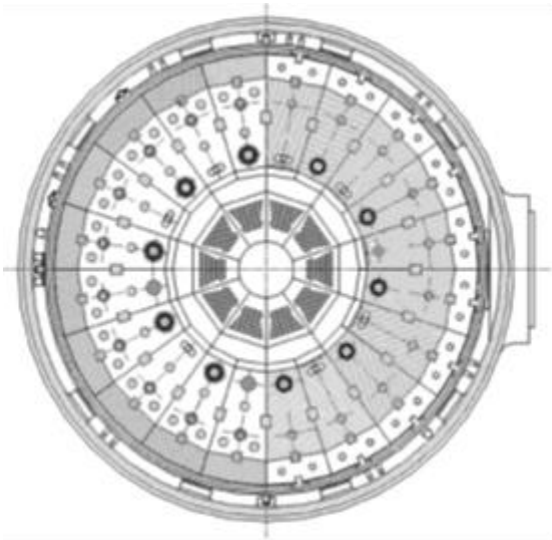
3. Design Philosophy

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.

4. Main Design Features

(a) 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.



(b) 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.

(c) 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.

(d) 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.

(e) 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.

(f) 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.

(g) 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.

(h) 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.

5. 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.

6. 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.

7. 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

8. 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.

9. Design and Licensing Status

HTR-10 is operational. The extension of operation license is undergoing.

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

To be included in the national plan of test facilities.

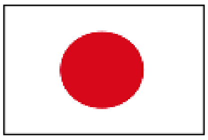
12. Research and Development Plan

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.

13. 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 distribution conducted.



HTR-50S (JAEA, Japan)

Plant layout (Phase II) [1]

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JAEA
Reactor type	Prismatic HTGR
Coolant/moderator	Helium / graphite
Thermal/electrical capacity, MW(t)/MW(e)	50/17.2 (max)
Primary circulation	Forced by gas circulator
NSSS Operating Pressure (primary/secondary), MPa	4 (abs)/12.5 (abs)
Core Inlet/Outlet Coolant Temperature (°C)	Phase I: 750 /325 Phase II: 900 /325
Fuel type/assembly array	UO_2 TRISO ceramic coated particle
Number of fuel assemblies in the core	33 fuel element / fuel block
Fuel enrichment (%)	5.9 / 9.4
Core Discharge Burnup (GWd/ton)	49
Refuelling Cycle (months)	730 days
Reactivity control	Control rod insertion
Approach to safety systems	Active and passive
Design life (years)	40
Plant footprint (m^2)	TBD
RPV height/diameter (m)	13.8
RPV weight (metric ton)	240
Seismic Design (SSE)	TBD
Distinguishing features	Multiple applications of power generation, cogeneration of hydrogen, process heat, steelmaking, desalination, district heating
Design status	Conceptual Design

1. Introduction

HTR50S is 50 MWt small-sized high temperature gas-cooled reactor (HTGR) for multiple heat applications with the reactor outlet coolant temperature of 750°C for the reference design. The reactor is designed to increase to outlet temperature up to 900°C.The design builds on using the existing technology and experience obtained in the HTTR construction and operation to minimize initial construction uncertainty and potential delay.

2. Target Application

The HTR50S is designed to provide both electricity and heat in remote or off-grid locations. It has a net power output of 17.2 MWe (Max.) and can supply 25 MWt of heat to non-electric application systems. The reactor supports efficient heat utilization, with up to77.0% heat utilization ratio in combined heat and power (CHP) configurations.

3. Design Philosophy

The HTR50S aims for potential near-term deployment. The following approaches are employed to meet the design strategy:

- The reactor outlet temperature was at 750°C to use Alloy 800H for steam generator heat transfer tubes, with a main steam temperature of 538°C.
- The reactor inlet temperature was set at 325°C, considering RPV materials like SA533B/SA508 with a service temperature limit of 371°C and accident limit of 538°C.
- The primary coolant pressure was set at 4 MPa, similar to HTTR, leveraging HTTR's experience and reducing RPV thickness for transportation.
- The fuel specifications match HTTR's to utilize existing knowledge and experience.
- Steam temperature/pressure was set at 538°C/12.5 MPa, aligning with Fort St. Vrain reactor conditions and existing small steam turbine capabilities.

4. Main Design Features

(a) Nuclear Steam Supply System

The HTR50S power generation system converts energy from steam generator (SG) into rotational energy via a steam turbine, which is then converted into electrical energy by the generator. The system includes a steam turbine, generator, condenser, degasser, feed water pump, feed water heater, piping, and valves. It utilizes a regeneration cycle of three-stage extraction to a deaerator and two high-pressure feed-water heaters. During summer operation, the power output is 17.2 MWe with an efficiency of 34.4%.

In winter, 25 MWt of heat is utilized for district heating, reducing power output to 13.5 MWe with an efficiency of 27.0%, but increasing the heat utilization ratio to 77.0% through local heat application.

(b) Reactor Core

The reactor core consists of stacked hexagonal graphite blocks, consisting 30 fuel columns and 7 control rod guide columns in the fuel region, and 18 replaceable reflector columns and 6 control rod guide columns in the reflector region, totalling 61 columns. The core is surrounded by a fixed graphite reflector block witha side shield and core restraint mechanism.Fuel elements (fuel rods) are loaded into graphite blocks with a height of 580 mm and a face-to-face distance of 360 mm. The coolant flows through designed channels to remove heat and maintain core temperature.

(c) Reactivity Control

It employs two independent reactor shutdown systems with: a control rod system and a back-up shutdown system using boron carbide and sintered graphite pellets. The control rod system is designed to ensure the core remains subcritical under all operational conditions. If control rods cannot be fully inserted, the backup system ensures the core remains subcritical to maintain safety

(d) Reactor Pressure Vessel and Internals

The reactor pressure vessel(RPV) supports the core through graphite and steel structures. The primary coolant enters at 325˚C, cooling various reactor components before flowing downward through the core. The upper plenum is equipped with a shroud made of low thermal conductivity material to prevent overheating due to thermal radiation.

(e) Reactor Coolant System

The reactor cooling system includes a primary cooling system that cools the reactor, a shutdown cooling system (SCS) for decay heat removal, and a Vessel Cooling System (VCS) thatremove decay and residual heat during abnormal transient and accident conditions.

A steam generator (SG) transfers heat from the reactor to a secondary system for steam production. In Phase II, with a 900°C reactor coolant outlet, an intermediate heat exchanger (IHX) will be added to generate gas turbine power and high-temperature steam, with potential applications in hydrogen production.

5. Safety Features

(a) *Engineered Safety System Approach and Configuration*

To ensure reactor safety, the HTR50S employs several key safety functions: reactivity control, heat removal from the core, and confinement of radioactive materials. A passive indirect cooling system, known as the VCS, effectively removes heat through radiation heat transfer to water-cooled panels on the bio-shielding concrete surrounding the reactor pressure vessel (RPV). The reactor's radioactive material containment is achieved with multiple barriers, including coated fuel particles, a graphite sleeve, the reactor coolant pressure boundary, and a reinforced concrete containment vessel (RCCV) lined with steel. Additionally, to manage chemical attack risks, the design limits the entry of air into the reactor post-depressurization. .

(b) *Safety Approach and Configuration to Manage DBC*

In order to ensure reactor safety under design basis conditions engineered safety features which have basic safety functions, namely reactivity control, heat removal from the core, and confinement of radioactive materials are employed. Plant behaviour analyses are performed for representative accident conditons and the analysis results demonstrated that evaluation items for the safety analysis met the evaluation criteria.

(c) *Safety Approach and Configuration to Manage DEC*

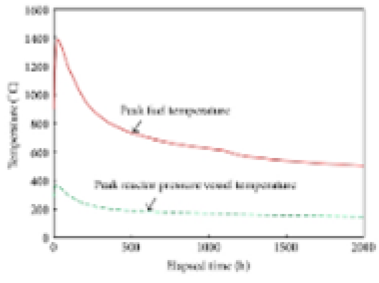
HTGR can intrinsically shutdown without exceeding fuel temperature limitation in case of loss of forced cooling accidents with failures of all reactivity control systems. Such cahracteristics are demonstrated by the HTTR safety demonstration tests. Because of the inherent safety characteristic of the HTGR, the amount of radionuclide released to environment would be small even in the design extention coditions which considers failures of engineered safety features.

(d) *Containment System*

The reactor containment vessel is one of the engineering safety facilities in the HTR50S, which is designed to prevent the release of radioactive materials to the outside and to ensure the safety of the general public and workers in the vicinity of the reactor facility. Isolation valves are installed in the piping that penetrates the reactor containment vessel in order to form a pressure barrier in the event of a double pipe rupture accident, etc. and to form a barrier against the release of radioactive materials. For the design of the containment vessel of the HTR50S, a RCCV was adopted from the viewpoint of reducing the amount of steel materials.

(e) *Spent Fuel Cooling Safety Approach / System*

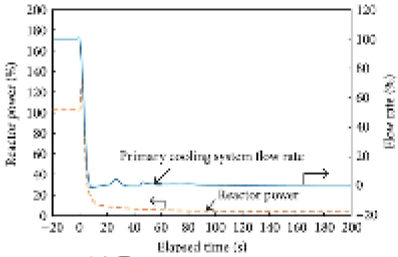
Spent fuel is transferred to the spent fuel storage facility in the reactor building by the fuel exchanger, and then transferred into the spent fuel storage building by the fuel in/out machine. The design for removing decay heat from spent fuel shall be air-cooled in the spent fuel storage building.



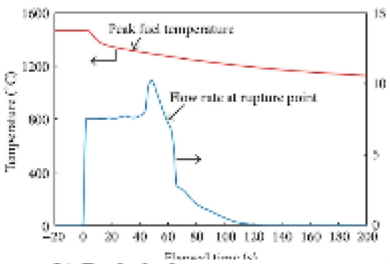
Depressurized accident behaviour [1]

6. Plant Safety and Operational Performances

During a depressurized accident, maximum fuel temperature peaks at 1386°C but falls to 500°C after 2000 hours, staying within safety limits. The maximum RPV temperature reaches 364°C but remains below critical thresholds. In a steam generator tube rupture scenario, the reactor scrams within a second. Reactor power briefly increases to 129%, with a fuel temperature rise of only 2°C. Steam and feedwater valves close after about 73 seconds, with 568 kg of water vapor entering the cooling



(a) Reactor power, reactor coolant flow rate [1]



(b) Peak fuel temperature, steam flow rate at rupture point [1]

system. Reactivity increase is 1.3×10^{-3} $\Delta k/k$, with sufficient shutdown margin. The primary cooling system flow rate drops due to stopped gas circulators, and coolant pressure boundary temperature stays stable.

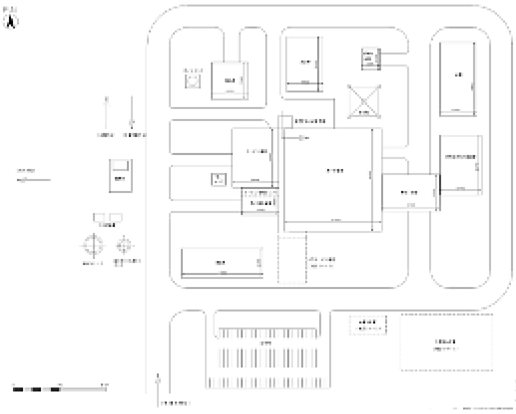
7. Instrumentation and Control Systems

The instrumentation and control system consists of reactor and process instrumentations, control systems, safety protection system, and engineered safety features actuating systems. The reactor protection system automatically shuts down the reactor by inserting control rods in response to signals from reactor and process instrumentation. It employs a two-train logic circuit that triggers reactor scram and interlock signals. Power is provided independently by an uninterruptible power supply, and system components are constructed from non-flammable or flame-retardant materials.

8. Plant Layout Arrangement

The steam turbine building is adjacent to the reactor building, aligned with the steam generator (SG), and the gas turbine building aligns with the intermediate heat exchanger (IHX) to minimize piping. The reactor building has four underground and two above-ground floors, measuring 53.0 m x 56.0 m, with heights of 32.2 m below ground and 24.8 m above. The steam turbine building is two stories, 25.0 m x 32.0 m, and 19.4 m high. Piping for steam, water, and helium is routed through 3 m wide trenches. Air coolers are installed on the north side of the reactor building to avoid interference with hydrogen production facilities and ensure proper ventilation.

Bird's eye view of HTR50S [2]



Site layout of HTR50S [2]

9. Testing Conducted for Design Verification and Validation

The test results using the HTTR will be utilized for the development of HTR50S. The test items cover fuel performance and radionuclide transport, core physics, reactor thermal hydraulics and plant dynamics, and reactor operations, maintenance, control, etc. However, an actual demonstration test is warranted for performance confirmation.

10. Design and Licensing Status

The design is developed at conceptual design stage. HTR50S design is based on the matured technology established through the HTTR design, construction, operation and maintenance. In addition, conventional technologies from other industries are also incorporated. The following approaches are employed to meet the design strategy:

11. Fuel Cycle Approach

HTR50S offers flexible fuel cycle options and can adopt the spent fuel management policy in the country to be deployed. For spent fuel reprocessing, methods include roasting and electrolytic deconsolidation to remove coated fuel particles. Various techniques such as the roller, jet mill, and rotating disk methods are used to remove coatings and recover uranium. The rotating disk method, in particular, has been employed successfully for uranium recovery from HTTR fuel. Additionally, HTR50S optimizes resource use by allowing up to 25 MWt of its heat to be utilized for district heating, achieving a heat utilization ratio of 77.0%.

12. Waste Management and Disposal Plan

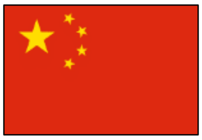
The design is applicable to options of direct disposal or reprocessing for spent fuel.

13. Development Milestones

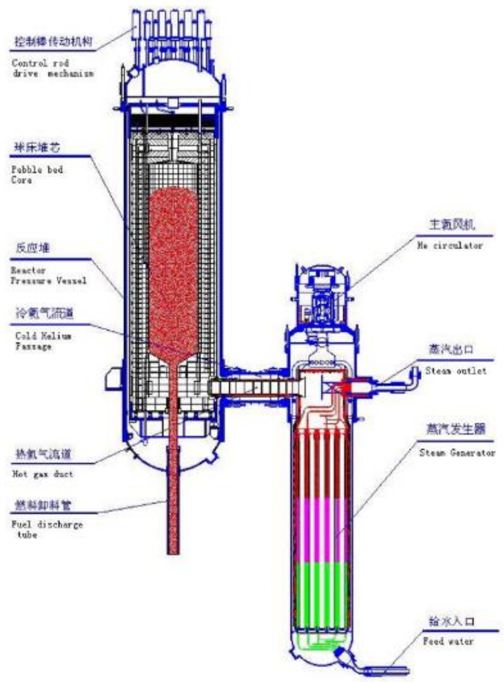
2010	Start of design	Complete
2013	Concept design completed	Complete
2017	Graphite irradiation test project	Complete
2019	High burnup fuel irradiation test project	Complete
2023	Monolithic fuel element specimen irradiation test project	Complete

Reference

1. H. Ohashi et al., “A Small-Sized HTGR System Design for Multiple Heat Applications for Developing Countries,” Int. J. Nucl. Energy, vol. 2013, p. 18, 2013.
2. H. Ohashi et al., “Conceptual design of small-sized HTGR system, 4; Plant design and technical feasibility,” JAEA-Technology 2013-016, 2013.



HTR-PM (Tsinghua University, China)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	**INET, Tsinghua University, China**
Reactor type	**Modular pebble bed HTGR**
Coolant/moderator	**Helium/graphite**
Thermal/electrical capacity, MW(t)/MW(e)	**2 × 250 / 210**
Primary circulation	**Forced circulation**
NSSS Operating Pressure (primary/secondary), MPa	**7 / 13.25**
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	**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)	**256 100**
RPV height/diameter (m)	**25 / 5.7 (inner)**
RPV weight (metric ton)	**800**
Seismic Design (SSE)	**0.2 g**
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	In operation

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 test 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.

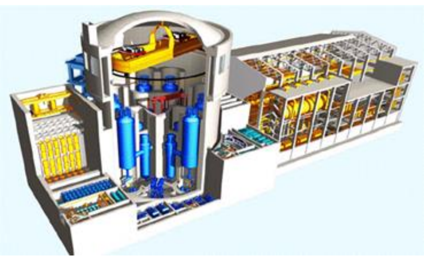
2. Target Application

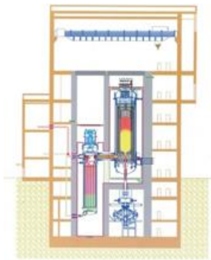
The HTR-PM is a commercial demonstration unit for electricity production. The twin reactor modules driving a single turbine configuration was specifically selected to demonstrate its feasibility. Following the HTR-PM demonstration plant, commercial deployment of HTR-PM based on batch construction is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

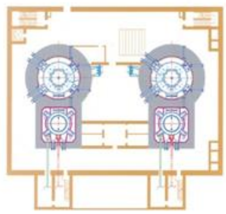
A standard design has been finished for the 600 MW(e) multi-module HTR-PM600 nuclear power plant, which consists of six reactor modules or Nuclear Steam Supply System (NSSS) modules coupling to one steam turbine. Each NSSS module has the same design as the HTR-PM demonstration plant, with independent 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. Future sites have been identified for possible deployment. There is a big market demand for high temperature steam supply from HTR-PM/HTR-PM600 in China.

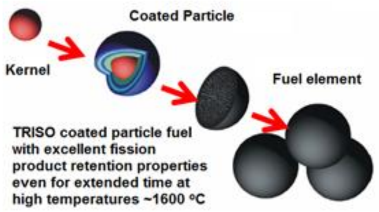
3. Design Philosophy

The HTR-PM consists of two NSSS modules coupled with a 210 MW(e) steam turbine, as shown below. Each NSSS module includes a reactor that contains reactor pressure vessel, graphite, carbon, and metallic reactor internals; a steam generator; a main helium blower; and a hot gas duct. 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.









4. Main Design Features

(a) 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.

(b) 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 ^{235}U. 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.

(c) 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 device 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.

(d) 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 graphite 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.

(e) 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.

5. 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. And air ingress accident is classified as design extension condition.

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 is practically eliminated (small pipes, connection vessel; no chimney effect). After water ingress accident, the way to remove humidity from primary circuit are provided.

6. Plant Safety and Operational Performances

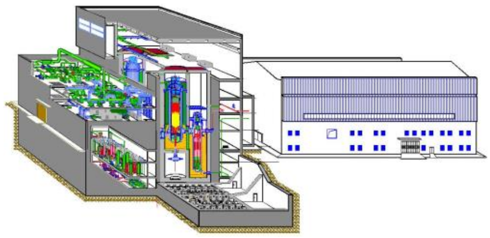
The HTR-PM demonstration power plant is under final commissioning test phase. Due to online fuel loading mode of HTR-PM, better availability factor can be expected compared with other power plants operating in a mode of periodic fuel loading.

7. Instrumentation and Control Systems

The instrumentation and control system of HTR-PM is similar to those of normal PWR plant. The two reactor modules are controlled in a coordinated manner to meet various operational requirements.

8. Plant Layout Arrangement

The nuclear island contains reactor building, nuclear auxiliary building, spent fuel storage building and Electric 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.



9. Design and Licensing Status

The preliminary safety analysis report (PSAR) was reviewed by the licensing authorities during 2008-2009. The Construction Permit was issued in December 2012. Final approval of the FSAR is achieved in July 2021, and Operation License was obtained in August 2021, followed by fuel loading, criticality and power operation.

HTR-PM Main Control Room

10. Fuel Cycle Approach

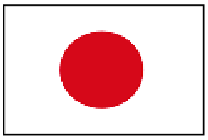
The air-cooled spent fuel canisters are placed in the spent fuel storage building with concrete shields. The canister can be placed in a standard LWR transport cask and be transported if necessary. HTR-PM currently adopts open fuel cycle. After the intermediate storage of spent fuel elements, the final 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).

11. 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. This waste can be underwent a final storage after conditioned.

12. 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
2021.8	Operation License
2021.9	First criticality
2021.12.20	Grid connection
2022	Full power operation



HTTR (JAEA, Japan)

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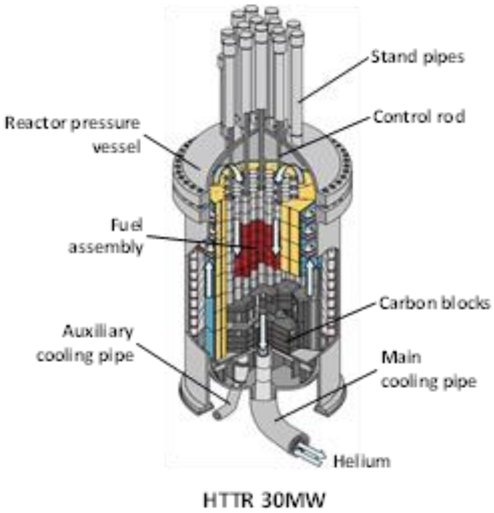
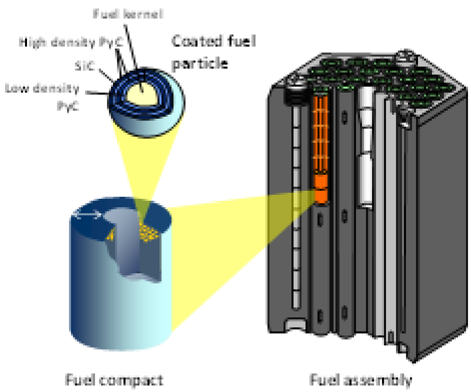




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JAEA in cooperation with MHI, Toshiba, IHI, Hitachi, Fuji Electric, NFI, Toyo Tanso, Japan
Reactor type	Prismatic HTGR
Coolant/moderator	Helium / graphite
Thermal power, MW(t)	30
Primary circulation	Forced by gas circulators
Primary coolant pressure, MPa	4
Core Inlet/Outlet Coolant Temperature, °C	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	In operation

1. Introduction

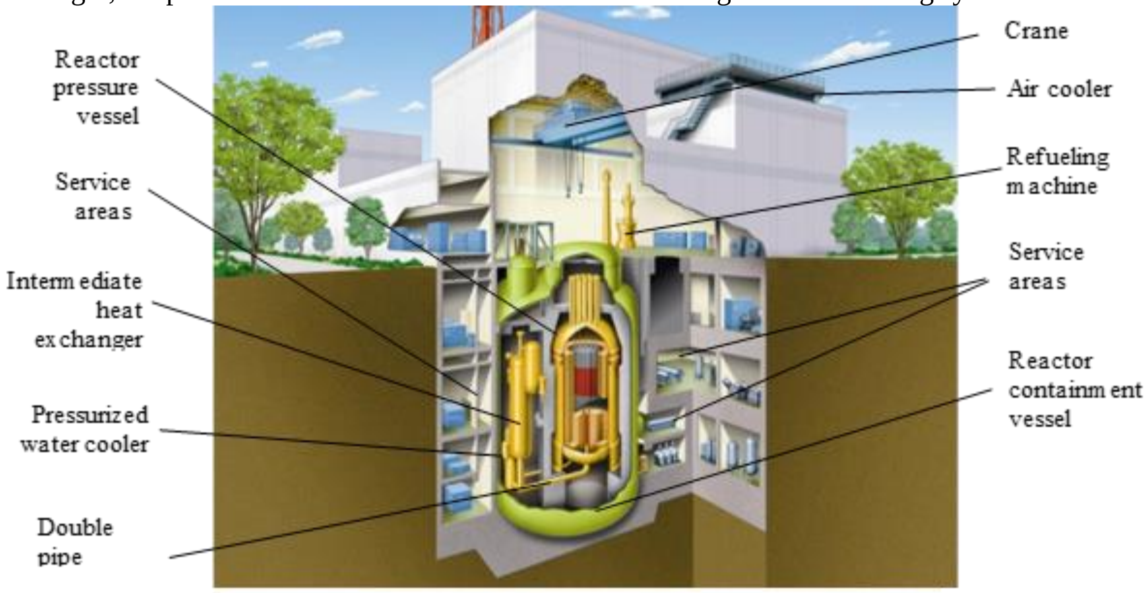
The High Temperature Engineering Test Reactor (HTTR) is Japan's first High Temperature Gas-cooled Reactor (HTGR) established in the Oarai Research and Development Institute of Japan Atomic Energy Agency (JAEA). The HTTR has superior safety features by using coated fuel-particle, graphite moderator, and 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 100% power) to demonstrate the inherent safety feature of HTGR in 2024. 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. 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.

4. Main Design Features

(a) 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.

(b) 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.

(c) 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.

(d) 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.

5. 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.

6. 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.

7. 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.

8. 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.

9. 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. In conformity with new regulatory requirements by Nuclear Regulation Authority of Japan, JAEA received permission and restarted operation of the HTTR on July 2021.

10. 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 850°C; Safety demonstration test
2004	Reactor outlet coolant temperature of 950°C
2007, 2010	850°C/30 days operation; 950°C/50 days operation and Safety demonstration tests
2014	Conformity review on the New Regulatory Requirements start toward resume operation
2020	Permission toward the restart of HTTR
2021	Operation restarted
2022, 2024	Safety demonstration tests

11. Future plans

New test project to demonstrate hydrogen production by 2030 using the high temperature heat from the HTTR has been launched. Other than these, a number of activities are planned to be carried out through operation of the HTTR including international cooperation and human-resource development. 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.

Reference

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

MHR-100 (JSC "Afrikantov OKBM", Russian Federation)

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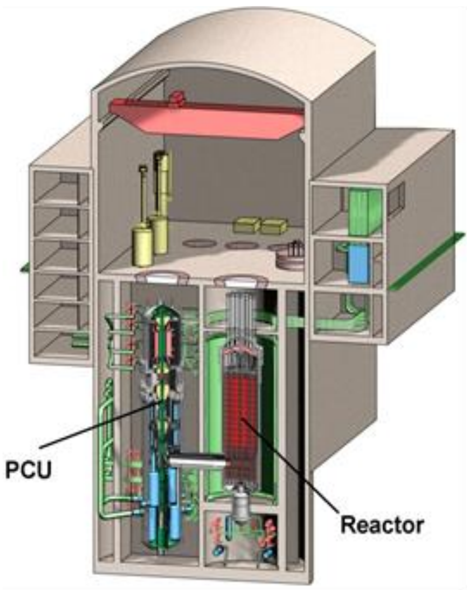


MHR-100 GT layout

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 (%)	< 20%
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. 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.

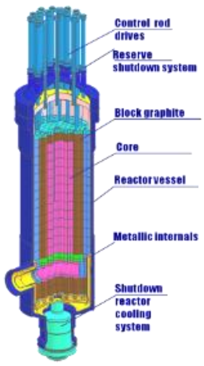
4. Specific Design Features

(a) Power Conversion Unit

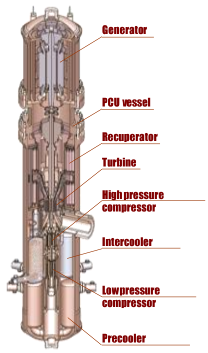
A power conversion unit is integrated in a single vessel and includes a vertical turbomachine, highly efficient heat exchanger, and coolers. The turbomachine includes a generator and turbo-compressor mounted on a single shaft on electromagnetic suspension. Gas turbine cycle of power conversion with the helium turbomachine, heat exchanger and intercooler provide thermal efficiency at 48%.

(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 × 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) 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.

(d) 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.

5. 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.

6. 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.

7. 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.

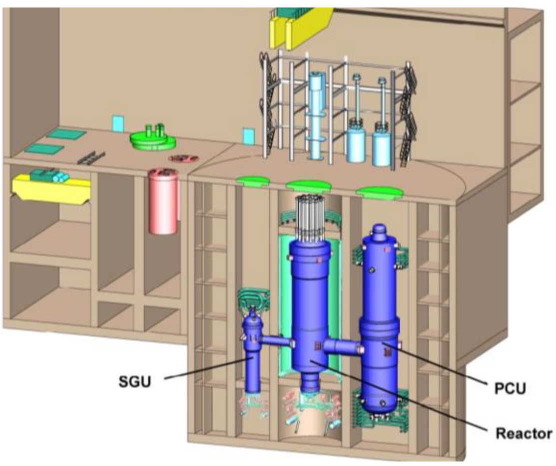
8. Design Variants and Plant Arrangements Based on the Modular MHR-100

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.

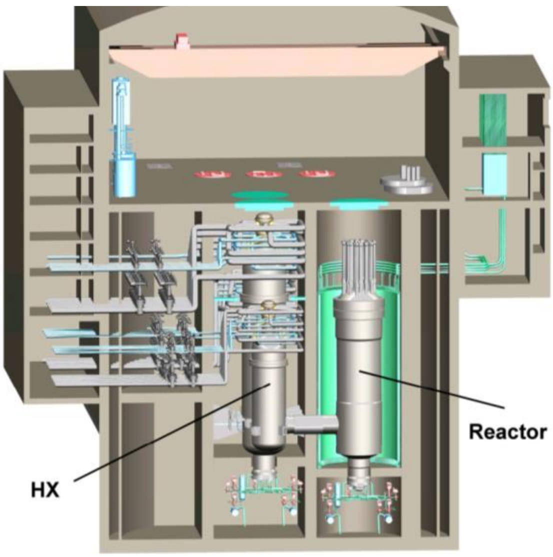


MHR-100 SE

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

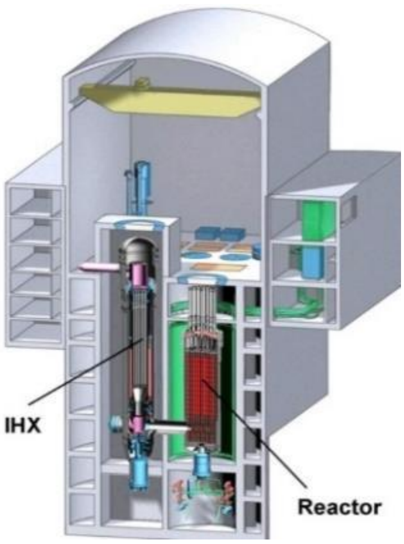


MHR-100 SMR

9. 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

10. 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.

11. 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.

12. 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

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

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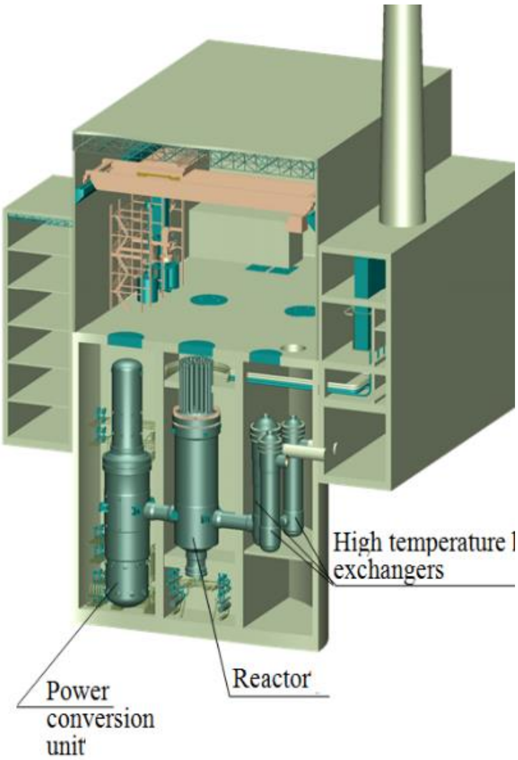


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, Russian Federation
Reactor type	Modular helium high-temperature reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	4 × 600 / 4 × 205.5
Primary circulation	Forced 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	~ 1 020
Fuel enrichment (%)	< 20
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. 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:

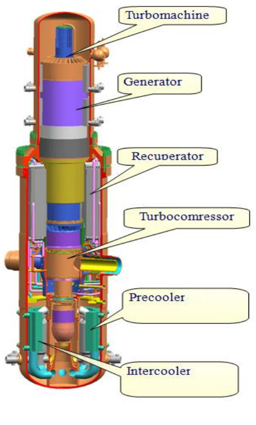
- Modular helium-cooled reactors with typical high level of inherent safety;
- Fuel cycle based on uranium dioxide in the form of multi-layer coated particles, high burnup and burial of fuel blocks unloaded from the reactor without any additional processing;
- Electromagnetic bearings operating almost 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.

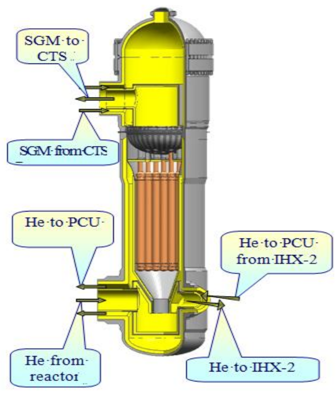
4. Main Design Features

(d) 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 provide a thermal efficiency of 48%.

(e) 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.

(f) 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.

(g) 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.

(h) 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.

5. 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.

(i) 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.

(j) 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).

(k) 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).

6. Plant Safety and Operational Performances

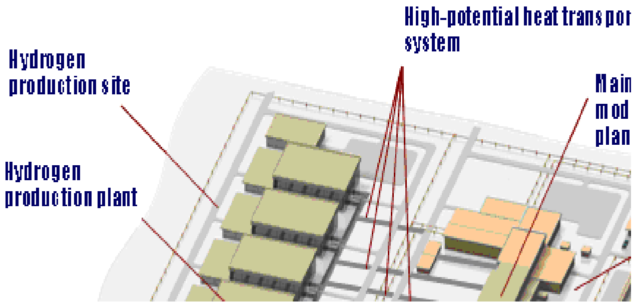
All safety systems are designed with two channels. Fulfilment of the regulatory requirements on safety, proven by a compliance with both deterministic and probabilistic criteria, is secured by an exclusion of the active elements in a channel or by applying the required redundancy of such active elements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

7. 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.

8. 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.



9. Design and Licensing Status

Feasibility study of plant application for large-scale hydrogen production completed. Currently, safety issues are the major area of R&D with the emphasis on mutual influence of nuclear and hydrogen production components of the facility.

10. 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.

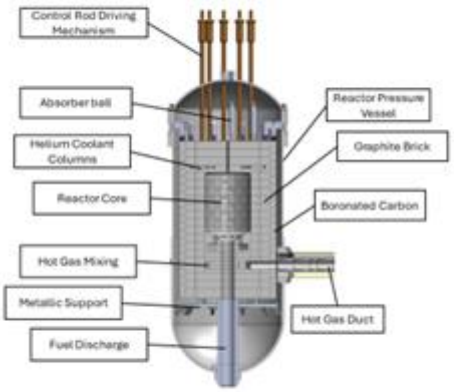
11. Waste Management and Disposal Plan

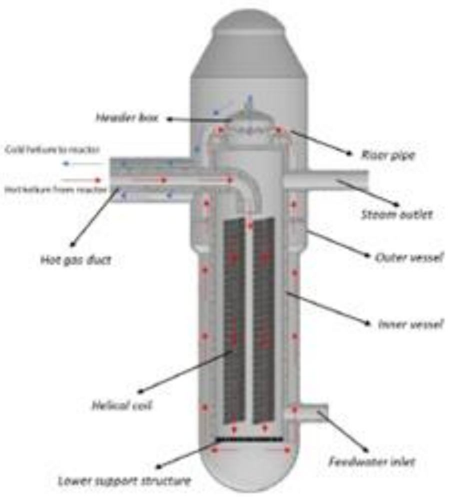
Facilities for long-term storage of spent nuclear fuel (SNF) and solid/solidified radioactive waste (RW) are included in the complex of an MHR-T commercial 4-unit NPP. The capacity of the designed SNF storage is determined from the condition of capability to store fuel unloaded from the NPP for 10 years. 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.

12. 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

PeLUIt-40 (BRIN – ITB, Indonesia)





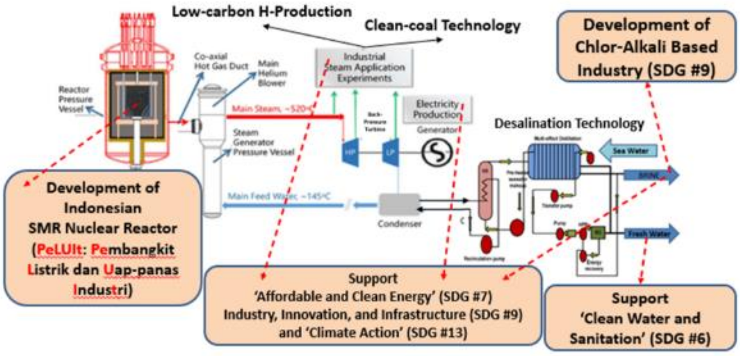
MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	National Research and Innovation Agency (BRIN) and Bandung Institute of Technology (ITB), Indonesia
Reactor type	Pebble bed high-temperature gas-cooled reactor
Coolant/moderator	Helium/graphite
Thermal Power	30MWt / 10MWe
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 Target (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 the plant
Distinguishing features	Inherent safety, no need for offsite emergency measures
Design status	The basic design of 30MWt/10MWe is in progress.

1. Introduction

PeLUIt stands for *Pembangkit Listrik dan Uap-panas Industri* (Indonesian) that means Nuclear Power Plant for Cogeneration of Electricity and Industrial Heat. Currently, main developer of PeLUIt-40 design is National Research and Innovation Agency (BRIN) and Bandung Institute of Technology (ITB). In Indonesia, the word ‘Peluit’ means a ‘whistle’ which represents the spirit and motivation to start the nuclear power plant era in Indonesia. Its initial name was *Reaktor Daya Eksperimental* (RDE). RDE was one of the national programs that supported the National Medium Term Development Plan (RPJMN) for 2015-2019. The main goal of the RDE development Programme was to build the national capability to develop a nuclear reactor technology by mastering the design, construction project management, commissioning, and operation of a nuclear power reactor. This goal is now continued with the PeLUIt-40 development. Furthermore, the nuclear reactor type selected for the program should become the prototype design to be commercialized to contribute to 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 this programme. A sound safety feature, optimum fuel utilization

and flexible applications of the PBR technology are among the reasons for this decision. PBR has a high safety level shown by a small radioactive release to the environment in any probable accident.
As part of the licensing procedure, BRIN (previously BATAN) already received the RDE Site Licensing from the National Nuclear Regulatory Body (BAPETEN) in January 2017. The design approval phase started in 2018 and 2019. However, the licensing process was halted due to a change in the national main policy of nuclear reactor development. Although the research and development of HTGR is continue, particularly related to safety analysis and its cogeneration potential. As the PeLUit-40 initiated and design development continue, recently communication with BAPETEN is already started looking for the opportunity to start design approval in late 2024 based on graded approach.

The general scheme of the PeLUIt system is shown in the figure below. The main components of the nuclear island system are the reactor pressure vessel (RPV) and its internal, coaxial hot gas duct, the main steam blower, and the steam generator pressure vessel. Parts of the RPV internals are shown on the right side of the figure below, such as the core and the graphite reflector surrounding the core. While uprating the power level to 40MWt, it is targeted that the core and RPV dimensions are kept.



PeLUIt Schematic System and Targeted Impact on Sustainable Development Goals

2. Target Application

The PeLUIt, a small modular reactor (SMR) type cogeneration reactor, targets two primary applications.
Currently, there are two target applications of PeLUIt as an SMR type cogeneration reactor. First, it aims to support Indonesia's de-dieselization by replacing costly diesel generators in remote areas with 10 MWe reactors. PLN, the national utility company, has a robust program for de-dieselization. A total 5200 diesel unit in in 2130 location with total capacity of 2.37GWe is planning to be replaced in the de-dieselization program. Second, it seeks to contribute to hydrogen production, aligning with global energy transition trends. Collaboration with a state-owned energy company PLN Nusantara Power is underway to develop the design and potential demo plant to prove the constructability, safety operation and maintenance, and its capability to be coupled to Solid Oxide Electrolysis Cell (SOEC) to produce a 172 kg/h low carbon hydrogen. To have the economic feasibility of the above utilization the power level is uprated to 30MWt/10MWe from the initial 10MWt/3MWe of RDE. Although, initial reactor physics safety shows the capability of PeLUit to be upgraded to 40MWt.
Another target utilization is for desalination as shown in the above figure of PeLUIt.

3. Main Design Features

(a) Design Philosophy

The PeLUIt is designed based on an established pebble-bed reactor principle. The design employs the TRISO-based fuel which provides 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 of graphite which gives good heat transfer and neutronic features. High heat conductivity and capacity of the graphite improve the heat transfer characteristic of the design. It helps to improve the thermal neutron spectrum of 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 are 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 in the range of 10MWt – 40MWt. Currently, main option which represent PeLUIt-40 is one with 30MWt or 10MWe power level. Each power level options have a different pebble fuel consumption. The helium temperatures at the reactor core inlet/ outlet are 250/700°C, and the steam parameters are 6 MPa/520°C at the steam turbine entrance. The nuclear island is coupled with a ~13MWe or 3 MW(e) steam turbine and additional heat exchanger to accommodate coupling with cogeneration

(b) Reactor Core and Power Conversion Unit

The primary helium coolant works at a pressure of 3.0 MPa. 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 flows 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 rod 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 feed water in the steam generator to have a superheated steam of 520ºC at 6 MPa flowing to the turbine to generate a 10MWe to ~3MWe. In the PeLUIt-40 with power level of 10MWe, a specific turbine is chosen to have the possibility of extracting the steam in medium/intermediate pressure to be directed to the heat exchanger and further to SOEC system.

(c) Fuel Characteristics

Fuel elements are spherical ones. Every fuel element contains 5 g of heavy metal. The equilibrium core has a 17% enrichment of ^{235}U. 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. Up to now, in the PeLUIt development the fuel is limited to the fuel which already tested and manufactured. In particular, the same fuel design as in the HTR-10 design is chosen.

(d) Fuel Handling System

The operation mode of 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 singular. 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 have already reached the burnup target will be collected in the spent fuel cask while the other will be redirected back into the core. For the multipass fuel scheme option, a 5-pass scheme is applied. A Once-Through-Then-Out (OTTO) fuel management scheme is considered as the main option compared to multi-pass. A simpler design that finally improves its techno-economic performance.

(e) Reactor Pressure Vessel and Internals

The primary pressure envelope of 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 the vessel portion, closure head, and nozzles. The RPV internals include the ceramic internal and metallic internal, as well as the control rod and control rod drive mechanism. The metallic internal includes the core barrel with guides and supports, the lower structure with a bottom plate, and the top thermal shield. The ceramic internal includes all the bottom, side, and top reflectors also the outer carbon bricklayer.

4. Safety Features

The PeLUIt incorporates the established safety features of pebble bed typt HTGR design as follows: (1) Maximum temperature of the fuel is below 1600°C in any condition, even in the most hypothetical accident; (2) With that maximum temperature, the TRISO-based fuels contain 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, and low excess reactivity (due to online refuelling). Passive cooling safety features are supported by the physical properties of the graphite which is the dominant material in the core and by the low diameter design of the core.

(a) Engineered Safety System Approach and Configuration

The PeLUIt employs a standard engineering safety system of the pebble bed type HTGR. This engineered safety system functions to localize, control, mitigate and terminate accidents and to maintain radiation exposure levels to the public below applicable limits. It applied the principles 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

PeLUIt is equipped with two (2) independent reactivity control or shutdown systems, 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 low excess reactivity.

(c) Reactor Cooling Philosophy

In normal operation, the core is cooled by the helium coolant flowing to the reactor 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 operates 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

The containment capability of the PeLUIt design is based on a multi-barrier system. The TRISO-based fuel design, particularly the SiC layer, acts as the first barrier that maintains almost all of the fission product. From the previous HTR-fuel test, in particular the HTR-10 fuel design which was adopted in PeLUIt, the fuel was 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 the engineered safety feature of the design.

5. Plant Safety and Operational Performances

The PeLUIt is not yet constructed so there are no empirical safety and operational performances. However, it is expected that due to online refuelling features 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 PeLUIt.

6. Instrumentation and Control Systems

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

7. Design and Licensing Status

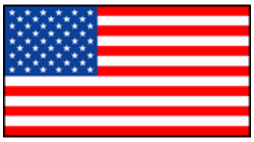
For the uprated 40MWt PeLUIt including the hydrogen production system is still in progress of working the basic design. While for the initial 10MWt RDE, the Site License was already issued by the Nuclear Energy Regulatory Agency (BAPETEN) in 2017. Although initial processes of Design Approval were already started in 2018 and 2019, the licensing of 10MWt RDE was halted. Communication with BAPETEN is already started looking for the opportunity to start design approval in late 2024 based on graded approach.

8. Fuel Cycle Approach

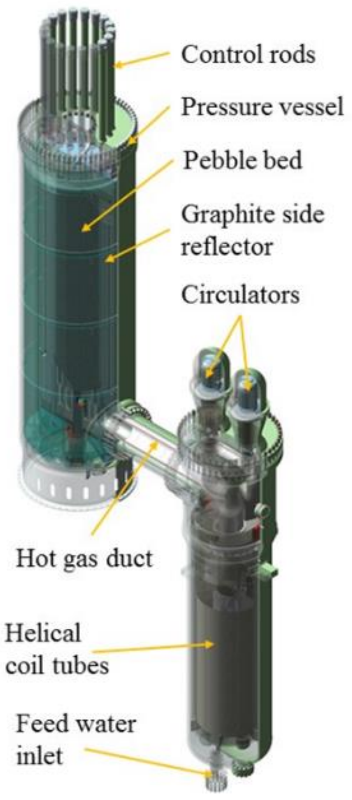
In the adopted fuel cycle, the pebble fuels that already reached their burnup limit are collected in the spent fuel casks. These spent fuel casks are placed in the spent fuel storage room in the reactor building. The spent fuel storage room is equipped with a radiation and temperature monitoring and control system. The casks and the room are designed to avoid the 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.

9. Development Milestones

2024	Start design approval at the Indonesian Regulatory Body (BAPETEN)
2026	Site Licensing
2027	Start Construction and Commissioning Licensing.
2028	Start construction of the first full-scale NPP.
2031	Initial operation.



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 (^{o}C)	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

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. 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.

4. Main Design Features

(a) 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.

(b) 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.

(c) 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.

(d) 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.

(e) Reactor Pressure Vessel and Internals

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

5. 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.

6. 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.

(d) 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

(e) 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.

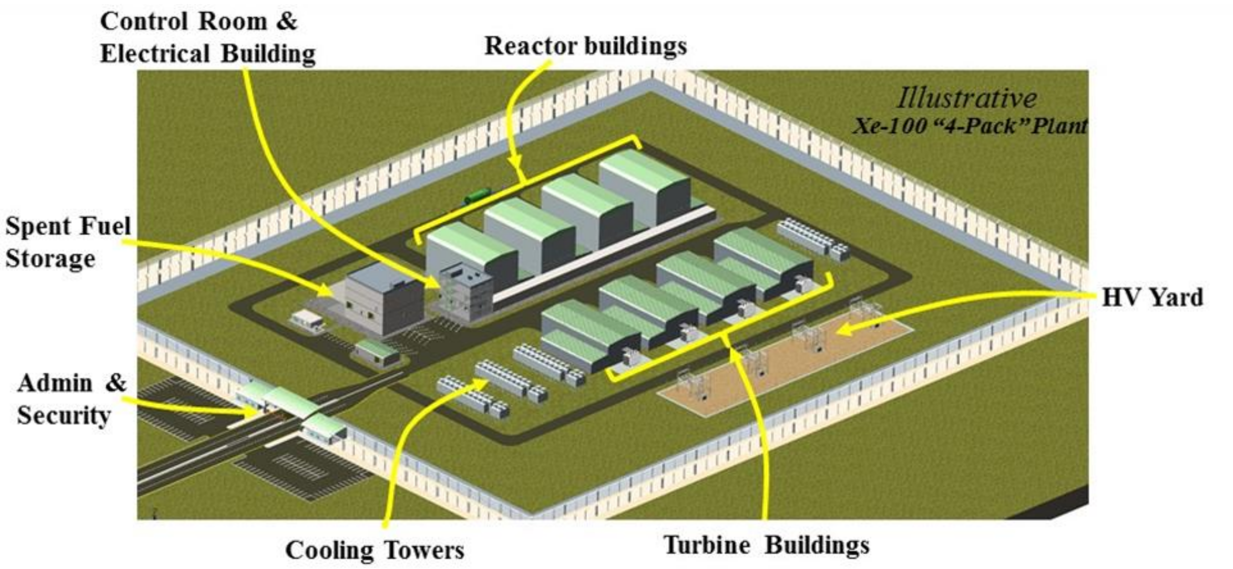
(vi) Worst accident scenario and release:

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.

7. 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.

8. Plant Layout Arrangement



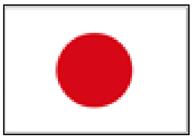
9. Design and Licensing Status

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

10. 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

LIQUID METAL COOLED FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS



4S (Toshiba Corporation, Japan)

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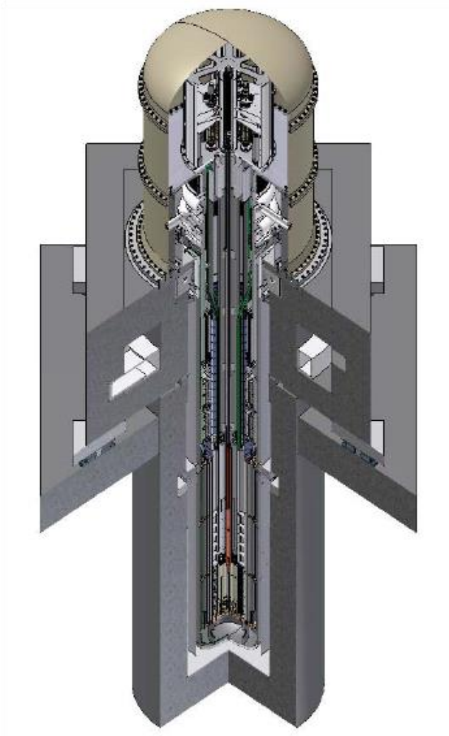


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Toshiba Energy Systems & Solutions Corporation, Japan
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant	Sodium
Thermal/electrical capacity, MW(t)/MW(e)	30 / 10
Primary circulation	Forced convection
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) enriched uranium
Number of fuel assemblies in the core	18
Fuel enrichment (%)	< 20%
Refuelling cycle (months)	N/A
Core discharge burnup (GWd/ton)	34
Reactivity control mechanism	Axially movable reflectors / fixed absorber
Approach to safety systems	Hybrid passive and 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
Fuel cycle requirements/approach	Either one-through scheme or closed fuel scheme is applicable
Distinguishing features	Core lifetime of ~30 years without on-site refuelling, 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. 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 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;
- 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. Design Philosophy

The 4S reactor is an integral pool type with all the primary components installed inside the reactor vessel (RV). 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.

4. Main Design Features

(a) Power Conversion

The steam turbine generator is used for converting nuclear power to electricity. 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.

(b) 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 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 consists of fuel slugs of U-Zr alloy, bonding sodium, cladding tube, and plugs at both ends. 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).

(c) 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.

(d) Reactor Pressure Vessel and Internals

The RV houses all the major primary components (integral type) including the IHX, the primary EM pumps, the moveable reflectors which form a primary reactivity control system, the ultimate shutdown rod which is a back-up shutdown system, radial shielding assemblies, core support plate, coolant inlet modules and fuel subassemblies. The RV provides a primary boundary for the primary sodium coolant, and is designed with a pressure/temperature of 0.3 MPa/550°C. The design lifetime of the RV is 60 years as well as the other components.

(e) 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.

(f) 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.

(g) Steam Generator

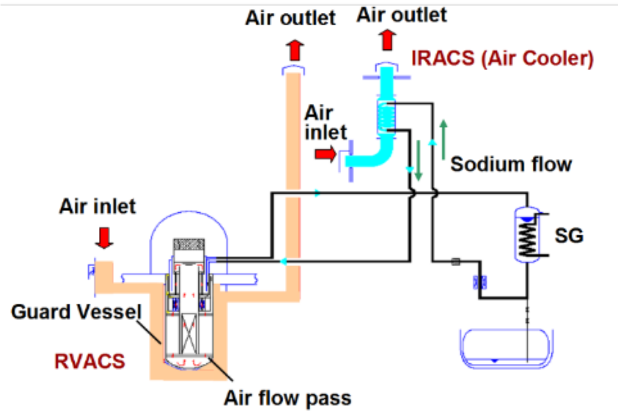
The 4S adopts a once through type double-wall tube SG with failure detection systems. The heat transfer tube of the SG is a double-wall type. Between the inner and outer tube, wire meshes are installed, which are filled with helium, to detect one side tube failure prior to failure of the other side tube. It enables to prevent sodium-water reaction.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

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. 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 shut down the reactor.

(b) Decay Heat Removal System / Reactor Cooling Philosophy



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) Spent Fuel Cooling Safety Approach / System

Spent fuel after 30 years of operation is cooled in the reactor vessel for one year and temporary stored in dry cask for the 10MWe-4S. Dry cast is cooled by natural convection of air. No need spent fuel pool is required.

(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.

(e) Chemical Control

4S does not have chemical control system.

6. Plant Safety and Operational Performances

The 4S is designed to operate safely without active involvement of the plant operators. The design features to support such operation 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.

7. Instrumentation and Control System

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.

8. 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 considerations and to enhance protection against extreme external events. 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 two class 1E buses are separated from each other and separated from the non-class 1E electric system. Each class 1E system is provided with a separate emergency diesel generator and batteries.

9. Testing Conducted for Design Verification and Validation

Many equipment, fuel and core tests have been performed to demonstrate the separate aspects of the 4S design. These tests have been performed both in Toshiba's new sodium test loop facility in Yokohama, as well as supporting locations worldwide. The following experimental tests for validation were performed:

- A critical experiment for the nuclear design method of a reflector-controlled core with metallic fuel.
- The pressure drop for a fuel subassembly has been confirmed by fuel hydraulic testing.

- The scale test of the reflector drive mechanism with fine movement has been performed.
- Heat transfer characteristics between the vessel and the airflow of RVACS have been confirmed.
- The heat transfer characteristics of the steam generator have been confirmed by sodium testing.
- Manufacturability investigations, electromagnetic oscillation testing and sodium flow testing of the EM pump completed.

10. Design and Licensing Status

Licensing activities for the 4S design initiated with the U.S. Nuclear Regulatory Commission (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 U.S. NRC. Toshiba is conducting the detailed design and safety analysis for design approval. In parallel, Toshiba Energy Systems & Solutions continues to look for customers.

11. Fuel Cycle Approach

The 4S 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. In the case of once-through fuel cycle scheme, spent fuel after 30 years' operation is cooled in the reactor vessel for one year and temporary stored in dry cask for the 10MWe-4S. Then, it is eventually shipped to a permanent repository.

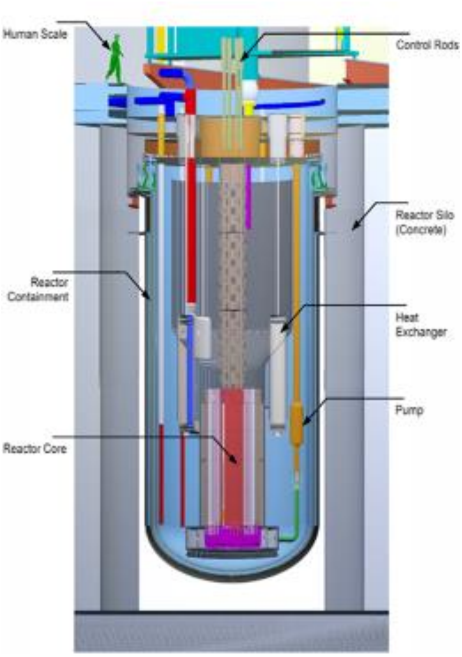
12. Waste Management and Disposal Plan

Decommissioning at the end-of-life was evaluated such as sodium deposition and reactor vessel deposition can be done by following decommissioning method of EBR-II and LWR plant in the US. Sodium will be disposed by following experiment of EBR-II. The RV without fuel and sodium will be filled with concrete and transported to disposal site.

13. Development Milestones

2007	Licensing activity for the 4S design initiated with the 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

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



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ARC Clean Energy, Canada
Reactor type	LMFR (pool type)
Coolant	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
Refuelling Cycle (months)	240
Core Discharge Burnup (GWd/ton)	77
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)	16.7 / 7.9
RPV weight (metric ton)	600
Seismic design (SSE)	0.3 PGA
Fuel Cycle requirements/approach	Metallic HALEU/Open cycle
Distinguishing features	Inherent safety with passive, diverse and redundant decay heat removal. Core lifetime of 20 years without refueling.
Design status	Preliminary design

1. Introduction

The ARC-100 is an advanced SMR 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, hydrogen production and water desalination. Applications.

3. 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 defence 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 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.

4. Main Design Features

(a) Power Conversion

The power conversion system of the ARC-100 consists of a sodium to water superheated steam generator that powers an air cooled non-reheat turbine generator set capable of producing 115 MWe.

(b) 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.

(c) 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.

(d) 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, principally the intermediate heat exchanger. A 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.

(e) 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. 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 forced sodium circulation within the reactor.

(f) 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).

(g) 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.

5. 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: (i) 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); (ii) The systems provide 'defence-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 / Reactor Cooling Philosophy

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: (i) Direct Reactor Auxiliary Cooling System (DRACS); (ii) 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) Spent Fuel Cooling Safety Approach / System

Used fuel assemblies are temporarily stored in-vessel at the outside of the core. Once sufficiently cooled, the used fuel assemblies are periodically extracted and placed into an on-site dry-cask storage.

(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. Damage to the 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. 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.

(e) Chemical Control

The ARC-100 sodium coolant and argon cover gas are continuously monitored. Both the primary and secondary sodium coolant are cleaned using sodium cold traps. The primary sodium can also be directed to a radionuclide trap in case of fuel pin failure.

6. 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 behaviour 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.

7. Instrumentation and Control Systems

The fail-safe safety-related shutdown Distributed Control and Information System (DCIS) is a three-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. The DCIS operates at low voltage as the fail-safe shutdown systems are designed to operate without electricity.

8. Plant Layout Arrangement

(a) Reactor Building

The Reactor Building is a cylindrical building made of reinforced concrete floors and walls. Roof trusses and their supporting columns are made of structural steel. The Reactor Building Structure houses the Primary Reactor System, reactor support and safety systems. The refuelling floor of the Reactor Auxiliary Building Structure includes the refuelling 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 steam turbine is a non-reheat, air cooled, single shaft, single casing turbine with separate HP/IP/LP sections. steam turbine with a single flow high pressure turbine and combined intermediate and low-pressure turbine. 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 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.

9. Testing Conducted for Design Verification and Validation

An extensive suite of analysis codes compiled over the 30 years of developmental history of sodium fast reactors provide a comprehensive theoretical and applied basis for the ARC-100 reactor. Efforts are in progress to Verify and Validate the codes.

10. Design and Licensing Status

Based on the regulatory requirements applicable to new build projects in Canada, Arc Clean Energy has successfully completed the CNSC VDR I process. Arc Clean Energy is now completing preliminary design while progressing through VDR phase II. The site safety assessment, environmental impact assessment, and PSAR have started.

11. Fuel Cycle Approach

The ARC-100 core uses HALEU in metallic form alloyed with zirconium. The reference plan for the FOAK demonstration unit is an "open fuel cycle" where spent fuel would be sent to a Deep Geological Repository (DGR). When the reactor is refuelled after 20 years, spent fuel will be moved out of the core region and eventually sent to on-site dry storage. After 60 years of operation and two refuelling cycles, all of the spent fuel can be transferred to a DGR. The ARC-100 will only produce about 300 spent fuel assemblies over its 60-year design life. The ARC-100 reactor is capable of closing the fuel cycle by recycling its own metallic spent fuel. Given the 20-year fuel cycle and a suitable decay period, recycling the first core load would occur about 50 years after that initial load. The pyroprocessing of spent metallic fuels was developed by ANL and demonstrated during EBR-II operation to successfully recycle metallic fuel pins. A closed fuel cycle would be considered when reprocessing spent fuel is a licensed activity by the regulatory body in Canada, acceptable to the public, and economically viable. The ARC-100 fuel fabrication process is suitable for shielded production of recycled fuel.

12. Waste Management and Disposal Plans

The low quantities of solid, liquid, and gaseous radioactive waste resulting from operations will be handled and processed in a responsible and safe manner consistent with the state-of the-art that ensures minimum exposure to all personnel handling, transporting, and processing the waste. Interim storage will be stored on the site in defined areas and transported to authorized processing facilities at appropriate times, dependent on the category and type of waste. Used fuel assemblies are temporarily stored in-vessel at the outside of the core. Once sufficiently cooled, the used fuel assemblies are periodically extracted and placed into an on-site dry-cask storage. Long term storage is planned in containers that meet the requirements for the deep geological repository design of the Nuclear Waste Management Organization (NWMO).

13. Development Milestones

2020	Conceptual Design complete
2023	Preliminary Design complete
2026	License to Prepare Site
2027	License to Construct for first unit
2029	License to Operate First Unit

BLUE CAPSULE (Blue Capsule Technology, France)*

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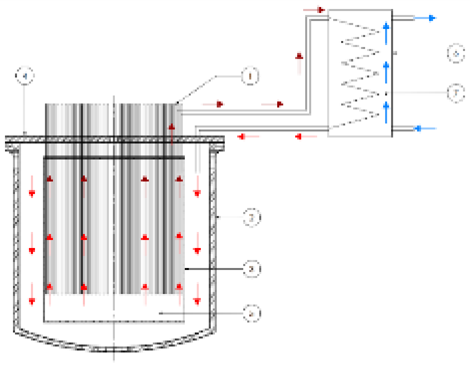
*** Note: Blue capsule design is liquid metal cooled but is not a fast spectrum reactor. It has been arbitrarily grouped with LMFR designs in this catalogue but belongs to the "Others" category in the ARIS database.**



1	Fuel channels (Ceramic immersion sleeves)
2	Main vessel
3	Inner vessel
4	Reactor slab
5	Bottom Plenum (Sodium turning zone)
6	Na/Air Heat exchanger
7	Secondary circuit (Air)

Coolant circulation in Blue Capsule core and Na-air heat exchanger

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Blue Capsule Technology, France
Reactor type	Graphite-moderated, liquid-metal-cooled, high-temperature reactor
Coolant/moderator	Sodium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	150 MWth / 50 MWe
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	0,1 sodium / 0,5 air
Core Inlet/Outlet Coolant Temperature (°C)	400 °C / 750°C
Fuel type/assembly array	Prismatic, TRISO with UO_2 kernel
Number of fuel assemblies in the core	
Fuel enrichment (%)	5% in ^{235}U
Core Discharge Burnup (GWd/ton)	Approx 30 GWd/t
Refuelling Cycle (months)	Continuous refuelling
Reactivity control	Control rods
Approach to safety systems	Passive safety (natural coolant circulation, TRISO failure temperature significantly higher than sodium boiling point), ultimate air coolant
Design life (years)	60
Plant footprint (m²)	690 (Reactor Building only)
External vessel diameter x height (m)	Ø8 x 6
RPV weight (metric ton)	Approx 130
Seismic Design (SSE)	
Distinguishing features	Fuel rods isolated from sodium ("dry fuel"), open secondary coolant circuit
Design status	Pre-conceptual design and technology validation

1. Introduction

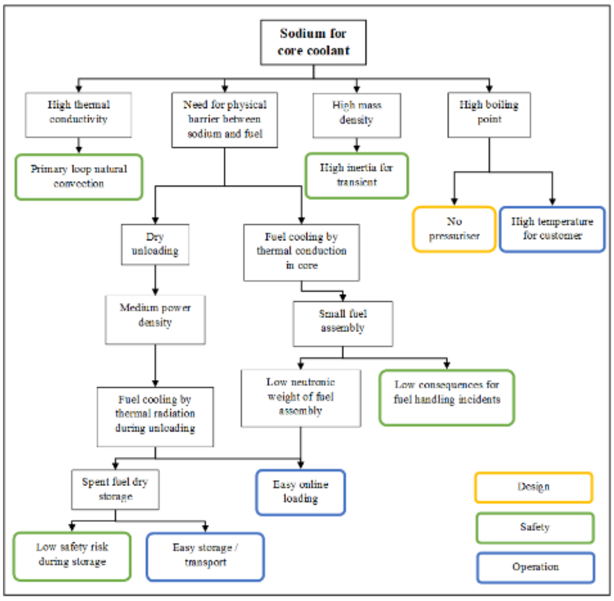
Blue Capsule is a 150 MWth reactor that allows onsite co-location at hard-to-abate industries and is designed to maximise the production of industrial-grade heat in the form of air at temperatures up to 700°C. This output can be used directly for preheating components or materials, transformed into high-grade steam at 650°C, or converted into electricity up to 50 MWe per unit. The innovation lies in deploying a cross-over of sodium-cooled and high-temperature reactor (HTR) technology, featuring TRISO fuel and primary loop at atmospheric pressure. The ultimate heat sink is ambient air in an open circuit, with the air passing through the sodium-air heat exchangers and then directed towards the conventional, versatile part of the system, which interfaces with the final user facility.

2. Target Application

Blue Capsule targets three key end users: (1) decarbonizing existing industrial sites using a "Plug-in" approach, with high-temperature steam up to 650°C, (2) Supporting “Multi-vector” type end users with co-generation capabilities, and (3) integrating into “Pre-heating” end user processes like glass, steel, or the fabrication of ceramics.

3. Design Philosophy

sodium-cooled fast reactors. Using sodium as the core coolant and graphite moderator – as already experienced with the NSRE and Halam projects – Blue Capsule offers several possibilities linked to key design options. First, the high thermal conductivity of sodium allows for a high thermal exchange capacity that, in turn, allows cooling of the core power by natural circulation. This implies a simple, robust means to cool the core during both normal and abnormal conditions. In addition, with the high mass density of sodium (compared with HTR cooling with pressurised helium), the transients are slow – giving time for the operators to cope with potential events. As concerns Blue Capsule’s potential competitiveness (i.e. vis-à-vis industrial heat sources such as natural gas), this stems from the cost-driven design and “plug-in” integration with industrial sites.



Options taken by the Blue Capsule designers, leading to design, safety and exploitation benefits

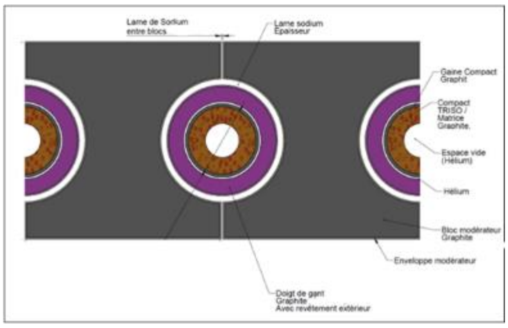
4. Main Design Features

(a) Power Conversion System

The heat from the assemblies is evacuated by conduction through the fuel, graphite, and thermal binder to the ceramic housing structure, which is then cooled by sodium in natural circulation. The sodium coolant is not in direct contact with the fuel; its outlet average temperature is 750°C. The sodium is directed towards five identical sodium/air heat exchangers upon exiting the core. The Blue Capsule secondary circuit is versatile, meaning that any transformation of the energy flow (hot air, various steam flows, electricity) downstream of the reactor does not affect its design or operation. This characteristic is central to adapting the reactor to the different industrial heat usage scenarios by end users. This versatility is achieved by using an "open" energy conversion cycle.

(b) Reactor Core

The core of Blue Capsule is similar to a core of gas-cooled HTRs. The fuel compacts Blue Capsule’s core pattern made of TRISO particles with UO2 kernels (enrichment 5% in 235U) placed in a graphite matrix, with a volume packing fraction around 30%. The main physical difference between the Blue Capsule core and gas-cooled HTRs is that Blue Capsule fuel is encapsulated in ceramic cylinders (Blue Capsule’s core pattern – not at scale) that separate the fuel compact from sodium flow, to protect graphite from interaction with sodium. Thus, Blue Capsule fuel is in a dry zone, separated from the sodium flow.



Blue Capsule’s core pattern

(c) Reactivity Control

Reactivity is managed using control rods from the reactor hall above the core. These rods are made of neutron-absorbing material and regulate both the level of reactivity and the power output. Nuclear power is intended to remain constant, with any variability in the final product being managed through a heat storage system, if necessary. The control rods are placed in sleeves like those housing the fuel assemblies. Their positioning within the core will allow for flux distribution adjustment.

(d) Reactor Internal and Main Vessels and Internals

The reactor core is enveloped by a non-pressurised main vessel containing hot sodium. It directs the ascending flow under natural convection and the descending flow of sodium. The fluid is distributed laterally to the heat exchangers in the upper section. Once cooled in the heat exchangers, the sodium descends into the annular space between the inner and main vessels, reversing direction in the lower section and rising back into the core. The main vessel is suspended from the reactor deck, and the inner vessel rests on top of it. A safety vessel is located around the main vessel in case of mechanical failure. The entire core and these vessels are situated within the vessel pit.

(e) Reactor Coolant System

After exiting the core, the sodium is directed towards sodium/air heat exchangers. These are located on the deck above the core within the reactor hall. The currently preferred option for the heat exchangers is shell-and-tube type, in two cylindrical sections, allowing for double-pass cooling. This design also helps reduce the height of these components, improving seismic performance. It also provides a high point that facilitates the easy installation of an expansion tank. The heat exchangers' design and positioning – as close as possible to the core minimise the piping length – reduce the risk of leaks.

(f) Pressuriser

There is no pressuriser in the reactor; it operates at atmospheric pressure.

(g) Primary pumps

The reactor operates under natural convection without primary pumps.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Blue Capsule takes advantage of the best properties of sodium as a coolant, and of TRISO particles, both in terms of safety and performance. Using liquid sodium as a coolant allows to keep the maximal fuel temperature of 1250°C in normal operation and to keep the fuel temperature below the TRISO failure limit in all conditions. The sufficient resilience of sodium boundaries, such as the shells of sodium-gas heat exchangers, the reactor deck, or the safety vessel, prevents significant off-site chemical risks. Blue Capsule uses air as the ultimate source term, and since no water is involved, sodium-water reaction is avoided by design. The Blue Capsule safety and security approach could be summarised as: generalised loss of the first barrier eliminated; core uncover eliminated; no risk of sodium release into the atmosphere; no risk of depressurisation; fuel qualification in accidental conditions covered by normal operation, and; maximum sodium temperature = 880°C.

(b) Safety Approach and Configuration to Manage Design Basis Condition

Firstly, the low neutronic weight of the fuel assemblies and the neutron absorbers allows for the implementation of slow automation, ensuring stable operation. Core cooling is based on a primary circuit with natural circulation, designed to ensure a wide range of physical stability. Furthermore, the compact design of the primary circuit with the positioning of the heat exchangers just above the core – so no pressurisation – leads to an almost complete absence of pipelines, valves, and tee-branch. This greatly limits the risks of accidents related to the primary circuit. Ventilation plays a central role as it ensures the cooling function of the primary vessel as well as the heat evacuation of the stored irradiated fuel, but redundancies on sensitive sub-systems and components will be put in place to reduce the probability of initiating events.

In general, the operators' actions are very limited, greatly reducing the risk of errors, and the algorithms rely on a level of redundancy to achieve significant reliability. Concerning external threats, especially considering co-location with the industrial customer site, the reactor building is placed at subterranean level for protection. The core's natural behaviour plays a major role in the safety of the installation, provided that the design parameters make the best use of this capacity. Coupled with the high robustness of the TRISO fuel, this eliminates significant fuel degradation and practically prevents substantial radioactive releases.

(c) Safety Approach and Configuration to Manage Design Basis Earthquake
Blue Capsule – and in particular the reactor core and safety systems – is being designed so that potential seismic events are being considered.

(d) Containment System
No credible scenario leads to a generalised failure of the first barrier (TRISO). In the event of particle failure, the primary circuit, the vessel pit, the loading bubble, and the biological shield ensure containment.

(e) Spent Fuel Cooling Safety Approach / System
The Blue Capsule core can be unloaded completely during operation, and the interim storage inside the reactor building is designed accordingly. When considered alongside the thermal robustness of TRISO, this allows a cooling by radiation during irradiated fuel handling – even with high residual power – immediately after the fission reaction stops.
The refractory features of TRISO also allow to obtain dry interim storage in racks, only cooled by air circulation, then by radiation once placed in casks, shortly after unloading from the core. The limited size of the fuel assemblies also leads to lower risks of significant radioactive releases during fuel handling, which is a strong safety asset, in addition to the impossibility of general fuel failure.

6. Plant Safety and Operational Performances
The target capacity factor is 90% (thanks to online refuelling), the thermal power output is 150 MWth, and the outlet temperatures are 750°C primary and secondary: 700°C air, 650°C steam, and 50 MWe. This means that Blue Capsule's output is versatile and can be adapted to the client's needs.

7. Instrumentation and Control Systems
Reactivity is managed using control rods from the reactor hall above the core. These rods are made of neutron-absorbing material and regulate both the level of reactivity and the power output. Nuclear power is intended to remain constant, with any variability in the final product being managed through a heat storage system or variation in end products, if necessary. The control rods are placed in sleeves like those housing the fuel assemblies. Their positioning within the core will allow for flux distribution adjustment.

8. Plant Layout Arrangement
The logic of the plant building's layout is based on the following principles: proximity of the reactor building to the consumer processes, as well as shared utilities and support functions. For multiple capsules co-located on an industrial site, the intention is to share all functions that are not present in the reactor buildings of each capsule, including electrical supply, support network supply, support buildings, fire service, physical protection, storage for spare parts: for parts not subject to physical protection, conventional network (secondary, turbine, alternator, compressor, etc.), temporary construction and heavy maintenance area. It is assumed there will be road access to facilitate the delivery of large equipment.

Coolant circulation in Blue Capsule core and Na-air heat exchanger

9. Testing Conducted for Design Verification and Validation
The design of Blue Capsule revolves around several experimental installations that are integral to the development plan and timeline:

- A laboratory-scale experimental loop, “proof of concept”, is specifically designed to demonstrate the heat removal performance of the Blue Capsule reactor core through a natural convection circuit, and will be put in operation at the beginning of 2026.
- A non-nuclear prototype scheduled for the first half of 2029 includes a complete thermohydraulic representation of Blue Capsule, with the output connected as an industrial heat source to test the representative processes for the final uses of the Blue Capsule nuclear solution. This prototype ensures that Blue Capsule is ready for commercial deployment by thoroughly testing and validating all key aspects before full-scale nuclear operation.

This structured approach ensures that the Blue Capsule design is rigorously tested and validated at every stage.

10. Design and Licensing Status

Blue Capsule conducted the sketch and feasibility study phase in 2022-2023, with the Preliminary Design phase starting in early 2024 and expected to finish by mid-2026. The reactor benefits from mature technological components and successful licensing experiences in France and internationally. Preliminary discussions with French nuclear authorities have begun. The reactor design combines sodium-cooled and HTR technologies, both of which are mature and well-understood, facilitating easier regulatory and safety approvals, especially in France and the European Union. The use of singularly robust TRISO fuel further enhances safety. Safety demonstrations (in France) will use methods developed for more complex sodium cooled fast reactors (Blue Capsule is a thermal-spectrum concept), such as Phenix, Superphenix, and Astrid projects. The design will be validated through an experimental program, including the laboratory scale testing starting in 2025-2026 (representative sodium loop) and at the reactor scale (non-nuclear prototype) by 2030.

11. Fuel Cycle Approach

TRISO-based fuels are traditionally considered for use in an open cycle. However, experiments have been conducted to recycle TRISO fuels and small-scale industrial capabilities for recycling non-irradiated TRISO fuels do exist internationally. Blue Capsule is already considering options for nuclear fuel reprocessing at the preliminary design phase. Since HTR/TRISO fuels are excellent at burning materials, Blue Capsule is also exploring options for using MOX-based TRISO and options for treating and recycling spent fuels.

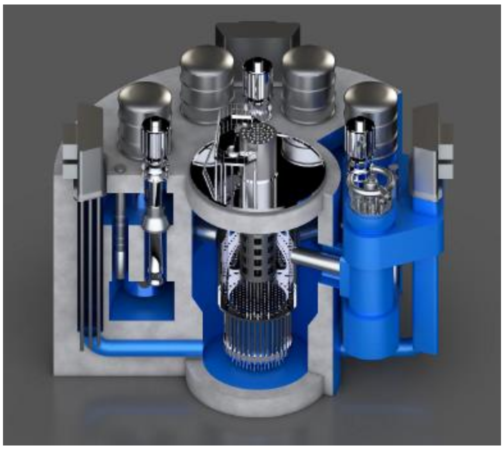
12. Waste Management and Disposal Plan

Blue Capsule’s waste management and disposal plan is designed to consider the French national framework for waste classification and management.

13. Development Milestones

2022 – 2023	Preliminary studies and technological innovation (using previously developed patents).	Complete
2024 – 2026	Pre-conceptual design phase and technology validation	On track
2026 – 2027	Conceptual Design Phase (and preparation for pre-licensing)	On track
2027 – 2028	Basic Design Phase	Planned
From 2027	Commercialisation	Planned
2028 – 2030	Detailed Design Phase	Planned
By 2029	Target first concrete for the FOAK plant	Planned

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	Lead
Thermal/electrical capacity, MW(t)/MW(e)	700 / 300
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	0.1 / 17 – 18.5
Core inlet/outlet coolant temperature (°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
Refuelling cycle (months)	36 – 78
Core discharge burnup (GWd/ton)	61.45
Reactivity control mechanism	Reactivity compensation (RC), emergency protection (EP) and automatic control (AC) members
Approach to safety systems	Passive
Design life (years)	30
Plant footprint (m²)	80 × 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	Construction in progress

1. Introduction

BREST-OD-300 innovative lead-cooled fast reactor is developed as a pilot demonstration prototype for base-type commercial reactor facilities of the future nuclear power industry with the closed nuclear fuel cycle (NFC). The reactor is fuelled with uranium plutonium mononitride (U-Pu)N and uses a two-circuit heat transport system to deliver heat to a subcritical steam turbine and generate electricity of 300 MW(e). The design documentation has enabled the approval for the construction of the BREST-OD-300 power unit as part of the pilot and demonstration energy complex (PDEC) aimed at the conformity with current common and specific standards for lead-cooled reactor facilities. Endurance of the reactor facility will be demonstrated. An extensive R&D program, required for commercial lead-cooled reactor facilities, will be carried out.

2. Target Application

The BREST-OD-300 reactor is designed for different modes of operation and optimizing all processes and systems that support the reactor operation. The main goal is practical confirmation of the 'inherent safety' concept of the lead-cooled fast reactor, operating in NPP mode in NFC. After operational tests, the unit will be commissioned for electricity supply to the grid.

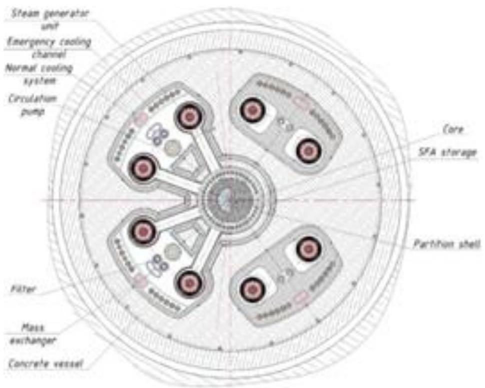
3. Design Philosophy

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 primary 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 adverse 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 reactor core, 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.

4. Main Design Features

(a) Power Conversion

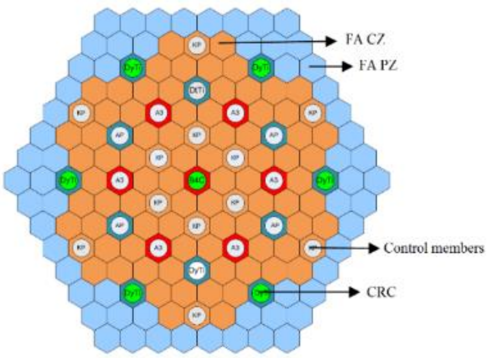
The pool-type reactor design has an integral lead circuit accommodated in one central and four peripheral cavities of the metal-concrete vessel. The central cavity houses the core barrel together with the side reflector, the CPS rods, the spent fuel assembly (SFA) storage and the shell that partitions 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.



BREST-OD-300 Layout

(b) Reactor Core

The lead coolant properties in combination with 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. The FA design allows radial coolant flow transfer in the core which prevents overheating of the damaged FA as shown in the figure on reactor core map.



BREST-OD-300 Reactor Core Map

(c) Reactivity Control

When operating a core with a "zero" reactivity margin for fuel burnup during a micro campaign, all reactivity compensation (RC) control members and emergency protection (EP) control members are removed from the core. Only two pairs of automatic control (AC) members remain partially inserted, compensating for the neptunium effect of reactivity ~ 0.08 % Δk/k and small reactivity swings when isotopic composition of fuel in the core changes. There is no need for the group consisting of all RC control members to participate in compensation of reactivity loss during fuel burnup; the function of RC control members is to compensate the temperature and power effect of reactivity; they are partially inserted into the core in the cold state at a minimum power.

(d) Reactor Pressure Vessel and Internals

An integral pool-type 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 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.

(e) 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.

(f) Secondary System

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) Steam Generator

Steam generator is designed with single-walled twisted tubes, corrosion resistant in water and lead, no welds along the entire length.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

To guarantee the safety the preferential use: neutronic properties and physico-chemical properties of fuel, coolant, materials, as well as special design solutions that allow to fully realize these properties. The lead coolant properties make it possible to implement in the 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 introducing 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 (pool-type) 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 / Reactor Cooling Philosophy

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) Spent Fuel Cooling Safety Approach / System

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

(e) Chemical Control

Regulations for lead coolant including cleaning and decontamination were developed and confirmed by the experience in operating lead test benches. The absence of corrosion of materials in excess of the specified limits at a regulated oxygen concentration in lead of (1–4) $\times 10^{-6}$ was justified.

6. 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 for the lead-cooled fast 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 fast reactors.

The probability of severe accidents due to internal reasons without core destruction involving fuel and cladding melting, coolant boiling, disruption of circulation in the primary circuit does not exceed $6.48 \cdot 10^{-9}$ 1/year; the release of radionuclides per day under the conservative scenarios does not exceed the control value. The absence of the need for evacuation and resettlement of the population in severe accidents at the power unit is ensured with a probability of $3.2 \cdot 10^{-8}$ 1/year.

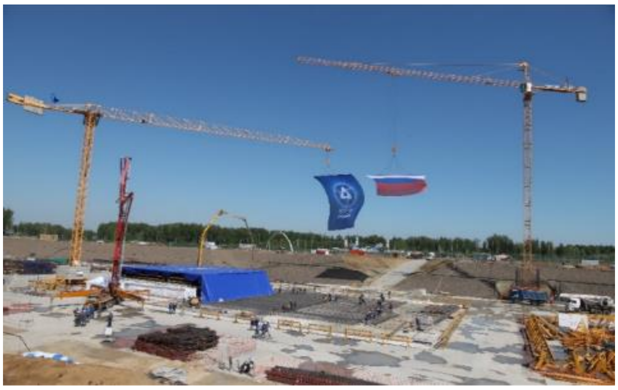
7. Instrumentation and Control System

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.

8. 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.

9. Testing Conducted for Design Verification and Validation


Pouring the first concrete into the foundation of the power unit with the BREST-OD-300 reactor

The complete detailed design of the BREST-OD-300 reactor facility has been carried out. To date, experimental justification of components, elements and equipment of reactor facilities has been carried out using small- and medium-scale mock-ups and pilot models. Verified and certified software tools were used for computational design justification.

10. Design and Licensing Status

The BREST-OD-300 unit design received a positive assessment of the Glavgosexpertiza (2018). Expert review of the Russian Academy of Sciences (RAS) was carried out, which confirmed that the design of the BREST-OD-300 power unit corresponds to the current level of science and technology, scientific ideas about the problems of the existing nuclear power and ways to solve them. The RAS recommended the construction of the BREST-OD-300 power unit (2019). Following the results of a long, detailed design review, in February 2021 Rostekhnadzor issued a license for the construction of the power unit with BREST-OD-300. A solemn ceremony was held on June 8, 2021, with pouring of the “first concrete” marking the beginning of the power unit construction (Fig. 3). The equipment fabrication and installation and construction of BREST-OD-300 is planned to be completed in 2026 and begin preparations for the reactor start-up.

11. 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.

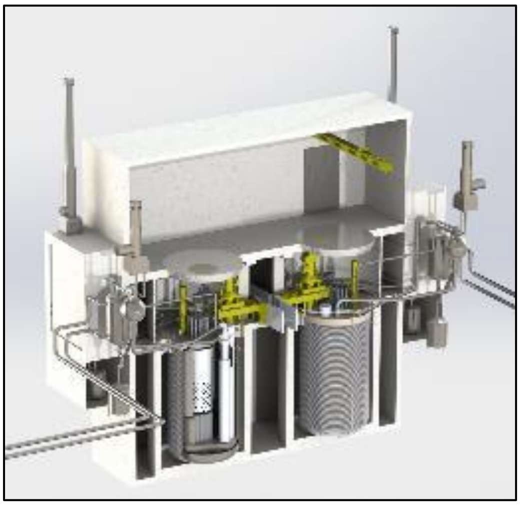
12. Waste Management and Disposal Plan

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

13. Development Milestones

1995	Conceptual design development initiated
2002	Feasibility study of the BREST-OD-300 NPP with an on-site nuclear fuel cycle (OSNFC)
2016	Design study of the BREST-OD-300 NPP with an on-site nuclear fuel cycle (OSNFC) at the Tomsk Site
2021	Start of construction of the power unit with the BREST-OD-300 reactor
2026	First of a kind pilot demonstration plant starts operation

HEXANA (Hexana, France)



KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	HEXANA, France
Reactor type	AMR
Coolant/moderator	Sodium/none
Thermal/electrical capacity, MW(t)/MW(e)	2 x 400 MW_{th} 2 x 150 MW_{e}
Primary circulation	Forced convection
NSSS Operating Pressure (primary/secondary), MPa	-
Core Inlet/Outlet Coolant Temperature (ºC)	400-530
Fuel type/assembly array	MOX
Number of fuel assemblies in the core	80
Fuel enrichment (%)	<28% Pu
Core Discharge Burnup (GWd/ton)	110
Refuelling Cycle (months)	12 to 18
Reactivity control	Control rods
Approach to safety systems	Diversified with passive
Design life (years)	60 years
Plant footprint (m^2)	15000
RPV height/diameter (m)	14/9
RPV weight (metric ton)	
Seismic Design (SSE)	
Distinguishing features	
Design status	Conceptual Design

1. Introduction

HEXANA is an industrial startup company in the field of nuclear. It was launched in 2023 with the support from the French Alternative Energies and Atomic Energy Commission (CEA), a key Research Technology Organisation (RTO) in scientific and technological research in France in low carbon energies (e.g. nuclear and renewable). It benefits from the CEA’s and EDF’s technologies, know-how and patents to develop a Gen4 small advanced modular reactors (AMRs). It intends to develop an AMR based on the sodium-cooled fast reactor (SFR) technology, a proven nuclear technology, that incorporates a high-temperature heat storage device.

SFRs are of great interest with respect to nuclear waste management, and there is renewed interest for this technology worldwide: they do not need natural or enriched uranium fuels to operate, but rather use depleted uranium combined with recycled plutonium collected from the reprocessing of Gen3 reactors spent fuels (MOX). With no natural or enriched uranium used, which represent half of conventional nuclear gas emissions and no critical resources (e.g. lithium, helium, cobalt, etc.), waste management, environmental protection and sustainability are greatly improved over the long term. Further, those reactors shall contribute to reducing its volume of high-level nuclear waste within the scope of closing the nuclear fuel cycle. Through the France 2030 programme “*Innovative Nuclear Reactors*”, CEA is committed to supporting startups looking to promote decarbonisation through the development of innovative reactor projects matching with a better management of fuel resources and the nuclear waste reduction.

2. Target Application

At present, 43% of global CO_2 emissions come from the industry and transports, and industrial energy is driven by fossil fuels at 90%. With its system, HEXANA has set itself the mission to remove their use of fossil fuels and to decarbonise energy-intensive and hard-to-abate industries (e.g. steel, chemistry, cement, refining, maritime and aviation fuels) with a sustainable nuclear solution that meets their requirements in terms of long-term competitiveness and power supply levels, while enabling upcycling using recyclable fuel.

The facility will comprise two small reactor units (400 MW_{th} each) that will supply a heat storage device specially designed to manage fluctuating demand for electric power from the industries and electric grids. It will be equipped with an energy conversion system to ensure the flexible production of electricity on demand, making it possible to compete with gas-fired power stations and to provide process heat directly to energy-intensive industries within proximity. This will also represent the expected solution to support renewable energies and complement their intermittency production thanks to its flexibility. Production is expected to be on site, directly based on an industrial cluster. With both electricity and high temperature process heat (500°C), the latter will be able to capture CO_2, to produce steam, hydrogen and synthetic fuels, and to commit firmly to the decarbonisation process.

3. Design Philosophy

HEXANA system provides two reactors associated with a thermal storage unit offering flexible power delivery in response to the needs of its customers (electricity and/or heat) for industrial process needs. HEXANA integrates design options offering high guarantees from the point of view of safety, relying in particular on passive safety devices. HEXANA is primarily targeting the decarbonization of energy-intensive and highly greenhouse gas-emitting industrial processes, with the combined production of heat and electricity, bases for the production of hydrogen or synthetic fuels. The SFR heat (500°C) gives access to all the amenities needed to defossilize our economy : steam, hydrogen, CO_2 capture, synthetic fuels, green steel, chemical molecules, fresh water by desalination, ammonia, fertilizers… HEXANA select the SFR technology also as part of a circular economy by using available nuclear materials : depleted uranium and plutonium from PWR spent fuel. This choice avoids any dependency on the import of uranium and offers a strategic advantage in terms of sovereignty and sustainability. With the SFR technology, HEXANA produces very low carbon energy estimated at 2.5 gCO_2/kWh, which is a key asset.

If the main features of the SFR reactor are already proven technologies, HEXANA brings some major breakthroughs in the reactor design and in the use of nuclear energy in order to perfectly match with industrial needs.

- The design is made modular thanks to, in particular, a compact vessel. This is crucial for competitive issues and this enables to twin two modules on site efficiently so that the energy delivery is continuous;
- It includes heat storage tanks to ensure a high flexibility of the heat delivery, and a strong decoupling between the operation of the nuclear island and the power conversion system. The nuclear island is working on a base-load mode, increasing its plant capacity and reliability, its lifetime and, as a consequence, its revenue;
- The whole concept is not built in a full electrical paradigm, neither in a small hybridisation prospect for which a small amount of heat is extracted from a Power Conversion System optimized for electricity production. Conversely, it is designed in an integrated way so that the industrial end-user is fed with the right balance of electricity and heat, and heat at the right temperature(s) level(s). In other words, HEXANA ambition is to become a taylor-made solution to guarantee an optimized energy delivery as an efficient substitute to fossil fuels. If optimized for the hydrogen production, this approach results in the production of 200 tons of H_2 per day for a twinned configuration. When compared to H_2 production with a PWR featured by the same core power and the same thermal coupling possibilities between the reactor and the Solid Oxide Electrolyzer Cell (SOEC), HEXANA produces 25% more H_2 than the PWR. This value reaches +50% if thermal coupling cannot be achieved with the PWR.

(f) Nuclear Steam Supply System

The Nuclear Island includes the sodium primary circuit with the core and main components and the sodium secondary circuits up to th sodium/salt heat exchangers. The molten salt thermal energy storage system, the steam generators and the Rankine power conversion system are located outside the nuclear island.

(g) Reactor Core

The reactor core is composed by 80 fuel sub-assembly. The fuel is MOX pellets with depleted or natural uranium and plutonium from PWR spent fuels. The fuel pins are located in an hexagonal can.

(h) Reactivity Control

For the reactivity control function, the design includes control rods made of moving rods inside the core with a neutron absorber (boron carbide). Two independent systems are implemented with strong redundancy and diversification. For HEXANA, we work also on a passive device that can drop neutron absorber when the sodium becames too hot, without I&C. That kind of device was studied during the ASTRID project.

(i) Reactor Pressure Vessel and Internals

The reactor is a sodium pool type reactor, meaning that all main components are located in the primary vessel (primary pumps, sodium heat exchangers, fuel handling machine).

(j) Reactor Coolant System and Steam Generator

Two sodium secondary loops are located between the primary circuit and the conventional island. The conversion system includes sodium/salt heat exchangers, a thermal energy storage system (TES) with two molten salt storage tanks, four steam generators and a Rankine power conversion system (PCS). For the heat removal function, an advantage of SFR is that we can use sodium/air heat exchangers and sodium circuits that operates without electricity or water, completely passively. This is related to the very good thermal properties of the sodium fluid (heat conductivity). For the HEXANA design, we plan to have two systems (again with strong redundancy and diversification), one with sodium circuits and sodium/air heat exchangers connected to the secondary circuits and one through the primary vessel, by cooling the reactor pit. That kind of system is efficient only if the power of the reactor is not too important, which is the case for HEXANA (400 MWth).

(k) Pressuriser

No pressure in a Sodium Fast Reactor

(l) Primary pumps

Two primary pumps are located in the primary vessel. No specific issue is expected as these materials have a good operational feedback in France in Phenix and Superphenix.

4. Safety Features

(a) Engineered Safety System Approach and Configuration

For the safety framework, the WENRA objectives will be applied, as they were already the reference for the previous SFR project in France (ASTRID). HEXANA wants to implement an holistic approach "3S by design" to take into account early in the design process not only Safety requirements but also Security and Safeguards. HEXANA is a Gen4 system and will apply the basis safety approach promoted within the International GenIV Forum. The SFR Safety Design Criteria will be implemented within the design.

(b) Safety Approach and Configuration to Manage DBC

Several systems are designed to fulfill the DBC requirements with the sufficient level of diversification to ensure enough reliability for prevention and mitigation of DBC. For Decay Heat Removal systems, at least one of the two systems will be designed to fulfill its requirements in fully passive mode.

(c) Safety Approach and Configuration to Manage DEC

HEXANA considers severe accidents conditions in DEC as part of the fourth level of defense-in-depth. A Practical Elimination approach will also be implemented to complete the safety demonstration.

(d) Containment System

For the containment function, one key point of SFR pool-type design like HEXANA is that all the primary components (pumps, exchangers) are inside the primary vessel. The primary sodium therefore does not leave the primary vessel (difference with the water in a PWR). Besides, we have no pressure in the circuits (sodium is liquid at atmospheric pressure -> no need for pressure instead for PWR). Therefore there is no LOCA transient as in PWR thanks to the pool (integrated concept). In case of primary vessel sodium leakage, the reactor pit is design to collect the sodium safely and to stabilize the sodium level in the primary vessel, and to ensure the coolability of the core. The risk of draining the sodium outside of the core is therefore excluded by design. It is a big strength in the safety demonstration of designs like Phénix, SuperPhénix and HEXANA. This is not the case for loop type reactors.

(e) Spent Fuel Cooling Safety Approach / System

Spent fuel are stored in a water pool to be cooled enough before their retreatment. No specific issues are expected.

5. Plant Safety and Operational Performances

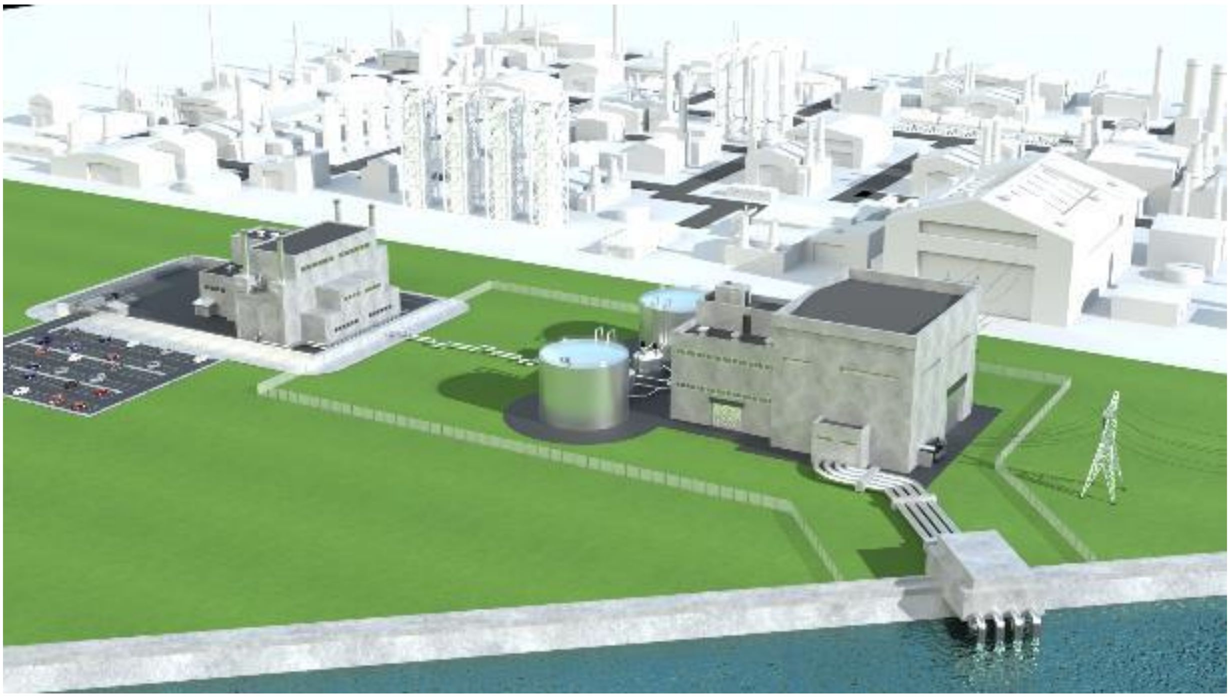
HEXANA has a speficic operating condition that allow a very good level of reliability and availability. The two reactors are operated in baseload mode without any transient too unsure the operation. The flexibility of the production for the grid or the end-user is ensured by the Thermal Energy Storage System.

6. Instrumentation and Control Systems

I&C systems developed fro Phenix, SuperPhenix and ASTRID in France are sufficiently known to provide a good operation and safety level. Some specific I&C systems will be developed by HEXANA for the overall control of the plant between the nuclear island, the thermal energy storage and the conventional island.

7. Plant Layout Arrangement

The nuclear island is segregated from the conventional island and the end-user industrial customer.



8. Testing Conducted for Design Verification and Validation

Most of components are well known due to SFR good operational feedback in France. Testing is necessary for specific design parts of the design :

- Sodium/salt heat exchangers

- Overall system operation (sodium+ thermal energy storage+steam generators)
- Fuel handling in the reactor
- Expansion bellows for sodium systems
- Reactor vault cooling system

The qualification is scheduled with several partners during the conceptual design and basic design.

9. Design and Licensing Status

The Conceptuel Design phase will be conducted during 18 months until march 2026, when a Safety Case will be submitted to the french safety authorities. Then the Basic Design will last until early 2029 with the submission of a Building Permit demand. In parallel, sever al security cases are planned concerning non-proliferation issues and physical protection.

10. Fuel Cycle Approach

The use of a SFR meets a triple challenge : the reuse of plutonium from PWR spent fuel, the reduction of dependence on uranium imports and the minimization of Long-Lived Radioactive Waste (LLRW) to be stored. HEXANA offers a SFR capable of meeting these challenges. A major challenge is that of plutonium from PWR spent fuel, which would be by far the largest contributor to LLRW if it were not valued as a resource. It is reused once in France with MOX utilization in some of EDF PWRs. However, the inevitable degradation of the isotopic vector requires having SFR to sustainably reuse plutonium and stabilize its inventory on French soil, in quantity and quality. As a reminder, the current french fleet accumulates 120 t/year of spent MOX fuels. HEXANA therefore aims to be able to valorize used MOX, in support of ad hoc upstream and downstream industrial sectors.

The second challenge is that of the uranium resource and its enrichment. Even if the known reserves, the diversity of suppliers and the current tension on the market do not suggest a real surge in prices in the short term, we can note a tripling of the price of uranium since the beginning of 2020 in the wake of other raw materials. These economic arguments remain relative for the moment: France has large stocks, a diversity of suppliers and the impact of the cost of U on the price of MWh is low. However, if a rapid and massive nuclear renaissance were to take place, tensions on the U market, or even on Pu stocks, could increase more quickly than the current rate of development of Fast Reactors. At the current rate, uranium stocks provide 100 to 150 years of resources. However, in the event of a global desire to achieve Net-Zero 2050 thanks in part to nuclear power, tensions could appear as early as the middle of the century. The issue remains geostrategic in the medium term (energy sovereignty) rather than a real short-term challenge.

It should also be noted that the equivalent CO_2 emissions of the current nuclear kWh are 50% due to uranium extraction and enrichment activities. A SFR with a closed fuel cycle can therefore reach an ultra-low carbon energy supply, up to 2.5gCO_2éq/kWh, which is unbeatable.

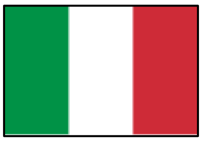
11. Waste Management and Disposal Plan

In terms of core physics, it remains undeniable that plutonium from spent MOX fuels and depleted uranium from enrichment are materials that can be used in SFR and therefore in HEXANA reactors. Fast reactor MOX has been retreated in France since the 60's, before PWR fuels… HEXANA fuel can be retreated with current knowledge, it is included in our fuel specifications. The plutonium from HEXANA spent fuels can be reused for unlimited MOX fabrication, paving the way to the close fuel cycle. Only minor actinides can be considered as the final wastes to be included in each national disposal plans.

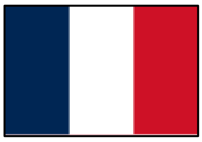
12. Development Milestones

2020 – 2023	Preliminary studies within CEA	Complete
2024 – 2026	Conceptual Design	On track
2026 – 2028	Basic Design	Planned
2028 – 2030	Detailed Design	Planned
2030	Target first concrete for the FOAK plant	Planned

2035	FOAK operation	Planned
2050	Up to 40 in operation in Europe	Planned



LFR-AS-200 (NewCleo, Italy/France)



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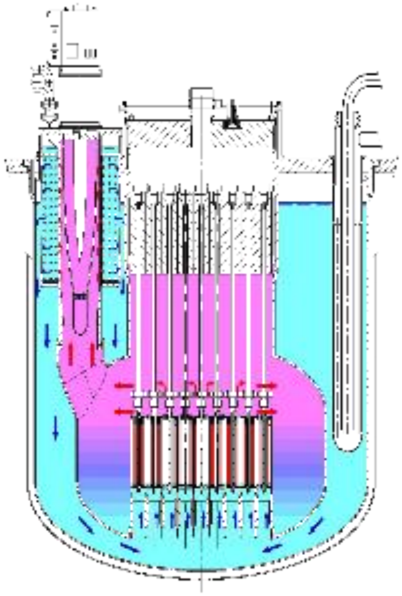
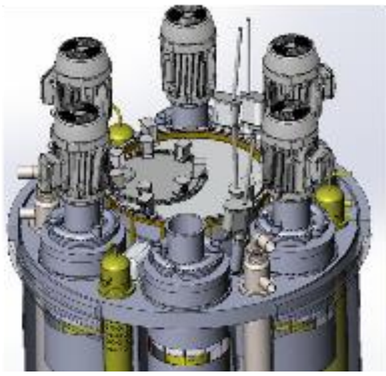




KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Newcleo, United Kingdom
Reactor type	Lead-cooled Fast Reactor (LFR)
Coolant/moderator	Lead (Pb)/None
Thermal/electrical capacity, MW(t)/MW(e)	480 / 200
Primary circulation	Forced circulation (6 pumps)
NSSS Operating Pressure (primary/secondary), MPa	0.01 / 18
Core Inlet/Outlet Coolant Temperature (°C)	420 / 530
Fuel type/assembly array	MOX, hexagonal FA with triangular pin pitch
Number of fuel assemblies in the core	61
Fuel enrichment (%)	19% average, 23.2% max (Pu)
Core Discharge Burnup (GWd/ton)	100 GWd/ton
Refuelling Cycle (months)	16 months or “continuous”
Reactivity control	Control rods, burnable absorbers (B_4C)
Approach to safety systems	Passive and active, no core melt
Design life (years)	60
Plant footprint (m²)	1,100 m² (nuclear island)
RPV height/diameter (m)	6.2 m / 6.3 m at flange
RPV weight (metric ton)	42
Seismic Design (SSE)	0.3g
Distinguishing features	Amphora-shaped inner vessel, lead coolant, no intermediate loops, compact design
Design status	Conceptual, pre-licensing discussions in France (2023)

1. Introduction

The LFR-AS-200 is an innovative, small modular reactor (SMR) of the Lead-cooled Fast Reactor (LFR) type, with AS referring to the "Amphora-Shaped" design of its inner vessel. It offers 200 MW(e) output with a strong focus on compactness, competitive cost, and the ability to use stockpiled plutonium for energy generation. The design simplifies the reactor structure by eliminating intermediate loops and enhancing passive safety features.

2. Target Application

The absence of intermediate loops, the small primary system specific volume and the compact reactor building are key factors for competitive kWh cost. Market application is energy production, also for non-electrical uses (ammonia and hydrogen). This reactor will use of stockpiled Pu and in perspective recycle of minor actinides without burden of longlived transuranics in the waste. The breeding ratio is 0.9 without blanket assemblies and can be reduced with core design adaptation where required.

3. 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 refuelling machine nor of intermediate loops. A broad R&D programme has been established by Newcleo to deploy the LFR-AS-200 and enable the full potential of the LFR technology.

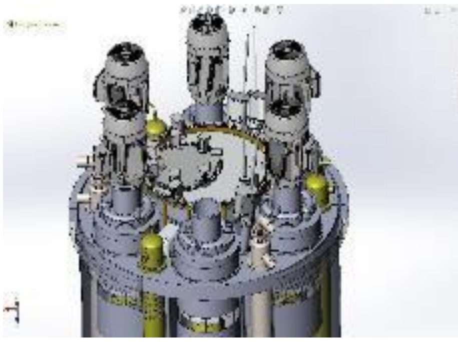
4. Main Design Features

(a) Nuclear Steam Supply System

The NSSS of the LFR-AS-200 incorporates a cylindrical reactor vessel with an oval bottom and flat roof, where lead coolant is managed for optimal thermal gradients. The Spiral-Tube Steam Generators (STSGs), six in total, feature spiral-wound tubes and integrated vertical axial-flow pumps, designed to handle high temperatures efficiently with minimized pressure loss. The system includes two redundant Decay Heat Removal (DHR) systems for reliable heat dissipation, ensuring safety even under prolonged operation without make-up water.

(b) Reactor Core

The core consists of 61 hexagonal Fuel Assemblies (FAs), each containing 390 fuel pins. Power distribution is shaped using zones of varying plutonium enrichment, and the FAs are buoyantly supported with stems extending into the gas space, anchored to the Amphora-Shaped Inner Vessel (ASIV), which supports and shields the core from neutron damage.



Reactor roof main components layout

(c) Reactivity Control

The reactor employs 6 ex-core control rods positioned in the lead pool to manage reactivity and ensure safe operation. Two shutdown systems are in place: a gravity-driven absorber system for rapid shutdown and a redundant safety system with an absorber pin bundle that responds to coolant level changes.

(d) Reactor Pressure Vessel and Internals

The reactor pressure vessel is designed with a cylindrical shape and an oval bottom to house the core and other primary components. The ASIV, integral to the vessel, supports the core, separates hot and cold lead collectors, and protects against neutron damage.

(e) Reactor Coolant System and Steam Generator

The reactor coolant system uses lead flowing through the reactor and the Spiral-Tube Steam Generators (STSGs). The STSGs, with their spiral-wound tubes and integrated pumps, ensure efficient heat transfer while maintaining minimal pressure loss.

Scheme of the tube bundle of the plane-spiral-tube steam generator

(f) Pressuriser

The pressuriser maintains the reactor coolant system at the required pressure to prevent lead freezing and ensure proper coolant flow through the steam generators.

(g) Primary pumps

The pump is characterised 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.

5. Safety Features

(f) Engineered Safety System Approach and Configuration

The LFR-AS-200 employs a comprehensive engineered safety system that capitalizes on lead's exceptional cooling properties and high boiling point of 1737°C. The reactor's safety features include passive shutdown and decay heat removal systems that operate without external power or operator intervention, significantly reducing the risk of cyber-attacks. The design integrates multiple safety layers: passive heat removal via diverse redundant systems, robust structural components, and innovative steam generators to manage potential accidents effectively.

(g) Safety Approach and Configuration to Manage DBC

The LFR-AS-200 employs a comprehensive safety approach to manage Design Basis Conditions (DBC) by leveraging the unique properties of lead coolant and advanced reactor design. Its high boiling point and chemical inertness minimize the risk of coolant boiling and loss-of-coolant accidents. The reactor features passive safety systems for shutdown and decay heat removal, which operate without external power or operator intervention. A short, innovative steam generator design reduces lead displacement and manages pressure effectively, while redundant decay heat removal systems ensure continuous cooling. Ex-core control rods in the lead pool provide precise reactivity control, enhancing the reactor's stability and safety under various operational conditions.

(h) Safety Approach and Configuration to Manage DEC

For Design Extension Conditions (DEC), including severe accidents, the LFR-AS-200 is designed with robust passive safety features. The reactor employs two diverse, redundant decay heat removal systems, each consisting of lead-air and lead-lead coolers, which are activated by thermal expansion without the need for external power. In the event of an extended loss of power, the reactor's design ensures that lead coolant can maintain core temperatures within safe limits, and the reactor can be directly cooled by water if necessary. The system's resilience to extreme conditions is further supported by lead's high retention capability for fission products.

(i) Containment System

The design of confinement remains based on the classic principle of three consecutive, independent barriers, with fission gas purge tubing doubly contained. The "secondary confinement" is provided with a concrete containment, external-missileproof, and a safety vessel to eliminate loss of coolant accident (LOCA) in case of reactor vessel failure. The confinement lacks a dedicated cooling system, and the plant's passive characteristic excludes extensive core damage, such as the fall of a spent fuel during refueling. The plant should also exclude the need for an "emergency zone" to respect the radiological limits of incidental conditions.

(j) Spent Fuel Cooling Safety Approach / System

The LFR-AS-200 uses passive and active systems for spent fuel cooling, including redundant decay heat removal systems and passively activated lead and air coolers. This safety approach ensures continuous cooling and mitigates risks in the spent fuel storage system.

6. Plant Safety and Operational Performances

The LFR-AS-200 reactor is designed for high safety standards and efficient operational performance. It features passive safety systems, lead coolant, and robust containment to ensure the reactor remains cool even in the event of power loss or active cooling systems. The reactor's high thermal efficiency of around 42% net efficiency reduces heat released to the condenser, improving overall plant performance. The modular design allows for scalability and flexibility in operations, with each module operating independently or in conjunction with others. The reactor can operate in various modes, including base load and frequency control, to meet varying grid demands. The extended fuel cycle of up to 80 months reduces the frequency of refueling and maintenance activities. The plant is equipped with advanced instrumentation and control systems to monitor and manage reactor conditions in real-time. Comprehensive training programs for plant operators and routine inspections and maintenance ensure a high safety culture. The reactor's design also allows for power output adjustments without impacting fuel performance.

7. 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.

8. Plant Layout Arrangement

The LFR-AS-200 reactor is a modular design with two reactors housed in a compact building, optimizing space and safety. It features shared facilities, easy construction, and transport, with cooling options for different conditions. The design minimizes emergency zones and offers operational flexibility, allowing it to function in interconnected and isolated grids.

9. Testing Conducted for Design Verification and Validation

Testing for the LFR-AS-200 has involved both component and material evaluations. Newcleo has performed small-scale tests on new reactor components, such as the spiral-tube steam generator and innovative fuel assemblies, though full-scale testing remains necessary. To address corrosion concerns with lead coolant, advanced materials like Alumina-Forming Austenitic (AFA) steels and PLD coatings are undergoing rigorous non-nuclear testing. Additionally, the LFR-AS-30 irradiation reactor is being developed to test materials under realistic conditions, including high temperatures and neutron fluxes.

10. Design and Licensing Status

Newcleo is in the pre-licensing phase for its advanced LFR-AS-200 design, which incorporates novel features and materials. The final design and licensing are pending feedback and regulatory approvals, leveraging existing SFR experience.

11. Fuel Cycle Approach

The LFR-AS-200 reactor uses plutonium-based fuels initially and may include minor actinides in the future, aligning with national fuel use policies. Its core has a breeding ratio of 0.9, allowing it to produce fissile material as it consumes. The spent fuel is managed through storage and reprocessing, with a spent fuel pool for five years and dry storage provisions. The centralized off-site approach aims to minimize nuclear waste, enhance resource utilization, and reduce the environmental impact of nuclear energy.

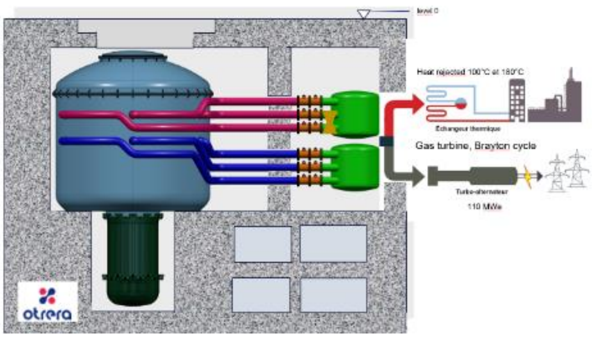
12. Waste Management and Disposal Plan

The LFR-AS-200 reactor has a comprehensive waste management and disposal plan to minimize waste production, ensure safe handling, and select appropriate disposal methods. The plan includes operational waste, including low- and intermediate-level radioactive waste, and spent fuel, containing high-level waste with long-lived isotopes. The reactor has a spent fuel pool for up to five years, dry storage systems for long-term storage, and reprocessing facilities for potential reuse or treatment. High-level waste is immobilized in deep geological repositories, while low- and intermediate-level waste is treated and disposed of in geological facilities. The plan complies with national and international regulations.

13. Development Milestones

2014	Completion of pre-conceptual design (by Hydromine Nuclear Energy S.àR.L)
2019	Completion of conceptual design
2021	Hydromine Nuclear Energy S.àR.L. is incorporate in newcleo for a fast development programme. Design restarted after an idle phase
2023	Start of pre-licencing discussions in France

OTRERA 300 (Otrera New Energy, France)



KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Otrera New Energy, France
Reactor type	SMR / SFR
Coolant/moderator	Sodium
Thermal/electrical capacity, MW(t)/MW(e)	300 MWth / 110 MWe
Primary circulation	Compact loop
NSSS Operating Pressure (primary/secondary), MPa	0.1 MPa / 18 MPa
Core Inlet/Outlet Coolant Temperature (°C)	400°C / 550°C
Fuel type/assembly array	MOX / Hexagonal
Number of fuel assemblies in the core	30
Fuel enrichment (%)	17%
Core Discharge Burnup	100 GWd/ton
Refuelling Cycle (months)	120 months
Reactivity control	Control rods
Approach to safety systems	
Design life (years)	40 years
Plant footprint (m^2)	< 30 000 m^2
RPV height/diameter (m)	5 m / 3 m
RPV weight (metric ton)	
Seismic Design (SSE)	
Distinguishing features	4th containment barriers Brayton Cycle
Design status	Conceptual Design

1. Introduction

OTRERA, a spin-off from CEA, designs, develops and operates a range of Generation IV nuclear reactors (SMR) to produce decarbonized electricity and heat, in particular the OTRERA 300, a sodium-cooled fast reactor (SFR) with a capacity of 110 MWe/300 MWth.
This sodium fast neutron technology is based on over 50 years of the French scientific, technical and industrial expertise, which has given rise to 3 reactors (Rapsodie, Phenix and Superphenix) and to the Astrid prototype project.
The key differentiators of the OTRERA 300 are:

- An innovative architecture including system simplification and additional 4th containment barrier.
- A modularity inspired by "New Space" enabling the development of a range of reactors from 30 to 450 MWe, while reducing costs and qualification period.
- Pre-industrialized sub-systems enabling an accelerated time to market by 2032.

The major challenge for OTRERA is to offer safe, reliable, and competitive electricity and heat to decarbonize the economy by 2032 and ensure the energy sovereignty of France and Europe.

2. Target Application

OTRERA 300 is a sodium-cooled fast neutron loop reactor. Primary sodium circuits are connected to a Brayton gas cycle without intermediate circuit. The gas cycle provides high temperature residual heat to 180°C for industrial end users. The simplicity, compactness and safety of the reactor are sought for installation close to industrial sites.

3. Design Philosophy

The research for an innovative architecture based on economical technological solution in terms of CAPEX and OPEX has made it possible to design a compact reactor with very high availability.
The architectural innovations concern:

- core performance allowing operating cycles of 10 years;
- safety improvements to simplify the need for safety devices including four containment barriers and underground reactor;
- outsourcing of fuel handling shared across several OTRERA reactors as well as specialized maintenance of sodium components.

Two primary vessels are used alternately in power mode and fuel assembly cooling mode and connected at one gas turbine.

4. Main Design Features

(a) Nuclear Steam Supply System

The energy conversion system uses a closed Brayton nitrogen gas cycle (18 MPa, 480°C / 530°C) and delivery electricity 110 MWe and controllable heat 180 MWth between 100°C and 180°C for industrial end users. If it is not possible to deliver heat to the end users, a nitrogen/air or nitrogen/oil or water cold source can be used to evacuate the power lost from the cycle.

(b) Reactor Core

The core is breakeven core with 30 S/A with hexagonal lattice. Each S/A is constitued with 217 MOX pins fuel. The core is managed in one batch, with a 10 years fuel time irradiation at nominal power. The average BU of spent fuel is 100 GWd/t and the DPA max less than 150.

(c) Reactivity Control

Control rods drive mechanism (CRDM) can be operated to control reactivity and safety margin. Complementary passive safety shut down rods are installed inside the core.

(d) Reactor Pressure Vessel and Internals

Reactor Vessel is composed of a cold collector outside the core limited with the primary vessel feeding the bottom of the vessel, an area inside which contain the core, separated with a downcomer and a hot collector upper the core. The design of the Reactor Vessel looks like a PWR Vessel without pressure.
The core is supported by a base plate and a strong deck anchored on the primary vessel. The operating temperature of the primary circuit is 400°C for entrance core and 550°C for outside core and a cooling flow rate of 7200 m3/h.

(e) Reactor Coolant System and Steam Generator

The primary circuit is composed of 6 sodium compact loops, each comprising an electromagnetic pump and a compact sodium-nitrogen heat exchanger. Primary circuit is installed inside a stell containment vessel filled with neutral gas.

(f) Pressuriser

No pressure in the primary sodium circuit

(g) Primary pumps

6 electromagnetic pumps are connected between Cold Collector vessel and the Compact Heat Sodium Gas Exchanger. The inertia of the pumps is ensured by an electric shaft.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The Reactor Vessel is located entirely below grade providing physical protection against aircraft hazards or missiles.

Passive decay heat removal from core and from spent fuel vessel is executed through heat exchange with dedicated passive heat exchangers which subsequently deliver the heat to the air.
3 passive heat decay heat removal systems are considered for the OTRERA design:

i. 2 independent sodium circuits with sodium/sodium heat exchanger in the primary hot and cold collector;
ii. 1 oil circuit around the safety vessel located in the reactor pit;
iii. 1 additional active oil circuit connected to the primary sodium purification system.

In the event of abnormal transients or postulated accidents, OTRERA 300 will employ non-safety active cooling systems as the first line of defence, failure of which will actuate the passive safety systems and the plant will shutdown and remain safely shutdown for an unlimited period without the need for power, make-up gas or operator actions. The core is configured as a basket with optimized flow paths (transverse and ascending) to reduce the risk of sub assembly blockage flow, and a upper hot collector vessel containing a large coolantinventory above the core. Therefore, even under an improbable large break LOCA the coreremains covered by a large inventory of sodium.
The OTRERA 300 design eliminates potential accidents and events that could cause coredamage or precipitate releases into the environment. The plant is designed to withstand the effectsof all natural phenomena including earthquakes and severe weather events.
The design of OTRERA 300 eliminated the primary sodium fire risk with neutral gas containment and the sodium water reaction using nitrogen as a secondary cooling system.

(b) Containment System

Control of the containment of radioactive materials is ensured by a system of 4 containment barriers: the fuel cladding; the primary circuit ; an added metal vessel surrounding the primary circuit; the concrete infrastructure that encases the primary circuit and vessels. Metal vessel is sized for internal abnormal transients or postulated accidents, the external concrete enclosure is sized for external abnormal transients or postulated accidents.

(c) Spent Fuel Cooling Safety Approach / System

The primary reactor vessel of the OTRERA 300 reactor also provides fuel cooling system after 10-year operating period. The same reactor's residual power removal systems are used for power reactor cooling mode.

6. Plant Safety and Operational Performances

With the alternating operation of two reactor blocks, OTRERA availability rate reaches 97%. The fuel loading and unloading operations are carried out in masked time as well as specialized maintenance on the reactor block components. The design of specialized maintenance operations is similar to dismantling operations that will be periodically carried out on the installation.

7. Instrumentation and Control Systems

Digital instrumentation and control systems (I&C) are adopted for OTRERA 300. The current Otrera design proposes using digital controls for operating each reactor from a control room installed in the electrical building. Reactor monitoring will be carried out in a centralized manner in order to optimize the maintenance needs of the reactor fleet.

8. Testing Conducted for Design Verification and Validation

The verification and validation of the OTRERA 300 design will be implemented throughout the design on mock up:

- Hydraulic model in water to simulate sodium flow in the reactor block;
- Full-scale prototypes of critical sodium components tested on a sodium loop in a representative environment;
- Series head rod control mechanisms in sodium tank.

9. Design and Licensing Status

OTRERA 300 meets international nuclear safety standards and requirements and which are based exclusively on proven technologies from the french development of fast neutron reactors during 50 years.

OTRERA 300 requires no major research and development for deployment, only industrial development. The detailed design stage would include industrial qualification of sub assemblies, control rods, compact heat exchanger and electromagnetic pump. Conceptual Design Phase is in progress. The pre-licensing phase will begin at the end of 2024 through exchange meeting with the safety administration.

10. Fuel Cycle Approach

The fuel management of the OTRERA SFR reactor fleet will enable the multirecycling of plutonium and the closure of the fuel cycle, lying on the french industrial experience for decades. The spent fuel will be, after a few years of cooling in the the facility, transported to the reprocessing facility of ORANO La Hague (France) where the fissile material (e.g. uranium and plutonium) will be separated from the fission products and other actinides. Plutonium, reprocessed Uranium as well as depleted uranium (by-product of front end enrichment), will be reused for the fabrication process of fuel pellets. The OTRERA SFR will greatly extend the uranium resources compared to thermal reactors, multiplying by almost two orders of magnitude the ability to produce energy from all types of uranium and not only the classical LEU used in the PWR. Thus, the OTRERA SFR fleet will continue to produce energy long after the running out of natural uranium.

11. Waste Management and Disposal Plan

Specialized maintenance operations are carried out periodically on the sodium components of the reactor blocks after removal of the sodium and transfer to the OTRERA decentralized hot facility. Liquid effluents are treated online on the installation and then transferred to the centralized hot facility for treatment.

12. Development Milestones

2022 – 2023	Preliminary studies and technological innovation (using previously developed patents).	Complete
2023 – 2024	Pre-conceptual design phase and technology validation	Complete
2024 – 2025	Conceptual Design Phase (and preparation for pre-licensing)	Planned
2025 – 2027	Basic Design Phase	Planned
2027 – 2030	Detailed Design Phase	Planned
By 2030	Target first concrete for the FOAK plant	Planned

SEALER-55 (Blykalla, Sweden)

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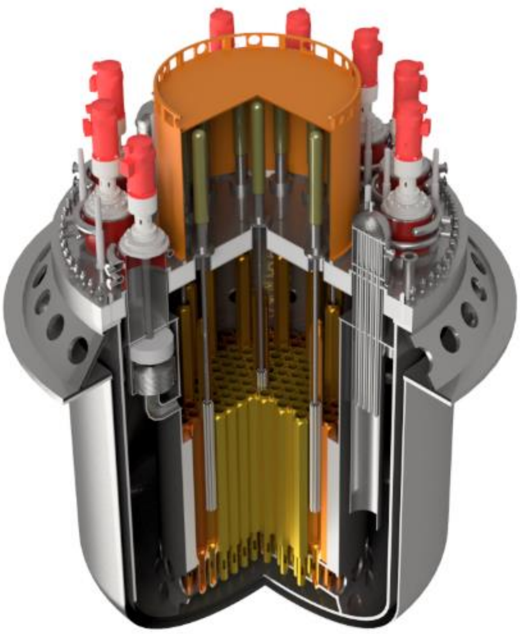

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Blykalla, Sweden
Reactor type	Lead-cooled SMR
Coolant	Lead
Thermal/electrical capacity, MW(t)/MW(e)	140 MWt / 55 MWe
Primary circulation	Forced
NSSS operating pressure (primary/secondary), MPa	Non-pressurised
Core inlet/outlet coolant temperature (°C)	420 / 550
Fuel type/assembly array	UN/Hex-can
Number of fuel assemblies in the core	84
Fuel enrichment (%)	12
Refuelling cycle (months)	300
Core discharge burnup (GWd/ton)	60
Reactivity control mechanism	B_4C, WB_2
Approach to safety systems	Passive
Design life (years)	28
Plant footprint (m^2)	600 / 20 000 (single unit/fence)
RPV height/diameter (m)	5.5 / 4.8
RPV weight (metric ton)	20
Seismic design (SSE)	None
Fuel cycle requirements / approach	TBD
Distinguishing features	Nuclear battery
Design status	Conceptual design

1. Introduction

SEALER-55 is designed by Blykalla for on-grid, commercial power production. It is intended to be fabricated in series of more than 100 units in an automated factory.

2. Target Application

The primary market for SEALER-55 is existing nuclear power sites, where several units can be clustered to replace the capacity of existing light water reactors. Other markets to be considered are large consumers of high temperature steam, such as hydrogen and bio-oil/bio-char plants. Remote and marine applications (mining, shipping) may also be considered.

3. Design Philosophy

Since small reactors in general have high specific costs for hardware and personnel, Blykalla 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
• Passive removal of residual heat using natural convection of liquid lead and of air.
• Minimisation of reactivity swing, hence control-rod bank worth, by application of 12% enriched UN fuel.

4. Main Design Features

(a) Nuclear Steam Supply System
Rankine cycle.

(b) Reactor Core
The core of SEALER-55 features 84 fuel assemblies containing 169 UN or (U,Hf)N rods, two B4C control rod assemblies, ten shut-down rod assemblies using WB2/B4C absorbers and 66 ZrO2 reflector assemblies. The active height is 1.3 m. The rod diameter, pitch and channel length have been optimised using an analytical approach published in Annals of Nuclear Energy.

(c) Reactivity Control
Reactivity control is accomplished by two B4C control rod assemblies and ten WB2 shut-down rod assemblies. The latter are inserted passively by gravity.

(d) Reactor Pressure Vessel and Internals
The primary vessel is manufactured from SS316 with an overlay weld of alumina forming austenite (AFA), for corrosion protection. The wall thickness is 40 mm, diameter 4.8 m, and height of the cylindrical section 4.3 meters.

(e) Reactor Coolant System
During normal operation conditions, forced circulation of the lead coolant is provided by variable speed reactor coolant pumps. Eight steam generators transfer the heat to the secondary system.

(f) Secondary System
Rankine steam cycle at 165 bar and T = 350-530°C.

(g) Steam Generator
Spiral tube, where the tube is a compound of Alloy 690 on the steam side and Fe-10Cr-4Al on the primary side.

5. Safety Features

(a) Engineered Safety System Approach and Configuration
Ten shut-down rod assemblies are parked above the core during nominal operation. These are divided into two diversified banks, each being capable of taking the core sub-critical by more than 1$. The first consists of high density WB2 absorber pellet rods (96% B-10 enriched), which will be inserted by gravity. The second bank consists of 96% B-10 enriched B4C pellets, which will be inserted by a spring-load mechanism.

(b) Decay Heat Removal System / Reactor Cooling Philosophy
In case the secondary system is unavailable for decay heat removal, a reactor vessel auxiliary cooling system based on natural circulation of air shall ensure that fuel rod cladding temperatures never exceed their rapid creep failure limit.

(c) Spent Fuel Cooling Safety Approach / System
The fuel resides in the primary system during the entire life of the plant. Following shut-down of the plant, the fuel will cool in-situ until the primary system is removed from site.

(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.

(e) Chemical Control
An oxygen control system consisting of oxygen sensors and oxygen pumps is ensuring that the concentration of oxygen in the primary coolant is such that lead oxide precipitation will not occur, and that a sufficient amount of oxygen is available for self-healing of any damage that may be inflicted on

alumina forming steel surfaces. All surfaces exposed to lead consist of alumina forming steels (ferritic, austenitic, or martensitic) which have been proven to be highly corrosion tolerant.

6. Plant Safety and Operational Performances

SEALER-55 is designed to provide a passively safe, secure and reliable power source on-grid or industrial site applications. The reactor is able to produce 55 MW of electric power for 25 full power years without reloading nor reshuffling of its UN fuel. A capacity factor of 95% is foreseen, permitting preventive maintenance exchange of pump and steam generator modules with a frequency of 1/year. No fuel reload is foreseen for the entire life of the reactor (25 full power years). The application of alumina alloyed steels provides corrosion protection that is deemed sufficient over the life of the reactor. Safety analysis shows that as designed, SEALER can survive unprotected loss of flow, loss of heat sink 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.

7. Instrumentation and Control System

The primary system is instrumented in order to measure neutron flux, temperature, oxygen concentration and hot leg lead free level position.

8. Plant Layout Arrangement

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, connected to the guard vessel. 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.

9. Testing Conducted for Design Verification and Validation

An electrically heated prototype of SEALER-55 will be built and operated in Oskarshamn, Sweden for the purpose of validating its safety concept, operational and maintenance procedures, as well as materials performance. The prototype is designed with a power of 1,2 MW, produced in one heated rod assembly with 127 rods. The height of the prototype is 1:1 with respect to the SEALER-55, in order to validate residual heat removal capability with lead and air. The prototype is currently under engineering design, with the intent to have it in operation in 2025.

10. Design and Licensing Status

Interactions with the Swedish Radiation Safety Authority (SSM) have taken place, and a regulatory path towards licensing of a lead-cooled research/demonstration reactor in Sweden has been identified. This path is based *on the Swedish system and requirements* with a combination of IAEA guidelines for research reactors, and existing regulation for light water reactors, when the latter might be applicable. Moreover, the Swedish government has given the task to SSM to develop a framework for licensing of SMRs.

11. Fuel Cycle Approach

The uranium nitride fuel of SEALER-55 will be fabricated by direct ammonolysis of 12% enriched UF_6, followed by spark plasma sintering of pellets.

The spent fuel of SEALER-55 reactors may either by disposed in a geological repository, or recycled. In the former case, disposal of entire cores in frozen lead, conversion of irradiated UN to UO_2 pellets or direct disposal of UN fuel rod assemblies will be evaluated.

12. Waste Management and Disposal Plan

See fuel cycle approach.

13. Development Milestones

2018	Concept design of SEALER-UK funded by UK BEIS
2020	SUNRISE project, including design of a demonstration unit funded by SSF.

2021	Blykalla and Uniper form joint venture SMR AB
2022	The SOLSTICE project including design, construction and operation of the electrically heated mock-up SEALER-E in Oskarshamn is funded by the Swedish Energy Agency.
2024	The construction of the SEALER-E electrical mockup starts. SEALER-E is commissioned.
2025	Supporting documentation for the licensing of the SEALER-One nuclear demonstration unit is completed
2030	The SEALER-One nuclear demonstration unit is taken into operation

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

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Representative picture of reactor

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	AKME-engineering JSC, Russian Federation
Reactor type	LFR (Lead-Bismuth Fast Reactor)
Coolant/moderator	PbBi / None
Thermal/electrical capacity, MW(t)/MW(e)	280 MW(t) / 100 MW(e) (condensing mode); 77.2 MW(e) (co-generation mode)
Primary circulation	Forced (2 pumps)
NSSS Operating Pressure (primary/secondary), MPa	0.11 / 7
Core Inlet/Outlet Coolant Temperature (°C)	335 / 477
Fuel type/assembly array	UO_2 (Oxide) / Triangular array
Number of fuel assemblies in the core	61
Fuel enrichment (%)	16.7% (average) / 19.5% (maximum)
Core Discharge Burnup (GWd/ton)	~65 GWd/ton
Refuelling Cycle (months)	72-84 or continuous
Reactivity control	Control rods with absorbing elements (Boron Carbide)
Approach to safety systems	Passive heat removal system (PHRS), integral NSSS layout
Design life (years)	50
Plant footprint (m²)	34 000
RPV height/diameter (m)	6.53 / 4.40
RPV weight (metric ton)	280 (transport weight)
Seismic Design (SSE)	Reactor: 0.25 g; buildings: 0.12 g
Distinguishing features	Modular design, capable of factory manufacturing and transportation to the site, passive safety features
Design status	Detailed Design

1. Introduction

SVBR-100 is a multipurpose small modular lead-bismuth (LBE) cooled fast reactor with an equivalent electric capacity of 100 MW. The design is based on dozens of reactor-years of experience in LBE cooled reactor operation. Its main features are:

- Enhanced inherent safety and passive safety systems as well as significant simplification of the design of the reactor and the entire nuclear plant that results in competitive economics;
- Capability to operate with different types of fuel in both open and closed nuclear fuel cycle with refuelling every 6 - 7 years;
- Compact design as well as maximum prefabrication of the reactor and its transportability, including railway;
- Capability to design modular NPPs and energy complexes with installed capacity being a multiple of the number of reactor units.

2. Target Application

Multi-purpose application of standardized modular nuclear energy complexes of different capacities (100-600 MW(e)) equipped with SVBR-100 reactors for mature, emerging and prospective goods and services markets: power, heat (process heat and centralized district heating), cooling, desalination, hydrogen and burning minor actinides.

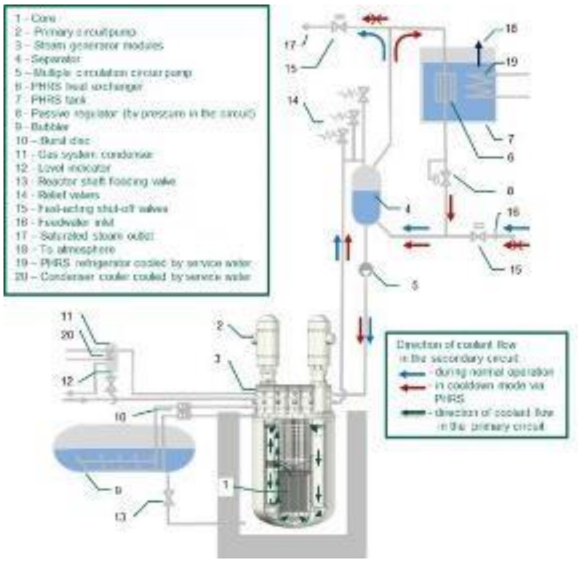
3. Design Philosophy

The SVBR-100 reactor is a safety-focused, modular, and flexible system that integrates primary circuit components into a single monoblock, reducing accident risk. It uses multiple fuel types and operates in open and closed cycles, making it suitable for power generation, process heat, desalination, and co-generation.

4. Main Design Features

(a) Nuclear Steam Supply System

The SVBR-100's NSSS is housed in a monoblock vessel, housing all primary circuit components, including the core, pumps, and steam generators. The steam generator has 12 evaporation modules, similar to PWRs, and no external primary circuit piping.



The basic hydraulic diagram of the NSSS SVBR-100

(b) Reactor Core

At the first stage it will be the mastered uranium oxide fuel. SVBR-100 reactor pursues resistance to nuclear fissile material proliferation by using uranium with enrichment below 20%. The reactor is designed to operate with 6 -7 years between refueling. Maximum local burnup in the FOAK design is about 10%. At the next stage MOX, uranium nitride or mixed uranium-plutonium nitride fuel can also be used.

(c) Reactivity Control

The system uses control rods for reactivity control, with the strongest absorbing rod having less than 0.5 βeff efficiency, preventing neutron criticality. A passive emergency protection system ensures reliable shutdown in anomalies.

(d) Reactor Pressure Vessel and Internals

The pressure vessel houses primary components and is protected by a casing. It efficiently transfers heat without high-pressure systems using lead-bismuth eutectic coolant near atmospheric pressure. It can withstand seismic events and shock waves.

(e) Reactor Coolant System and Steam Generator

The primary coolant is lead-bismuth eutectic (LBE), a low-pressure, high-thermal capacity material that allows efficient heat removal and inherent safety by eliminating the risk of pressure-driven coolant loss. Steam generators are incorporated within the monoblock, using multiple forced circulation loops for reliable steam production under normal and emergency conditions.

(f) Pressuriser

The system operates near atmospheric pressure, eliminating the need for a traditional pressurizer, simplifying the process and reducing the risk of over-pressurization.

(g) Primary pumps

The reactor vessel utilizes electromechanical pumps for coolant circulation, ensuring long-term reliability and integration into the primary system.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The SVBR-100 reactor uses passive safety features, allowing natural circulation to remove decay heat even in blackout event. It features a gas-vapor mixture receiver system to manage leaks or tube ruptures in steam generator, automatically discharging any mixture into a bubbler tank.

(b) Safety Approach and Configuration to Manage DBC

The reactor's integral design ensures the containment of all radioactive materials within the vessel, allowing for depressurization without significant pressure rise, and passive heat removal systems (PHRS) provide cooling during design-basis accidents without the need for active systems.

(c) Safety Approach and Configuration to Manage DEC

The LBE coolant's inherent properties prevent catastrophic failures or barrier destruction in severe accidents. The reactor design eliminates coolant leakage with no external radioactive loops. Fast neutron reactor physics, low burnup, and negative reactivity effects ensure stability in severe scenarios.

(d) Containment System

If the primary circuit is depressurized, the pressure in the reactor compartment does not increase. The reactor compartment is designed for an accident with depressurization of the second circuit with a non-radioactive water-steam mixture with a pressure of 7 MPa, and also for a set of external effects (seismic, shock wave, plane crash).

(e) Spent Fuel Cooling Safety Approach / System

The system cools spent fuel in dry storage after unloading it at the end of its life cycle, ensuring safe storage without overheating or radioactive material leakage due to its ability to handle high decay heat levels.

6. Plant Safety and Operational Performances

The SVBR-100 reactor is designed for high operational performance and inherent safety. It operates at near-atmospheric pressure due to its lead-bismuth eutectic (LBE) coolant, minimizing the risk of high-energy accidents. Passive safety features, such as natural circulation for decay heat removal, ensure continued operation without the need for active safety systems. The compact reactor core and the integration of primary components within a single vessel enhance both operational reliability and plant safety. The long refueling intervals (~6-7 years) improve the reactor's uptime and decrease maintenance needs. The reactor’s core reactivity and thermohydraulic design prevent unauthorized material use, contributing to non-proliferation. The plant is designed to withstand external hazards, such as seismic events and crashes, ensuring robust protection of the core and surrounding infrastructure.

7. Instrumentation and Control Systems

These systems and equipments use industry well-mastered solutions; the specific execution depends on the placement site, regulatory requirements, and customer requirements.

8. Plant Layout Arrangement

The FOAK NPP will be a single-module with installed capacity of 100 MWe. The reactor compartment is monolithic reinforced concrete, the reactor is separated from the external environment by two reinforced concrete shells: the outer one without cladding and the inner one lined with metal cladding. The turbine compartment is an above-ground part of a metal frame. The controlled access zone is in the reactor compartment and the RW processing

The layout of the FOAK NPP

building. Spent fuel dry storages are part of the reactor compartment. A evaporative cooling tower is used for the technical water supply of the NPP. The output electric energy parameters from NPP into grid are ~89 MW, 50 Hz, 110 kV. NPP own needs (~11 MW) are powered by a turbogenerator and an external 110 kV grid. If regular energy supply is lost, the backup energy supply (batteries, diesel generators) is automatically connected. The NOAK NPP will have from one to six modules, depending on a particular customer need.

9. Testing Conducted for Design Verification and Validation

The SVBR-100 design has undergone extensive testing and R&D, including fuel behavior and burnup performance tests, thermohydraulic experiments, structural integrity testing in LBE environments, safety tests for passive heat removal systems, and control and protection system testing. Further R&D focuses on certification of structural materials, safety justification, and real-world fuel testing.

10. Design and Licensing Status

The SVBR-100 reactor has undergone significant development in compliance with Russian Federation Government Resolution No. 87 (February 2008). The FOAK design documentation has been developed, and the siting license for the first plant was issued in 2015. Over 60 patent families have been successfully granted, securing the intellectual property behind the SVBR-100 design. Engagement with regulatory bodies is ongoing, including the development of a regulatory framework and certification programs to ensure international compliance.

11. Fuel Cycle Approach

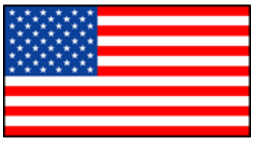
The SVBR-100 reactor is versatile, operating with both open and closed nuclear fuel cycles. It uses uranium oxide fuel with a non-proliferation compliance and can operate with MOX fuel or mixed uranium-plutonium nitride fuels. In a closed fuel cycle, its breeding ratio minimizes the need for fresh uranium, making it cost-effective and stable. Its core reactivity is sensitive to fertile material additions.

12. Waste Management and Disposal Plan

The SVBR-100 uses a spent fuel handling system, storing spent fuel assemblies in dry storage after their 6-7-year life cycle. The reactor's high burnup capability reduces spent fuel volume, and it must be cooled using specialized casks for long-term disposal. The closed fuel cycle option minimizes radioactive waste generation, and the system is designed to handle high-level waste safely and prevent environmental contamination.

13. Development Milestones

2009	Set up of AKME-engineering JSC – the project operator;	Complete
2015	Release of the design documentation of the FOAK NPP;	Complete
2016 – 2024	Prospecting partners in countries where the project can be potentially deployed;	On track
2029	To complete R&D and obtain a construction license for the FOAK NPP;	Planned
2029	To complete the construction and commission the FOAK NPP as well as to obtain the operation license.	Planned



The Natrium™ Project (TerraPower, LLC)

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	TerraPower, LLC USA
Reactor type	Fast neutron spectrum
Coolant/moderator	liquid metal sodium
Thermal/electrical capacity, MW(t)/MW(e)	840 / 345
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Low pressure/atmospheric
Core Inlet/Outlet Coolant Temperature (°C)	540
Fuel type/assembly array	Metallic Uranium Alloy HALEU fuel/hexagonal
Fuel enrichment (%)	Maximum of 19.75%
Core Discharge Burnup (GWd/ton)	150 GWd/ton
Refuelling Cycle (months)	18-24
Reactivity control	control rod drive mechanism
Approach to safety systems	passive
Design life (years)	60
RPV height/diameter (m)	20.4/11.6
RPV weight (metric ton)	2450
Seismic Design (SSE)	>0.3g
Distinguishing features	Molten salt, thermal integrated energy storage system, providing built-in gigawatt-scale energy storage
Design status	Detailed Design

1. Introduction

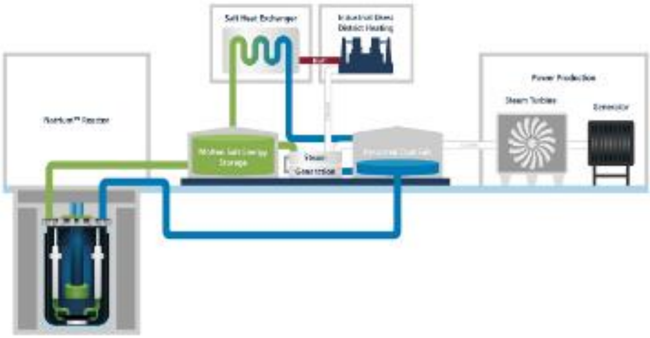
TerraPower, with the support of GEH, is developing the Natrium™ reactor, a 345-megawatt sodium cooled fast reactor coupled with a molten salt-based thermal integrated energy storage system that will provide clean, flexible, and reliable energy and stability for the grid. Utilizing our thermal energy storage system, the Natrium plant can flex up to 500 MWe for up to 5.5 hours.

2. Target Application

TerraPower will deploy the Natrium technology to contribute to the production of carbon-free electricity, heat, and steam to support the decarbonization of power and industrial sectors.

3. Design Philosophy

The Natrium Reactor Plant is an 840 MWt/345 MWe pool-type sodium fast reactor that contains a compact and simple safety envelope. The reactor is integrated with our transformative innovation--a molten salt thermal energy storage system which enables the plant to vary its supply of energy to the grid, up to 500 megawatts electric (MWe) net, while maintaining constant reactor power. The system can boost output to 500 MWe for more than five and a half hours to serve peak demand. Natrium's energy storage system can also fully discharge and recharge three times per day, orders of magnitude faster than grid-scale batteries currently on the market. The Natrium reactor maintains its thermal reactor power constant during its entire operating period, eliminating reactor power cycling and maximizing its capacity factor and value. The technology provides dispatchable power at a scale that can make a difference in efforts to decarbonize electricity and stabilize grids with high penetrations of renewables in addition to having the capability of providing heat and steam for industrial decarbonization.



4. Main Design Features

(a) Nuclear Steam Supply System

Heat is transferred from the reactor core to the primary loop within the reactor pool and further to an intermediate sodium piping loop via intermediate heat exchangers located in the reactor pool. The intermediate system transports reactor heat from the intermediate heat exchangers to the salt transport system via sodium-salt heat exchangers. The salt transport system moves heat to the hot salt storage tank for thermal energy storage. The heat stored in the hot salt tank is used to generate steam to provide electricity or heat for industrial decarbonization.

(b) Reactor Core

Because it is a fast neutron spectrum reactor, the Natrium reactor is fueled by high assay low enriched uranium (HALEU). The fuel employs a uranium (U) metal fuel design instead of a uranium oxide design; the HALEU enrichment level of the fuel is up to 19.75 % weight 235U. The reactor core comprises various removable core assemblies (e.g., fuel, control, reflectors, shields, etc.). Thermal power and reactivity are controlled by a combination of fuel management operations on a cycle-by-cycle basis and the insertion and withdrawal of control rods throughout a cycle.

(c) Reactivity Control

Reactivity control is performed by control drive mechanisms on the top of the reactor head that move drivelines and control absorber bundles within the core. There two types of control absorber bundles that provide diversity in design and multiple instances of each control rod within the reactor.

(d) Reactor Pressure Vessel and Internals

The reactor pressure vessel and internals are supported by the reactor enclosure system (RES), which supports the reactor core and contains the majority of the primary sodium coolant. The RES includes the reactor vessel and head, reactor internal structures, the guard vessel, rotatable plug assembly (RPA) and the reactor support structures. The RES provides radionuclide retention, reactivity control, and reactor heat removal functions during normal and off-normal conditions, providing functional containment barriers, reactivity control, and reatctor heat removal need for primary coolant circulation within the reactor vessel. The reactor internals and core barrel structures direct the flow of primary sodium to the core and to heat transfer surfaces in the Intermediate Heat Exchanger (IHX) and RV walls for power generation and decay heat removal.

(e) Reactor Coolant System and Steam Generator

The Natrium Reactor is a pool-type sodium fast reactor. The entire primary coolant circuit is located within its reactor vessel resulting in no penetrations into the primary coolant boundary below the reactor head, minimizing the probability of loss of coolant accidents.

(f) Pressuriser

TerraPower's Natrium design is a low-pressure system that does not incorporate a pressurizer component.

(g) Primary pumps

The Natrium Reactor primary sodium pumps (PSPs) are part of the primary heat transport (PHT) system which transports heat from the reactor core to the intermediate heat transport (IHT) system in a closed-cycle flow path driven by the PSPs during normal operation, or natural circulation between hot and cold pools when not in operation. The Natrium Reactor PSPs supply sodium coolant flow to the core inlet plenum before exiting the core, mixing in the hot pool volume within the reactor vessel, and flowing past the RES Upper Internal Structure. The primary sodium then flows into the PHT IHX, where heat is transferred to the intermediate sodium. The flow circuit is completed as the cooled primary sodium at the outlet of the IHXs enters the PSPs and discharged back into the RES inlet plenum.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The Natrium reactor has a number of inherent safety features that are based on decades of research by the United States Department of Energy (DOE) as well as other countries. The Natrium reactor is designed so that it can passively shutdown from abnormal conditions and maintain a safe, stable state with no need of operator action, AC power, or control systems. These self-protecting safety features include gravity-driven control rod SCRAM, in-vessel primary sodium heat transport, reactor air cooling and intermediate air-cooling natural draft flow, and low primary and secondary pressure. The Natrium reactor includes both passive and inherent safety along all three fundamental safety functions for sodium fast reactors: Control, Cool, and Contain. The emergency decay heat removal system is always in operation providing extreme reliability and simplicity since the system has no moving parts and uses an infinite heat sink (air). The low-pressure design of the decay heat removal systems, elimination of loss of coolant accidents, and the retention of radionuclides by liquid sodium allow for a functional containment approach. The Natrium Reactor design implements passive safety systems that ensure reliable shut-down with no operator action required.

(b) Safety Approach and Configuration to Manage DBC

The Natrium Reactor Plant considers a diverse set of passive barriers that are credited in a functional containment strategy. The strategy considers all SSCs between the fuel, as the radionuclide source, and the environment. The cladding that is inherent to the fuel design for the Natrium reactor provides the first radionuclide retention barrier and a low leakage and low-pressure Reactor Enclosure System serves as the primary coolant boundary. Decay heat removal is achieved via passive air convection facilitated by the Reactor Air Cooling system and natural circulation of sodium through the primary sodium system. Reactivity control is achieved through failsafe functions that ensure gravity actuated control rods are inserted into the core for safe shutdown.

(c) Safety Approach and Configuration to Manage DEC

The Natrium Reactor Plant has multiple passive, active, and inherent features to protect against extension conditions. Additional barriers outside of the primary coolant boundary include physical building spaces, seals, and HVAC controls that form containment envelopes to support functional containment success. In addition to physical barriers, inherent plant design features, such as the high boiling point of sodium, ensures that the core remains submerged in at near atmospheric pressure during adverse conditions and further mitigates the transport mechanisms for radionuclides. The intermediate sodium systems can operate in passive modes via natural circulation of sodium to provide an alternative means for decay heat removal from the reactor air cooling system, or active modes to facilitate sodium circulation and forced heat removal. Control rod insertion is assured via control rod drive motors that drive the control rods into the core following SCRAM. In addition, the Natrium fuel design features inherent reactivity feedback that results in a stable power level at which heat production and heat removal are in balance.

(d) Containment System

The Natrium Reactor utilizes a near-atmospheric pressure functional containment system design that provides three fundamental safety functions for the control of reactivity, heat removal, and retainment of radionuclides. The functional containment design offers an additional confinement barrier in the unlikely instance of an abnormal event. Control of reactivity is gravity-driven and inherently stable with increased power or temperature, immediate air cooling offers passive and consistent heat removal, and our low primary and secondary pressure system offers multiple radionuclide retention boundaries.

(e) Spent Fuel Cooling Safety Approach / System

Spent fuel is stored in an outer ring (In-Vessel Storage) for 1-2 cycles, where it is moved back into the fuel transfer lift and up to the operating deck. The spent fuel is grappled by the ex-vessel handling machine (EXHM) and transported to ex-vessel storage tank (EVST). The EVST's principal purpose is to permit shorter refueling outages by providing sodium “buffer” storage of spent fuel before it it transferred into water storage, similar to light water reactor (LWR) spent fuel pools.

6. Instrumentation and Control Systems

The Natrium I&C Architecture evolved from extensive operating history and experience with previous I&C designs to meet the specific requirements for the TerraPower Natrium project. The I&C system architecture incorporates the following fundamental I&C design principles consistent with Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews that include independence, communications and logical independence, redundancy, diversity, system integrity, reliability, and human factors engineering. Safety-related systems are physically and electrically isolated, as well as independent from NSRST and NST systems, to ensure no adverse interaction can compromise the functions of the SR systems. Independent communication networks ensure a reliable flow of data communications, thereby incorporating a defensive cyber-security architecture. Redundancy functions ensure that single failure with not result in the failure of any SR function, and that appropriate levels of reliability are achieved for SR, NSRST, and NST functions. The Natrium Reactor’s I&C systems also use deterministic principles to ensure predictable and reliable operation, ensuring that each and every function is executed in every logic processing cycle continuously.

7. Plant Layout Arrangement

The Natrium Reactor Plant layout contains a Nuclear Island and an Energy Island. The Nuclear Island contains the Reactor Building, Reactor Auxiliary Building, Fuel Handling Building and the Control Building. The Energy Island contains the thermal storage tanks, the Steam Generation Facility, the Turbine Facility, switchyard and the cooling tower. The Natrium Reactor Plant layout has been developed to provide siting and deployment flexibility without affecting plant costs or efficiency. The Nuclear Island and Energy Island are decoupled, and only the Nuclear Island is subject to United States Nuclear Regulatory Commission regulation. Therefore, the spacing between these two islands can be changed site-to-site without any impact on site operations or the safety case. The Natrium reactor is designed to be deployed in multi-unit configurations. Dual unit configuration is the preferred commercial configuration, with quad and other higher MWe output configurations possible.

8. Testing Conducted for Design Verification and Validation

As part of the DOE Advanced Reactor Demonstration Program (ARDP), TerraPower is engaging in an intensive testing program to demonstrate the technology prior to start up, including full scale testing of selected components such as the primary sodium pump, the control rod drive mechanisms, and the in-vessel fuel handling equipment. In addition, TerraPower is building a large-scale sodium testing facility adjacent to the Kemmerer power plant to perform this testing.

9. Design and Licensing Status

TerraPower completed 60 pre-application meetings, submitted 17 Topical Reports, three Technical Reports and six White Papers/informational papers. On March 20, 2024, NRC completed a pre-readiness assessment for the CPA and no gaps were identified. TerraPower submitted the Construction Permit Application (CPA) for Kemmerer Power Station Unit 1 to the NRC on March 28, 2024. The NRC accepted the submittal for review on May 21, 2024. On June 12, the NRC provided the following information on its review schedule:

- Draft Safety Evaluation March 2025
- Advanced Safety Evaluation November 2025
- Final Safety Evaluation August 2026

The Draft Environmental Impact Statement (EIS) will be completed July 2025 and the Final EIS will be completed by May 2026. TerraPower anticipates submitting the Operating License Application and Final Safety Analysis Report by October 2027 and anticipate Operating License Issuance by April 2030.

10. Fuel Cycle Approach

The refueling of the Natrium reactor is outage-based; baseline fuel is reloaded in 1/3 core batches, or 54 assemblies. In addition, six to seven control rod assemblies and a number of shield and reflector assemblies also accompany each outage. Initially, plants will burn fuel for 1050 EFPD over 12-month cycles, while subsequent advanced fuel types will have EFPD of 3500 to 4200 EFPD (zone-based) over 24-month cycles. The advanced fuel has an equilibrium discharge of 30 fuel assemblies per cycle with a slightly larger fuel load of 168 positions. The advanced core swaps all 13 control assemblies and a higher proportion of shields and reflectors due to the longer cycle length.

11. Waste Management and Disposal Plan

The Project will generate solid low-level radioactive wastes through normal plant operation, including anticipated operational occurrences. These wastes will be addressed via a case-by-case treatment and the disposal process will be tailored to the waste stream. Spent fuel will remain in the spent fuel pool approximately 10 years, enough time for the decay heat to support loading in dry storage canisters. Nothing specific to the Natrium technology is foreseen that would prevent the ability to complete removal of all spent fuel from the spent fuel pool and emplacement in dry cask storage within 60 years of the end of the reactor’s life. Although not yet designed, structures and equipment will be made available for fuel packaging and continued onsite storage. Handling of transuranic and GTCC waste is managed by the fuel transport and storage system. Those types of waste would most likely be stored onsite in a facility like an ISFSI and ultimately dealt with at the time of facility decommissioning.

12. Development Milestones

2008 – 2016	Preliminary studies and technological innovation (using previously developed patents).	Complete
2017 – 2019	Pre-conceptual design phase and technology validation	Complete
2019 – 2023	Conceptual Design Phase (and preparation for pre-licensing)	Complete
2023 – 2025	Basic Design Phase	On track
From 2025	Commercialisation	Planned
2025 – 2026	Detailed Design Phase	Planned
2026	Target first concrete for the FOAK plant	Planned

MOLTEN SALT SMALL MODULAR REACTORS

Copenhagen Atomics Waste Burner (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 / N/A
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.05 – 0.25 / 0.1 – 0.25
Core Inlet/Outlet Coolant Temperature (°C)	600 / 650 - 700
Fuel type/assembly array	$^{7}LiF-UF_4$ or $^{7}LiF-ThF_4-(TRU)F_3$ / none
Power conversion process	Heat source
Fuel enrichment (%)	LEU or TRU/RGPu
Refuelling cycle (months)	N/A
Core Discharge Burnup (GWd/ton)	900 – 1000
Reactivity control mechanism	Heavy water level adjustment
Approach to safety systems	Passive
Design life (years)	5 years for the reactor vessel, minimum 50 years for the surrounding building, and unlimited lifetime for the salts and heavy water.
Plant footprint (m²)	50 000 per 2.5GW(th) plant
RPV height/diameter (m)	12 / 2.55
RPV weight (metric ton)	40
Seismic design (SSE)	N/A
Fuel cycle requirements / approach	LEU or transuranic initiated / conversion to Th-U cycle
Distinguishing features	Liquid moderator, Low fissile inventory, and Potential for breeding
Design status	Detailed design / Equipment manufacturing in progress

1. Introduction

The Copenhagen Atomics Waste Burner is an autonomous small single module 100 MW(t) heavy water moderated, fluoride salt based, thermal spectrum, molten salt reactor the size of a 40 feet shipping container. The reactor doesn't require refuelling, human intervention, or maintenance for its 5-year design life and operates completely autonomously with passive decay heat removal. The salts and heavy moderator have, in principle, unlimited lifetime and will be reused in subsequent reactors, while the reactor vessel is the consumable and fission products are left as a by-product.

The Copenhagen Atomics Waste Burner can burn transuranics and has the potential to transition to breeding within 3 years of operation. The core and liquid moderator, fuel salt loop, blanket salt loop, coolant salt loop, fission product separation systems, dump tank, heat exchangers, and pumps, are all contained in a leak-tight steel containment, the size of a 40 feet shipping container, which in tern is placed inside a leak-tight steel 'cocoon' together with auxiliary equipment. In this way the Copenhagen Atomics Waste Burner employs three barriers to nature.

2. Target Application

Copenhagen Atomics will design, licence, build, operate, and decommission all Copenhagen Atomics Waste Burners. Each reactor delivers heat as a service, in the form of up to 560°C nitrate salt, to the customer. In order to facilitate early mass deployment, Copenhagen Atomics plans to initially sell to customers that need double digit GW thermal capacity to produce commodities, such as ammonia, steel, aluminium, and desalinated water.

3. Design Philosophy

Copenhagen Atomics see autonomous operation without maintenance and road transportable completely factory assembled single unit reactors as essential for mass deployment and scaling of nuclear power to terawatt levels. To this end Copenhagen Atomics are developing thermal spectrum reactors with high specific fissile inventory (thermal power to total fissile inventory) and online fission product separation and targeting low enriched uranium and burning of spent nuclear fuel transuranics, as kickstarter fuel for a thermal spectrum thorium breeding cycle.

Copenhagen Atomics is focused on building and testing ‘minimal viable product’ iterations of our Copenhagen Atomics Waste Burners before going through a commercial licensing process. As part of this development process Copenhagen Atomics have developed most of the components that go into a reactor from scratch and with the ability to operate for 5 years without need of maintenance. This includes canned active electromagnetic pumps, electrochemical sensors, autonomous consensus steering electronics and software.

4. Main Design Features

(a) Power Conversion Unit

The Copenhagen Atomics Waste Burner delivers heat as a service and is operated by Copenhagen Atomics. The customer of the heat can choose to couple the reactor heat output with a power conversion system.

(b) Reactor Core

The Copenhagen Atomics Waste Burner uses $^{7}LiF\text{-}UF_4$ or $^{7}LiF\text{-}ThF_4\text{-}(TRU)F_3$ kickstarter fuel salt, $^{7}LiF\text{-}ThF_4$ blanket salt, and unpressurized room temperature heavy water moderator. Bred uranium from the blanket is transferred to the fuel salt online from within the reactor container.

The heavy water moderator is circulated and cooled, and thermally isolated from the molten salts, so that the vast majority of moderator heating is through the thermalization of neutrons. The moderator cooling is achieved with an external commercial chiller, consuming a small fraction of the produced heat to maintain the moderator temperature. Employing a blanket salt allows for a compact core while maintaining a low neutron leakage. The reactor core design enables self-sustained breeding in thermal spectrum with the thorium fuel cycle, if fission products are removed online and composite core construction material is employed. Current versions of the Copenhagen Atomics Waste Burner prototype’s core uses stainless steel core construction material and fission product separation limited to volatile gases.

(c) Reactivity Control

During normal operation the liquid moderator allows core reactivity control through a simple heavy water level adjustment. Gradual lowering of fissile inventory from transuranic burning period can be compensated by gradually increasing the heavy water level. Gradual build-up of fissile inventory from breeding when predominantly running on thorium cycle can be compensated by gradually lowering the heavy water level.

The highly negative temperature reactivity feedback of the Copenhagen Atomics Waste Burner is relied upon as a second and independent reactivity control mechanism.

The fuel salt, blanket salt, and heavy water moderator are all respectively being continuously and passively drained from the core at a high rate and actively pumped and cooled before going back into the core. If power is shut off or any of the pumps tripped the reactor will shut down, due to the lack of fuel, moderator, and or reflector blanket.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The entire reactor operation and shutdown is designed to be completely autonomous with decay heat removed passively. Safety is ensured through inherent safety instead of operator action.

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

The fuel salt is pumped through the dump tank so that in the case of a pump trip the fuel salt drains quickly to the dump tank, where the decay can conduct through the tank bottom and through the primary radiation shielding to the surroundings. This enables completely passive decay heat removal but also means that there's always a parasitic heat loss through the same heat path during normal operation. This is an economic trade-off for enhanced passive safety.

(c) Containment System

The reactor containment consists of three physical barriers to nature. The first barrier is the fuel salt and fission product trap wetted structure and components. The second barrier is the leak tight 40 feet size vessel that surrounds the first barrier, reactor core, reactor coolant salt, and other components. The third barrier is an exterior onsite constructed leak tight 'cocoon' containment that also serves as radiation shielding. Another path is through the reactor coolant and secondary coolant salt which is served to the customer. Here the three heat exchangers serve as three physical barriers, first the fuel salt to reactor coolant salt heat exchanger, second the reactor coolant salt to secondary coolant salt, both of these are located inside the 40 feet sized vessel, and third the heat exchanger or boiler on the customer side.

(d) Spent Fuel Cooling Safety Approach / System

If online fission product removal is employed then the fuel salt can be transferred to a new reactor unit, at the end of life or after failure of one unit. If only online volatile fission product removal is employed then intermediate salt cleaning will be necessary before transfer to a new unit. Thus, due to the continuous use of the fuel there is no spent fuel. Fission product traps are envisioned to be left in the used reactor vessel, where their decay heat can be passively removed through the normal decay heat removal system for a cool down period before decommissioning. In the case of a failure of the salt wetted barrier where the fuel salt can't be transferred through the normal means then it would be left for a cool down period before decommissioning, where the fuel would be separated, cleaned, and reused.

6. Plant Safety and Operational Performances

The Copenhagen Atomics Waste Burner's safety is not dependent on the use of the heat that is delivered to the customer. The reactor doesn't require refuelling, human intervention, or maintenance for its 5-year design life and operates completely autonomously with passive decay heat removal. A plant is envisaged to have dozens of reactor units, placed together and inaccessible during operation, besides for swapping of reactor vessels under observation. Satisfactory safety statistics for commercial operation will be determined from reactor prototype operation.

7. Instrumentation and Control System

Copenhagen Atomics makes/writes all of our own electronics and software for the Copenhagen Atomics Waste Burner. The system uses redundant consumer grade electronics running consensus steering for autonomous operation of the reactor throughout its 5-year lifetime. All instrumentation and control systems are sealed inside the 40 feet sized reactor vessel, only serve a data uplink through data diodes, and can't be accessed or altered after production.

8. Plant Layout Arrangement

The reactor and containment structures function as an adjacent heat source for existing or new industrial plants on site but are inaccessible from the rest of the site with a land claim of roughly 50,000 m^2 per 2.5GW(t)/1GW(e) plant. Since the customer is served up to 560°C molten nitrate salt the reactor site and user can also be physically separated by up to kilometres with pipelines in between.

9. Testing Conducted for Design Verification and Validation

Copenhagen Atomics relies extensively on prototype testing and is envisaging testing several prototype reactors before offering a commercial reactor.

Currently, Copenhagen Atomics are operating a non-fission test reactor prototype, multi-ton scale salt production, two dozen pumped molten salt loops, four dozen static molten salt test systems, and are setting up a large scale thorium and uranium salt production, lithium enrichment facility, and production of several more non-fission test reactor prototypes at Copenhagen Atomics's 11,000 m^2 facility in Copenhagen.

10. Design and Licensing Status

Copenhagen Atomics plans to run multiple test reactors in multiple countries before moving to commercial deployment. In 2024, Copenhagen Atomics and Paul Scherrer Institute (PSI) started a collaboration to build a facility for running moderate power criticality experiments, and are planning to license Copenhagen Atomics's 1MW test reactor for operation at PSI.

11. Fuel Cycle Approach

The Copenhagen Atomics Waste Burner uses $^7LiF\text{-}UF_4$ fuel salt with 27% mol low uranium or $^7LiF\text{-}ThF_4\text{-}(TRU)F_3$ fuel salt with or 3% mol $(TRU)F_3$ or $RGPuF_3$ kickstarter fuel salt and $^7LiF\text{-}ThF_4$ blanket salt. The reactor can transition to a breeder reactor using the heavy water level for coarse reactivity adjustment. The reactor core design enables self-sustained breeding in the thermal spectrum and thorium fuel cycle if fission products are removed online and carbon or silicon carbide composite core construction material is employed. The 100MW(t) Copenhagen Atomics Waste Burner requires an initial load of roughly 2500 kg of 4.95% enriched uranium or 300 kg of TRU/RGPu, respectively, and doesn't require any refuelling for its 5-year operation. After 5-year operation most of the fissile material in the fuel salt is ^{233}U that was produced in the blanket. The fuel and blanket salt can be reused in subsequent reactors without adding makeup fuel.

12. Waste Management and Disposal Plan

The Copenhagen Atomics Waste Burner can transition to breeding within the first three years of operations with online removal of fission products and carbon or silicon carbide composite core construction material. However, the high power density of the small core limits the operation life of the reactor due to neutron irradiation, but the salts and heavy moderator have, in principle, unlimited lifetime and will be reused in subsequent reactors, while the reactor vessel is the consumable and fission products are left as a by-product.

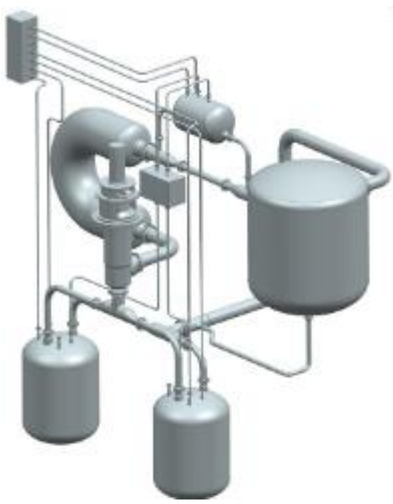
After operation the reactor vessel and fission product trap are left to cool down before decommissioning. For decommissioning it's envisaged to separate fission products for vitrification or other uses and remelting of the vessel for non-nuclear reuse with low remaining activity and non-metals separated in the slag.

13. 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.
2023	Moved to 11,000 m^2 production facility.
2024	Started collaboration to build and license first test reactor.
2026	Planned first 1 MW(t) test reactor.
2028	Planned first 100 MW(t) commercial reactor.

CMSR (Seaborg Technologies, Denmark)

CMSR POWER BARGE



CMSR

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Seaborg Technologies ApS, Denmark
Reactor type	Molten salt reactor / thermal spectrum
Coolant/moderator	Fluoride fuel salt / graphite
Thermal/electrical capacity, MW(t)/MW(e)	250 / 110 per CMSR
Primary circulation	Forced circulation
Operating Pressure (MPa) (primary/secondary/NSSS)	0.45 / 1.05 / 18
Core Inlet/Outlet Coolant Temperature (°C)	600 / 650
Fuel type/assembly array	LEU / Molten salt fuel
Power conversion process	Superheated steam driven turbine / generator (Rankine Cycle)
Fuel enrichment (%)	Approx. 2%
Refuelling cycle (months)	Online refueling with 5% LEU
Core Discharge Burnup (GWd/ton)	Approx. 6 GWd/tU
Reactivity control mechanism	Negative temperature coefficients, regulating and safety rods, fuel salt draining
Approach to safety systems	Automatic shutdown, Passive decay heat removal
Design life (years)	12 per CMSR / 24 per Power Barge
Plant footprint (m^2)	200 - 800 MWe – 5 000 – 14 000 m^2 Depending on the Power Barge model.
RPV height/diameter (m)	5.5 / 6.0
RPV weight (metric ton)	138
Seismic design (SSE)	Not applicable
Fuel cycle requirements / approach	LEU with online refuelling Off-site reprocessing
Distinguishing features	CMSR integrated into a floating non-self-propelled Power Barge; new CMSR units installed after 12 years. CMSR Power Barge operates for 24 years. Modular nature provides output from 200-800 MWe Liquid fuel which is the primary coolant.
Design status	Conceptual design

1. Introduction

The Seaborg CMSR is an advanced, small and modular molten salt reactor characterized by using a liquid fluoride molten salt fuel in direct contact with graphite moderator. The CMSR has a thermal power capacity of 250 MWth which is transformed into more than 100 MWe for grid transmission and to cover house loads. It operates on standard LEU with enrichment below 5% U-235, making use of high-frequency online refuelling to achieve its operating lifetime of 12 years while maintaining low excess reactivity and minimizing control rod movement. To facilitate inspection and maintenance, the CMSR is being designed as a loop-type reactor implementing a single fuel salt circulation loop. Due to its very high melting point, the CMSR needs no pressurization above what is needed to circulate the fuel salt which together with the avoidance of phase changing material in close proximity to the fuel salt circuit yields important and unique inherent safety benefits. The CMSR is deployed on modular and standardized non-self-propelled CMSR Power Barges, scalable from 200 MWe to 800 MWe (containing one to four so-called Power Modules, each equipped with two concurrently operating CMSRs), with an operating life of 24 years.

2. Target Application

The CMSR Power Barge is designed to generate power for the electrical grid and for industrial applications,
such as hydrogen electrolysis. The CMSR Power Barge can supply high temperature heat for non-electrical applications such as district heating, water desalination, and industrial processes.

3. Design Philosophy

The overarching design goal with the CMSR Power Barge is to centrally produce highly standardized, scalable and cost-competitive nuclear power plants that can be reliably deployed worldwide within three years from the date of order. Hence, the following constitutes the design philosophy:

- Non-proliferation enabling worldwide deployment.
- CMSR safety and integrity achieved by passive means as far as reasonably practicable.
- Number and complexity of nuclear-grade components reduced as far as practicable.
- Modularity of design to facilitate efficient construction, transportation, installation, commissioning, maintenance, and decommissioning.
- Internationally harmonized regulatory approach integrated into the CMSR life cycle.

4. Main Design Features

(a) Power Conversion

Each CMSR molten salt fuel loop transfers heat generated in the core to a secondary liquid heat-transfer fluoride salt via the primary heat exchanger. The secondary circuit then transfers heat to a tertiary salt circuit which circulates solar salt from the secondary heat exchanger and into the steam generator, using technology from concentrated solar power plants. The Power Conversion System receives heat in the form of superheated steam from two concurrently operating CMSRs, at pressures and temperatures up to 18 MPa and 565ºC. The steam produced from the two steam generators drives a single conventional condensing steam turbine with exhaust positioned axially into the seawater-cooled condenser. Electrical output from the connected generators allows delivery of 200 MWe net from each Power Module through a common electrical interface to the high-voltage onshore electrical transmission grid. Through this arrangement, a CMSR Power Barge can provide between 200 and 800 MWe net to the grid depending on the number of modules installed.

(b) Reactor Core

The CMSR reactor core consists of several hundred graphite blocks arranged into columns in a central active core region and a periphery reflector region. In total, the graphite core measures approximately 5.5 m in diameter and 5 m in height. The fuel salt flows through fuel channels in the graphite core configuration and transports heat deposited into the fuel salt, graphite, and structural materials. At certain positions in the core, metallic control rod guide tubes extend down from the top of the core, providing an unobstructed path for inserting the control rods.

(c) Reactivity Control

The CMSRs credits two main methods for reactivity control: control rods and fuel salt draining, which are relatively fast and slow acting, respectively. The control rods are subdivided into regulating rods for compensating short-term reactivity changes during operation, and safety rods for extinguishing the nuclear chain reaction promptly when needed. The draining of the fuel salt is initiated by opening one of several fuel salt drain valves that are redundant, independent, and diverse. Inherent reactivity control means include negative temperature coefficients and the geometrical configuration of the fuel salt drain tanks.

(d) Reactor Vessel and Internals

The CMSR reactor vessel measures approximately 6 m in diameter, accommodating the graphite active core and reflector regions followed by a fuel salt channel which cools the reactor vessel during power operation. Due to its relatively low operating pressure, the vessel will be welded rather than forged. It has upper and lower plena that sit above and below the graphite core, respectively, which guide and distribute the fuel salt flow through the core. On the top of the vessel are penetrations for the control rod guide tubes and inspection channels. The reactor vessel internals include the control rod guide tubes, structural supports that keep the graphite in place, and equipment for material surveillance.

(e) Fuel Characteristics

The fuel consists of NaF, KF and UF4, with partial reduction of UF4 to UF3 to limit degradation of the structural materials and graphite. The uranium enrichment differs through the fuel cycle, with the initial fuel loading at approximately 2% and subsequent online refueling at approximately 5%. Most fission products form soluble fluoride species which are retained in the fuel, while those that are not either plate out on fuel-facing surfaces or bubble out. The volatile species that bubble out are collected and passed to an off gas system that delays and conditions the gases, effectively removing them as a source term in postulated reactor accidents.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The CMSR and CMSR Power Barge design implements a tailored approach to defense in depth with a top-down focus on the three fundamental safety functions, taking into account the inherent safety features of the CMSR in a risk-informed and performance-based manner. The inherent safety features include:

- For control of reactivity: negative temperature coefficients; use of draining to reach a subcritical configuration; low excess reactivity due to online refueling.
- For cooling of fuel and waste stores: high boiling point and large margin to boiling; volumetric heat capacity of fuel and graphite; low decay heat power density.
- For confinement: retention of fission products in fuel; low-pressure operation; no material that can phase change, burn or explode in proximity of the fuel.

The engineered safety systems include:

- For control of reactivity: regulating and safety rods.
- For cooling of fuel and waste stores: a combination of active and passive decay heat removal systems.
- For confinement: a number of independent and isolatable containment barriers provided around different sources of radioactivity, with the number and leak-tightness requirements based on risk-informed and performance-based principles.

(b) Decay Heat Removal System

The CMSR implements a decay heat removal system that removes heat from the fuel salt drain tanks. During normal operation and hot standby, heat is removed through the salt and turbine circuits. By opening one out of several redundant, independent and diverse fuel salt drain valves, the fuel salt is drained by gravity to the fuel salt drain tanks. There decay heat is transported to the decay heat removal system through radiative heat transfer, and the heat can subsequently be dissipated in the ultimate heat sink through either active or passive means. The passive decay heat removal system relies on the natural convection of water, and ultimately rejects the heat to the atmosphere. It is dimensioned for at least 72 hours of decay heat removal without operator interaction.

(c) Emergency Core Cooling System

The CMSR does not implement a traditional emergency core cooling system. Due to the low-pressure operation, high thermal inertia, passive fuel salt draining, and passive capabilities of the decay heat removal system, the omission of an emergency core cooling system is thought to be acceptable following the risk-informed and performance-based safety approach.

(d) Containment System

Radioactive sources onboard the CMSR Power Barge are located in different compartments and separated from the environment by a number of physical containment barriers. The leak-tightness requirements of and number of containment barriers surrounding a compartment are commensurate with its risk importance. Penetrations through containment barriers are equipped with means of isolation that actuate in accident conditions.

(e) Chemical Control System

Chemical control systems are implemented for each of the three molten salt circuits, ensuring that salt chemistry remains within operational design limits during normal operation and in case of contamination of the salts. All rely on some combination of monitoring and addition of chemical reactants as required. Only the fuel salt circuit will experience a gradual drift in chemical parameters due to the effects of fission, including accumulation of fission products and excess fluorine. The fuel salt chemical control system maintains the partial reduction of UF4 to UF3 to maintain the redox potential that is necessary to minimize uniform corrosion during normal operation. Due to a considerable fuel inventory that extends outside of the active core, the fuel salt is expected to have a sizable redox potential buffer against changes. Consequently, it is considered appropriate to implement monitoring through frequent fuel salt sampling.

(f) Spent Fuel Cooling Safety Approach / System

Each CMSR in the CMSR Power Barge has an operational lifetime of 12 years, after which the fuel salt drain tanks in conjunction with the dedicated decay heat removal system fulfils the role of spent fuel cooling system while decay heat is significant. The fuel salt is later allowed to solidify in the fuel salt drain tanks and passive means of decay heat removal are employed in a similar fashion to dry cask storage for conventional spent nuclear fuel.

6. Plant Safety and Operational Performances

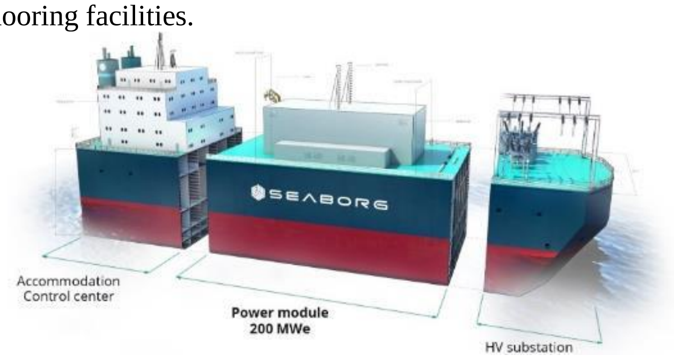
It is not yet considered meaningful to calculate absolute damage and release frequencies for the CMSR and the CMSR Power Barge. Many fundamental design decision – including relating to the low operating pressure, non-reliance on phase change or exothermically reactive materials close to the fuel, high retention of most fission products and the continuous removal and handling of those that are not retained – have been made to ensure a very low radioactive release frequency at highly competitive cost. Due to online refueling, reactor outages are dictated by inspection and maintenance needs. Similarly, Power Module outages are dictated by the need to inspect and maintain shared equipment, importantly including the steam turbine. After 12 years of operation, a longer reactor outage is needed to install and commission new CMSRs. The CMSR is being designed to be capable of achieving a capacity factor of 90%.

7. Instrumentation and Control System

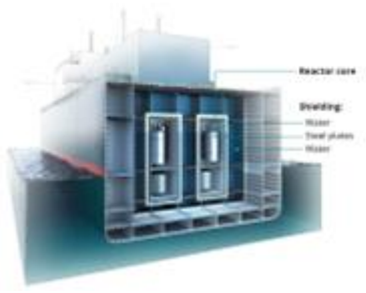
The Seaborg CMSR Power Barge I&C systems provide automated control, monitoring, and protection functions of the CMSR Power Barge. The Seaborg CMSR I&C architecture reflects the modular nature of the CMSR Power Barge and is designed to ensure separation of safety and non-safety systems. Each CMSR reactor has its own protection system which monitors critical parameters and if required will automatically initiate reactor trip and relevant safety functions to achieve and maintain the CMSR systems in a safe state.

8. Plant Layout Arrangement

The CMSR Power Barge consists of one or more power modules connected to an aft end module (accommodation, offices, conference rooms, main control room, supplementary control room, emergency power, and barge utilities) and a fore end module, where the high voltage substation shall be positioned. The entire hull is designed with a complete double hull, which protects the safety enclosure, the CMSR auxiliary systems, and the emergency diesel generators, diesel tanks, and switchboards. The CMSR Power Barge is not self-propelled and shall be equipped with suitable towing and mooring facilities.



200MWe Power Module of the CMSR Power Barge



Cross-sectional view

9. Testing Conducted for Design Verification and Validation

Seaborg's overall approach to technology development and maturation follows a systematic and phased framework structured around the American Bureau of Shipping's (ABS) New Technology Qualification process. It includes a large-scale research, development and testing campaign that is currently being executed by Seaborg and its collaborators in support of the development of the CMSR and CMSR Power Barge. The campaign covers research, development and testing at a wide range of scales and complexities, from small-scale salt chemistry experiments to large integrated test facilities and irradiation testing in research reactors. It is being carried out partly at Seaborg's own facilities which include two molten salt laboratory facilities, one testing facility for medium-scale tests including flow loops, and a radionuclide laboratory under approval by local authorities. Other experiments have and will be carried out at partner facilities to accelerate development. Testing to date includes, but is not limited to:

- A very significant number of static corrosion and chemistry tests covering a wide range of salts and conditions.
- Five generations of thermal convection flow loops with nitrate, hydroxide and fluoride salts, equipped also with salt chemistry monitoring and control provisions.
- Molten salt natural convection testing.
- Preliminary scaled model testing with surrogate liquids to explore integrated effects and start validating computer codes.
- Fuel salt phase diagram exploration, with and without fission products.
- Tests on mixing and purification of fuel salt.
- Investigation and validation of thermophysical salt properties.

10. Design and Licensing Status

The CMSR is at a conceptual design stage with an emphasis on a research and development approach through the development of test facilities and loops to support licensing of a first of a kind CMSR, simplify design and reduce regulatory risk. The licensing of a first of a kind CMSR focuses on a standard design assessment based on a technology-inclusive international framework and complemented by the ongoing development of risk-informed performance-based approaches. The first-of-a-kind CMSR Power Barge licensing will occur in the Republic of Korea. Regulatory engagement is on-going. Additional preliminary engagement with other regulators to support international deployment of the CMSR Power Barge is on-going. In the longer term, Seaborg is working towards a harmonized international licensing framework modelled on maritime industries.

11. Fuel Cycle Approach

The initial fuel salt loading for the CMSR will be produced at a centralized fuel mixing facility, packed and transported under a controlled atmosphere as fluoride salt pellets, and eventually loaded into the fuel salt drain tanks of the CMSR prior to nuclear commissioning. This fuel has an enrichment of approximately 2% U-235. During operation, additional fuel salt with an enrichment of approximately 5% U-235 will be added at regular intervals to compensate for fissile depletion and fission product accumulation while ensuring very low excess reactivity and minimal control rod movement. The added fuel salt volume, together with volumetric changes due to thermal expansion, is accommodated by the fuel salt drain tanks. At the end of the CMSR lifetime, the fuel is returned to the fuel salt drain tanks and cooled. At the end of the CMSR Power Barge lifetime, the barge is returned to a specialized and dedicated decommissioning facility where the fuel salt is handled and ultimately reprocessed.

12. Waste Management and Disposal Plan

The waste management and disposal approach follow a "greenfield" philosophy and is intended to be flexible and adapted to fit national policies of the host state and contractual arrangements. Industry best practices are applied to manage and classify nuclear waste, with higher level waste (incl. used fuel salt) predominantly aimed for storage, disposal or, preferably, reprocessing in dedicated facilities. Lower-level waste is aimed for storage or disposal in the host state country or abroad provided adequate additional provisions.

13. Development Milestones

Seaborg has established a consortium with Samsung Heavy Industries (SHI) and Korean Hydro & Nuclear Power (KHNP) with the aim to export factory build turnkey CMSR Power Barges from Korea. The consortium is targeting South Korea as its first of a kind country with the CMSR Power Barge being licensed by the South Korean nuclear regulator. For the maritime classification of the CMSR Power Barge, Seaborg and SHI are working with ABS following the ABS New Technology Qualification process for the CMSR and the Novel Concept process for the CMSR Power Barge enabling delivery of the first Power Barge.

2020	CMSR Feasibility Statement received by ABS
2022	Power Barge Approval in Principle by ABS
2025	End of concept verification
2029	End of design
2033	Delivery of first Power Barge

FLEX Reactor (MoltexFLEX, United Kingdom)

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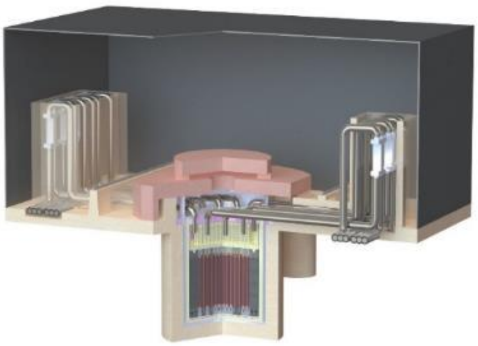


Schematic of single FLEX reactor unit

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	MoltexFLEX, United Kingdom
Reactor type	Molten Salt Reactor
Coolant/moderator	Molten eutectic AlF_3/NaF coolant salt / commercial grade graphite moderator
Thermal/electrical capacity, MW(t)/MW(e)	Single unit: 60 MWth/ 24 MWe, deployable in arrays
Primary circulation	Natural convection loop
NSSS Operating Pressure (primary/secondary), MPa	Primary coolant unpressurised (~0.1 MPa) Note secondary loop is GridReserve®
Core Inlet/Outlet Coolant Temperature (^{o}C)	725 / 795
Fuel type/assembly array	Molten salt fuel within vented fuel tubes hexagonal array
Power conversion process	Molten salt to steam to turbine
Fuel enrichment (%)	5
Refuelling cycle (months)	36-60
Core Discharge Burnup (GWd/t)	30-35
Reactivity control mechanism	Strong fuel temperature coefficient, liquid neutron absorber thermometer
Approach to safety systems	Hazard elimination, inherent/passive safety
Design life (years)	60
Plant footprint (m^2)	1 unit ~450, 20 units (0.5 GWe) ~ 10,000
RPV height/diameter (m)	7.5 / 6 (low pressure)
RPV weight (metric ton)	Empty ~100 / Filled ~500
Seismic design (SSE)	EUR standard - peak ground acceleration of 0.3g horizontally and vertically
Fuel cycle requirements / approach	3 batch refuelling cycle with 3-5 year cycles, zoned refuelling pattern
Distinguishing features	Costs lower than fossil fuels, full passive safety, simplicity - no active controls systems within the core, and less than one tenth of the control systems in a PWR
Design status	Pre-conceptual design

1. Introduction

The FLEX reactor is a molten salt reactor (MSR). It is graphite-moderated, with a patented two-fluid molten salt core, and is being developed by MoltexFLEX Ltd., a UK-based subsidiary of British company Moltex Energy Ltd. It is designed to contribute towards the CO2 emissions reduction challenge, delivering clean energy at a lower cost than unabated coal or gas at pre COVID-19 prices. The FLEX reactor's design is radically simpler than other nuclear technologies. The FLEX reactor has around 10% of the engineered systems of a conventional nuclear reactor (e.g. PWR). The FLEX reactor is inherently safe, operating at near-atmospheric pressure, which substantially reduces the cost of containment. Its operational safety is provided through inherent and passive safety systems, drastically reducing capital and operating costs.

The FLEX reactor is designed with simplicity in mind. It is small and modular, allowing most components to be factory-produced, facilitating transportability, reducing on-site work, and expediting

construction, all of which minimises overall costs and build time. The FLEX reactor maximises use of materials and technology already proven within the nuclear industry, making safety and design substantiation easier and quicker by eliminating the need for extensive research programmes. The FLEX reactor is the thermal neutron (moderated) version of Moltex Energy Limited's globally patented stable salt reactor technology. That technology is shared with MoltexFLEX's sister company, Moltex Energy Canada Inc., which is developing the fast neutron version of the stable salt reactor.
Stable salt reactors differ fundamentally from all other MSRs in that they restrict the liquid fuel salt to fuel tubes, similar to fuel pins in conventional reactors. A separate, non-active coolant salt transfers heat from the reactor core. This avoids the possibility of pumps, filters, conditioning units and heat exchangers in the primary circuit being contaminated with actinides and fission products, which would put severe demands on those components and make monitoring and maintenance complex and expensive.

2. Target Application

The reactor is flexible in its application; it is suitable for grid connection or standalone electricity and heat supply. It enables cost-effective storage of thermal energy for hours to days in MoltexFLEX's chloride salt-based GridReserve® system through its high temperature output. The FLEX reactor primary circuit outlet temperature is 795°C, and GridReserve® provides heat at 700°C. which can be directly used for electricity generation or additional downstream applications in industries requiring high-temperature heat. A typical site may contain any configuration of modular FLEX reactors, each with a thermal power output of 60 MW, equivalent to 24 MWe using conventional steam turbines/generators. During longer periods of low demand, the FLEX reactor can reduce its power output. Its low capital cost makes this economically practical.
FLEX reactors can also be used to directly feed industrial processes, including thermo-chemical hydrogen production, which offers increased efficiency compared to electrolysis. The hydrogen itself can be used in several ways to support domestic, commercial, and industrial applications, such as a direct substitute for gas use in industry, domestic heating fuel, fuel cells for heavy transport, and as a feedstock for synthetic fuels, including ammonia as a bunker-oil substitute for shipping. The reactor is also well suited to combined heat and power generation for district heating.

3. Design Philosophy

The MoltexFLEX design philosophy is predicated by reducing cost, bringing the technology to market as quickly as possible, and developing a reactor that can be deployed worldwide to meet its mission of having a meaningful impact on global CO_2 emissions. MoltexFLEX prioritises simplicity, inherent safety and using materials and manufacturing techniques which already exist at a high TRL. This has led to an extremely simple atmospheric-pressure design which requires no active engineered systems in order to ensure safety. For example, the use of molten salt eliminates the pressure hazard associated with LWRs, placing the fuel in pins eliminates the vast majority of radioactive contamination from the primary circuit and the use of passive decay heat removal removes the requirement for active decay heat removal systems and associated backup electricity generators.
MoltexFLEX's approach is to adopt solutions that are already proven in the nuclear sector, meaning a significant proportion of the design is already mature. The use of proven nuclear materials and well validated modelling tools, mean that in combination, it is possible to rapidly move through the Technology Readiness Levels (TRL). Global deployability is ensured through the use of low-enriched uranium (LEU), and its modular construction.

4. Main Design Features

(a) Power Conversion

Each FLEX reactor outputs 60 MWth as hot salt at 700°C. This heat can be directly fed to a steam boiler to generate steam at typical subcritical conditions, where there is already a wealth of experience in flexible operation of generating plant, generating ~24 MWe. Alternatively, the heat can be fed to external thermal storage tanks (located outside the nuclear island), known as the GridReserve® system.
The use of the GridReserve® technology permits electrical generation to follow a dispatched load profile or to operate in a frequency responsive mode. It is also possible to dynamically allocate energy

to other heat loads. The ability of one or more FLEX reactor systems to operate with different sizes of GridReserve® means that different energy load profiles are readily accommodated.

(b) Reactor Core

The FLEX is a fluoride salt reactor with separate fuel and coolant salts. The patented core design comprises an array of vented fuel tubes in a graphite matrix which fills most of the tank. Each tube sits in a separate channel, and the primary coolant molten salt circulates up through the channel by natural convection. It then flows through a heat exchanger and flows around the core back to the fuel channels. The fuel salt circulates in the fuel tube by natural convection and heat is transferred through the tube wall to the coolant surrounding the tube. The fuel salt does not mix with the coolant salt under normal operation. Under fault conditions the miscibility of fuel and coolant salt fundamentally changes the impact of core damage. In a conventional reactor, core damage can result in the reactor moving to a substantially more hazardous state. In the FLEX, similar scenarios are tolerable since even severe core damage reduces k-eff. Breach of the tube wall results in a reduction in core reactivity as the fuel salt is diluted in the large coolant volume. As a result, core damage does not result in further damage or the loss of further containment barriers.

(c) Reactivity Control

In the FLEX, the bulk of the excess reactivity present immediately after fuelling is neutralised through burnable absorber material within the core. Any residual excess reactivity is neutralised by a soluble neutron absorber in the coolant salt, which is added to the primary coolant using a periodic injection system. This neutron absorber burns out slowly and is replenished at intervals of several weeks or months. Fine reactivity control for reactivity changes during power demand shifts is provided by a patented novel mechanism akin to a conventional mercury thermometer, but instead filled with a liquid neutron absorber. The “bulb” of the thermometer sits in the cold coolant salt entering the bottom of the core and the stem extends upwards through the graphite matrix. An increase in heat demand from the reactor causes a drop in primary coolant temperature, cooling the thermometer bulb, which causes the neutron absorber to withdraw down the stem of the thermometer, thereby increasing reactivity. When heat demand is reduced, the temperature of the primary coolant rises, heating the thermometer bulb, driving the neutron absorber up through the stem and into the coolant, reducing reactivity.

(d) Reactor Vessel and Internals

The reactor vessel for the FLEX comprises an outer and inner tank, separated by a refractory layer. The outer tank, constructed from a stainless steel, will serve as the primary structural vessel for the reactor core and primary coolant loop. The inner tank, also constructed from a stainless steel, will serve as the liner to contain the primary coolant salt at temperatures above 700°C, and is designed to accommodate radial expansion. The refractory layer will act as the insulation layer for the inner tank, whilst also providing neutron irradiation shielding to the outer tank. The inner and outer tank vessels are placed in a concrete pit underground and covered from the top with a lid containing plugs for refuelling. A concrete shield covers the lid.

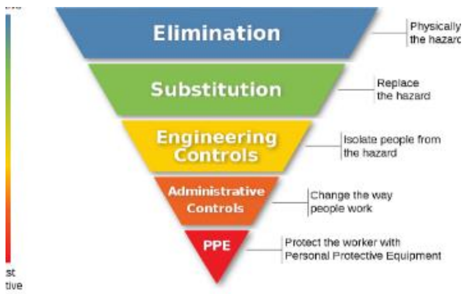
(e) Fuel Characteristics

The fuel is 5% enriched uranium within a mixture of UF_3, UF_4 and NaF molten salt, which is contained at low pressures by the patented vented fuel tube. The fuel expands when it is heated, contributing towards the fuel salt’s negative temperature coefficient.

(f) Fission Product Management

Many of the fission products that would often form gases in other reactors, notably iodine and caesium, are immediately captured as non-volatile salts and remain contained in the fuel salt within the fuel tube. The major releases are the Xe and Kr noble gases, and the great majority of the radioactive species decay within the tube. The top of the fuel tube contains a ‘bubbler’ assembly, which is essentially a series of molten salt gas traps. This limits the speed at which fission gases can reach the reactor gas space and primary coolant, allowing them time to decay. This significantly reduces the ability of ^{135}Cs, a long-lived fission product, to contaminate the coolant salt.

5. Safety Features



(a) Engineered Safety System Approach and Configuration

The design philosophy adopted is to follow the internationally accepted principle of the risk mitigation pyramid. The focus is to eliminate hazards wherever possible, to use inherent and passive safety features where this is not possible, and only to rely on engineering or administrative controls when that cannot be achieved. As a result the core has no moving parts or active control systems.

(b) Decay Heat Removal System

The decay (or residual) heat removal system (RHRS) is completely passive and utilises natural convection with no need for mechanical components. Utilising an arrangement of ducts around the reactor tank, it promotes air cooling of the base and side walls of the reactor tank, drawing cool air down and around the external tank walls and exhausting the heated air to the reactor building.

(c) Emergency Core Cooling System

A traditional Emergency Core cooling system is not required for the FLEX reactor as the design ensures that the core is kept submerged by the coolant and never exposed, and decay heat is removed by convection as previously explained.

(d) Containment System

Containment is progressively provided by the fuel tube, the inner tank, and the outer tank. Shielding is provided by the pit walls and the concrete biological shield. The salt chemistry of the fuel/coolant immobilises some otherwise volatile fission products.

(e) Chemical Control

The FLEX has distinct fuel and coolant chemical environments. The fuel salt chemistry is very simple and is redox stabilised by a mixture of uranium oxidation states acting as a buffer in a eutectic mixture with sodium fluoride diluent. The redox buffer enables maintenance of redox potential and mitigation of potentially corrosive fission products generated through life. The primary coolant is also a fluoride eutectic. Tritium production is mitigated by eliminating lithium from the chemistry. Similarly, beryllium is also eliminated to prevent highly corrosive and toxic environments. The primary coolant salt is compatible with graphite and mixes safely with the fuel salt. Corrosion of reactor components by the coolant salt is managed using reducing agents. Moltex has been operating a laboratory since 2021. It is set up to deliver on salt manufacturing for both fuel and coolant, alongside corrosion trials, thermophysical properties and test rigs. Significant progress has been made in the understanding of the fluids within the system and the demonstration of their compatibility with structural materials..

(f) Spent Fuel Cooling Safety Approach / System

Once the FLEX's fuel is depleted, the reactor is put into the standby state before the fuel is removed from the reactor. During this period, there is no requirement for additional cooling systems. The normal heat transfer systems are disengaged and the decay heat removal system, as described in (b), provides sufficient heat removal until the fuel is ready for extraction.

6. Plant Safety and Operational Performances

The design philosophy is such that there will be very limited operational requirements to attend the nuclear island during each fuel cycle. The significant reduction in the quantity of engineered safety and component systems will also substantially reduce the number of operating staff required for maintenance.

The FLEX can passively ramp up and down in output, but Moltex considers the use of GridReserve® a better and more economical option for the provision of flexible electricity generation. The FLEX reactor system can be designed to operate at full output throughout the 3-5 year fuel cycle.

7. Instrumentation and Control System

Instrumentation includes ex-core flux detectors for commissioning and startup, as well as regular physics tests (at least on early cores). An array of thermocouples are located above the fuel assemblies to detect the outlet temperature. Thermocouples are placed in the cold leg of the heat exchangers. Level sensors determine the coolant salt level within the core. Radiation detectors are present in both primary and secondary heat exchangers to determine whether fuel to coolant or primary to secondary leaks have occurred. Thermocouples are present on the outer tank to determine whether a breach in the inner tank has occurred. The reactor power is entirely regulated by the primary heat exchanger demand as the system passively load-follows in normal operation. During commissioning and refuelling, an electric heater array underneath the core provides sufficient heat to match/exceed the RHRS and maintain/increase the core temperature. There is also a periodic coolant poison injection system to trim the reactivity as the fuel burns up. This can also deliver a 'lethal dose' of poison if necessary to permanently shut down the reactor.

8. Plant Layout Arrangement

The FLEX is flexible and can be deployed in different array sizes as demanded by the application. An illustrative FLEX reactor plant layout arrangement would be an array of 20 reactors on the nuclear island, which will provide circa ~1.2 GWth energy which can be converted into ~500 MWe of electrical output. The external GridReserve® storage tanks, turbine hall, electrical sub-station, and potential hydrogen production plant are located outside the nuclear island. This configuration will enable a truly 'hybrid' energy generation, storage, and release scheme.

Artist's impression of an array of 32 modules configured in line with turbine hall, GridReserve® storage facility and sub-station

9. Testing Conducted for Design Verification and Validation

Current testing is focused on materials & corrosion, confirmation of thermo-physical properties of salts for use in modelling, and thermohydraulic tests. MoltexFLEX has established its own laboratory and is running a range of corrosion and materials compatibility tests, alongside thermophysical properties tests. Industrial and academic partners are supporting MoltexFLEX. The FLEX reactor is currently at TRL 2, and is close to TRL 3. A number of test rigs are planned to move the design to TRL 4 by summer 2027. The TRL assessment and plan to achieve this has been independently verified by three recognised experts in the UK nuclear industry.

10. Design and Licensing Status

Early engagement with UK nuclear regulators has been initiated and the first session is being scheduled. A number of potential sites are under consideration and discussions are ongoing.

11. Fuel Cycle Approach

The FLEX operates on a three-batch refuelling cycle with a cycle length of 2-5 years. Interim storage of spent fuel will be in multi-purpose dry storage cannisters on site until site decommissioning. The proposed spent fuel management route is geological disposal, but it may be possible to recycle the waste in a fast spectrum reactor.

12. Waste Management and Disposal Plan

Moltex is developing the design to be consistent with international good practice for waste management and spent fuel, aligned to the IAEA. The reactor system itself is being developed to design out as much operational, refuelling, and decommissioning waste as is reasonably practicable. The waste hierarchy is superimposed across the design process, by first seeking to avoid waste creation, and then minimising the category of unavoidable waste. Moltex envisages minimal waste production during normal operation of the FLEX. Argon 41 is likely to be generated as a by-product of the RHRS; however, this is unlikely to approach the argon discharge limits. Refuelling will generate argon, krypton & other gaseous wastes, but these will not exceed their designated discharge limits. A small amount of solid waste will be produced through spent fuel, which can be stored in multi-purpose cannisters on site until decommissioning. Decommissioning will produce solid, liquid, and gaseous wastes, the largest proportion of which will be graphite. Moltex seeks to recycle and re-use graphite wherever possible in new FLEX reactors.

13. Development Milestones

Q2 – 2020	Initial concept defined
Q1 – 2021	Initial Safety Assessment Report Completed
Q2 – 2023	Initial concept design matured, progress made towards concept design
Q3 – 2024	Independent reviews conducted by 3x UK nuclear experts
Q4 – 2025	TRL3 Fully Achieved
Q3 – 2027	TRL4 Achieved
T* + 2 years	Concept design achieved (* T = date when significant funding can be raised)

FUJI (International Thorium Molten-Salt Forum, Japan)

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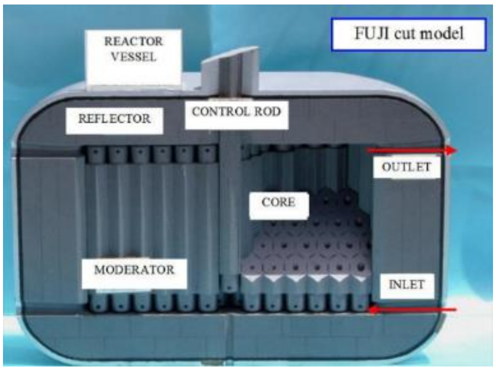
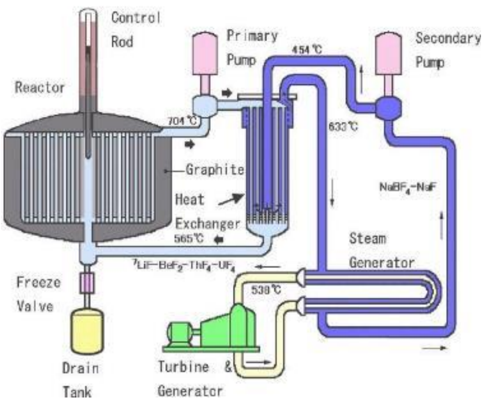


1. *MSR-FUJI cut model*



2. *Schematic diagram of MSR-FUJI*

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 (^{o}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%^{233}U + 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 (RB+SGB+TGB)
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 gaseous FP removal. Spent fuel salt is reprocessed
Distinguishing features	Self-sustaining at FUJI-U3. Spent fuel salt is reprocessed at off-site facility.
Design status	3 experimental MSRs were built. Detailed design not started

1. Introduction

The Molten Salt Reactor (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 25 MW(e) to 1000 MW(e), as is shown in the reference [1]. But, a latest and typical design (FUJI-U3) is 200 MW(e), which can be categorized as small-sized reactors with modular designs (SMR). The thermal output of FUJI-U3 is 450 MW(t) and thus a 44% thermal efficiency can be attained. In addition, the simple core structure and high fuel efficiency should facilitate a favourable economic performance.

Molten fuel salt can contain thorium (Th) as fertile material and ^{233}U as fissile material, and the FUJI-U3 design can attain a self-sustaining fuel cycle with a conversion factor of 1.0. Since MSR-FUJI applies the Th-cycle, generation of plutonium (Pu) and minor actinide (MA) is very small compared with Light Water Reactors (LWR). Furthermore, it can consume Pu, 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. 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.

4. Main Design Features

(a) Power Conversion

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 schematic diagram of MSR-FUJI 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 of 252 kg/cm^2 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.

(b) Reactor Core

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.

(c) 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.

(d) Reactor Vessel and Internals

The reactor vessel is made of Hastelloy N. Since the operating pressure is very low (0.5 MPa) a ‘pressure vessel’ is not required. The only one core internals is the graphite moderator blocks.

(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.

5. 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

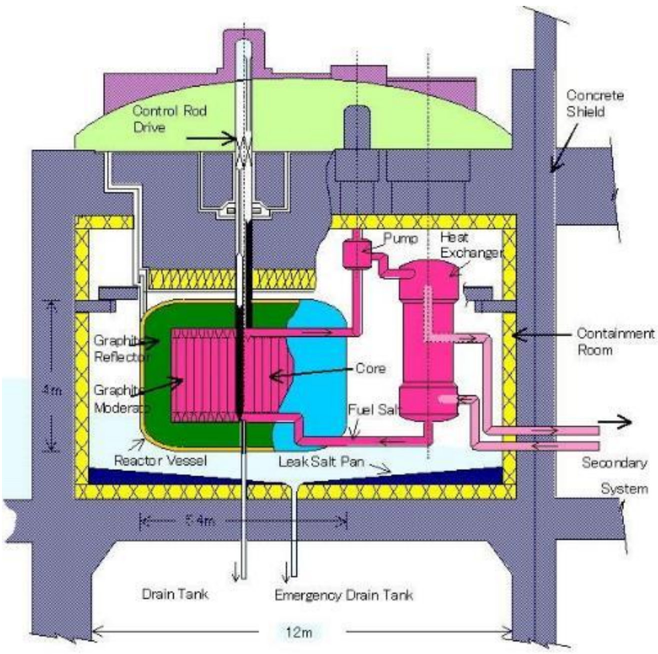
In normal shutdown condition, decay heat is transferred to a secondary loop and a steam-line loop, and disposed to the ultimate heat sink (seawater etc.). If all pumps in a primary or secondary loop stop, fuel salt is drained to a drain tank through a freeze valve. Decay heat at the drain tank is cooled 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.



Vertical cross-section of primary system of MSR-FUJI

6. 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.

7. Instrumentation and Control Systems

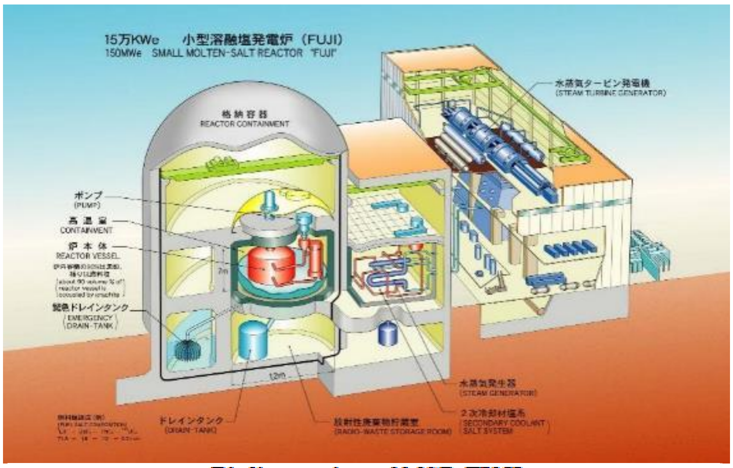
Instrumentation and control (I&C) systems in the MSR-FUJI design are the same as recent LWR designs. It must support operators in making decisions and efficiently operating the plant during plant start-up, shutdown, normal operation, surveillance testing, and accidental situations. It adopts the man-machine interface more useful, and expands the scope of automatic control.

8. 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.



Bird's-eye view of MSR-FUJI

(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

Turbine Generator (T/G) contains the supercritical T/G to produce electricity. Also, it contains condensers for disposed steam, which use outside water (sea water etc.) for cooling. The electric power systems include the main generator, transformers, emergency diesel generators, and batteries, besides external power lines. In case of station blackout, it can be shut down and cooled without electricity.

9. Testing Conducted for Design verification and Validation

MSR-FUJI is based on 3 experimental reactor experience. But some of technologies such as a supercritical SG or remote maintenance equipment have to be established.

10. 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 [1].

11. 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.

12. 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.

13. Development Milestones

1980's	Start of conceptual designs of MSR-FUJI reactors
1980's	Completion of conceptual design of Accelerator Molten-Salt Breeder (AMSB) design for a large production of fissile material (similar to Accelerator Driven System ADS)
Until 2008	Completion of conceptual design of FUJI-U3 as a typical SMR, a pilot plant (mini-FUJI) a large-sized plant (super-FUJI), and a Pu-fuelled plant (FUJI-Pu),
2024	Conceptual design of a micro-sized MSR (miniFUJI II) as one of SMR

[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, 2024(2nd edition).

IMSR400 (Terrestrial Energy Inc., Canada/USA)

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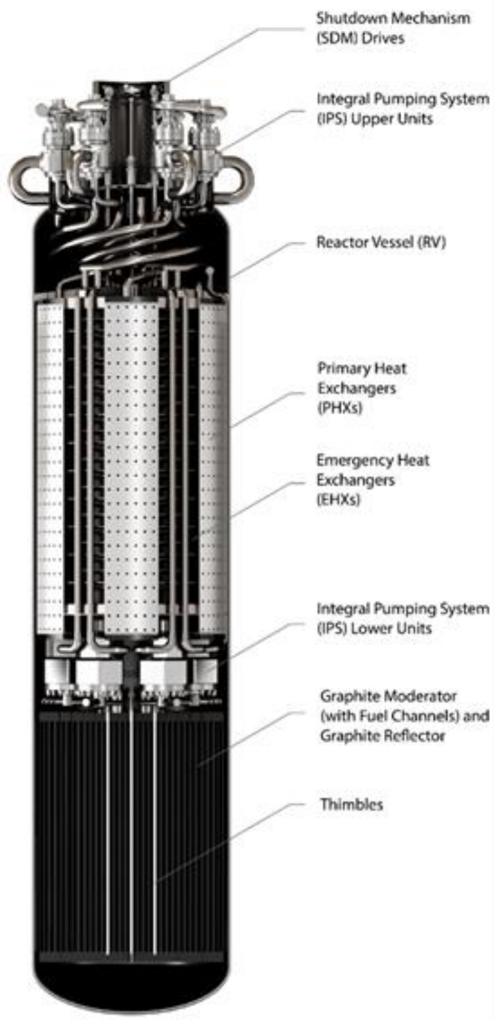


Fig. 1 IMSR Integral Core-unit, Section View

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Terrestrial Energy, Canada/USA
Reactor type	Molten Salt Reactor
Coolant/moderator	Molten salt / graphite
Thermal/electrical capacity, MW(t)/MW(e)	442 / 195 per operating Core-unit, 884 / 390 dual Core-unit reference configuration
Primary circulation	Forced
NSSS Operating Pressure (primary/secondary), MPa	0.4 / 0.45
Core Inlet/Outlet Coolant Temperature (°C)	610 / 700
Fuel type/assembly array	Molten salt fuel
Number of fuel assemblies in the core	Not applicable
Fuel enrichment (%)	<5%
Core Discharge Burnup (GWd/ton)	N/A
Refuelling Cycle (months)	84; before Core-unit replacement
Reactivity control	Negative reactivity coefficient
Approach to safety systems	Passive
Design life (years)	56
Plant footprint (m^2)	35,200
RPV height/diameter (m)	18/4.1
RPV weight (metric ton)	385
Seismic Design (SSE)	0.3g
Distinguishing features	Core-unit replaced completely as a single unit every 7 years; Fuel salt is reused from one unit to the next.
Design status	Detailed Design

1. Introduction

Terrestrial Energy's Integral Molten Salt Reactor (IMSR) is a commercial Gen IV advanced reactor, to be deployed in the early 2030s, that employs a thermal-spectrum, graphite-moderated, near-atmospheric pressure, self-contained and integrated reactor design. The standardized dual IMSR nuclear facility is designed to operate at high temperature and low pressure (near atmosphere) with a rated thermal capacity of 884 MW, providing 390 MWe (net) in electrical power, or 822 MWth (net) at 585°C of thermal power, or a combination of both, for a broad range of commercial and industrial users over the 56-year plant life. The IMSR fuel salt is selected to have robust coolant properties and intrinsically high radionuclide retention capabilities, where Standard-Assay, Low-Enriched Uranium (SALEU) is used with less than 5% U-235.

2. Target Application

The IMSR plant is designed to serve electric power grids and industrial heat and power application as represented in Figure 2.

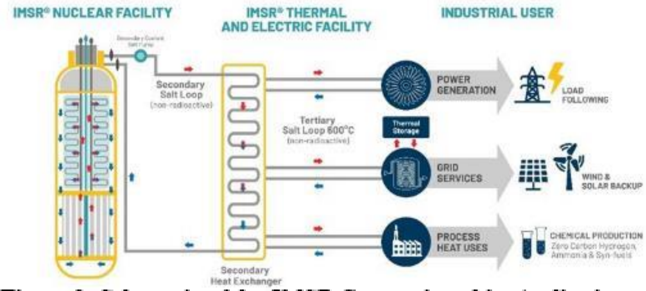


Figure 2: Schematic of the IMSR Core-unit and its Applications

3. Design Philosophy

IMSR design utilizes proven and well-understood techniques, mechanisms, materials and components to reduce challenges associated with novel designs. The IMSR design is heavily based on the Molten Salt Reactor Experiment (MSRE) of Oak Ridge National Laboratory in the 1950's - 1970's, where extensive R&D was done in support of the MSRE design and operation. The IMSR conceptual design and basic engineering program provide a solid foundation for the ongoing detailed design. A number of agreements and Memorandums of Understanding have been signed with fuel suppliers and manufactures to secure the supply chains for the IMSR deployment.

4. Main Features

(a) Nuclear Steam Supply System

The Steam Generation System (SGS) is based on nitrate salt to water/steam heat exchangers, similar to a solar steam generator. Heat is transferred to the SGS from the primary circuit PHXs via two intermediary molten salt circuits in series. The Primary circuit is located entirely within the Reactor Vessel, forming the major part of the Core-unit. The primary circuit is comprised of multiple parallel loops each with a separate PHX, primary pump and piping. Additionally, it includes the reactor core, where the molten fuel salt flowing through vertical fuel channels in the Graphite Moderator acts as both the fuel and the primary coolant.

Heat generated within the Core-unit is transferred to its end use by two intermediary salt circuits: the Secondary Coolant System (SCS) and the Tertiary Coolant System (TCS). The two intermediate circuits in series are designed to enhance safety and improve operability. Each circuit has its own salt pumps, isolation valves and piping. The fluids for both circuits are molten salts, chosen for their excellent heat transfer properties, low vapour pressures and thermal/chemical stability. Downstream of the TCS, the power generated by the Core-unit can be utilized by a Steam Plant to generate electricity, or as high temperature process heat, or some combination thereof.

(b) Reactor Core

The IMSR core uses molten fluoride fuel salt, with graphite acting as the moderator and reflector. Fuel channels are created with the graphite moderator which provides passages for fuel salt to flow upwards, based on forced circulation due to the primary pumps (or natural circulation during certain accident scenarios), to the PHXs. During normal operation, the liquid fuel salt enters at the bottom of the core and flows up through the vertical fuel channels of the core. When the fuel salt passes through the graphite core, criticality is reached, releasing energy via the fission chain reaction within the fuel salt. This energy is mainly deposited within the graphite moderator, which is cooled by the fuel salt, and the fuel salt itself. The heated fuel salt is then pumped through the PHXs which are located above the graphite core, and once is cooled returned to the core inlet via a downcomer duct to complete the circuit. Core integrity is maintained by ensuring that the reactor vessel, which contains the fuel salt, moderator, and heat exchangers, is not damaged because of any temperature excursion past safe levels.

(c) Reactivity Control

The IMSR design has a strong negative reactivity coefficient of temperature to prevent reactivity excursion accidents. The inherent safety design feature provides a self-regulating, stable temperature regime, and establishes the inherently safe operating profile of the IMSR. The Shutdown Mechanism (SDM) is also provided to be independently capable of a prompt shut down of the reactor from all plant states.

(d) Reactor Pressure Vessel and Internals

The Core-unit consists of the following key components:

- Reactor Vessel (RV)
- Reactor core and internals, including Graphite Moderator and Reflector
- Primary Pumps
- PHXs
- Emergency Heat Exchangers (EHXs)
- Thimbles, housing the Shutdown Mechanism (SDM) & instrumentation

Additional supporting components are also housed within the Core-unit including fuel salt and off-gas transfer lines, shielding, and internal supports.

(e) Reactor Coolant System and Steam Generator

The IMSR transfers the heat from the reactor to the steam plant for electricity production or for direct use in industrial heat applications. The IMSR plant's design profile and simplified schematic demonstrating the transfer of nuclear heat to electric power production and/or a variety of industrial applications is shown in Figure 2. Two intermediate coolant systems in series, the SCS and the TCS, are designed to enhance safety and better operability. Each system has its own salt pumps, isolation valves and piping. The fluids for both systems are molten salts, chosen for their excellent heat transfer properties, low vapour pressures and thermal and chemical stability. The SCS transfers heat away from the primary heat exchangers (PHXs) integrated inside the Core-unit to the TCS, which is pumped out of the nuclear island to a separate building where it heats steam generators that generate superheated steam for either power generation or process heat applications.

(f) Pressuriser

A pressuriser is not required for the IMSR design. The RV of the Core-unit operates at low pressure.

(g) Primary pumps

The primary pumps are based on the MSRE salt pump design, and are modified to suit the IMSR Core-unit geometry, pump duty and operating conditions. Pump motors are located above the Reactor Vessel head, with shafts extending into the Core-unit and pump volutes immersed in the Fuel Salt. Design co-ordination work with a major nuclear pump design and manufacturing company is in progress.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Safety management of IMSR technology is less complex than conventional nuclear reactors and is founded on the three concepts of nuclear safety. Control the reactor, Cool the fuel, and Contain the radioactivity. The IMSR design achieves these goals by taking advantage of the passive and inherent attributes of IMSR technology that are set out in the table below.

Control	Cool	Contain
IMSR has a strong negative reactivity coefficient of temperature for inherent reactivity control. Passive shutdown that does not rely on the traditional use of control rods.	IMSR design assures heat dissipation in all circumstances as IMSR fuel salt is molten and serves also as the coolant. This uniquely enables convective cooling of the nuclear fuel. IMSR design is small and operates at 700°C. Small high temperature reactors have greater inherent cooling capabilities.	IMSR fuel salt forms strong ionic chemical bonds with many fission products to uniquely contain radioactivity within the salt and the reactor. No chemical driving forces are present on the containment, in part, as zirconium metal-water reactions are not possible. No physical driving forces for fission products release out of the containment as IMSR operates at atmospheric pressure.

(b) Safety Approach and Configuration to Manage DBC

For the IMSR design, Postulated Initiating Events (PIEs) have been systematically identified employing both top-down and bottom-up processes by Probabilistic Safety Assessment. PIEs cover all credible failures or malfunctions of SSCs, operator errors, Internal and external hazards (human-induced or naturally occurring). All plant operating modes are considered. Generic site conditions are assumed.

(c) Safety Approach and Configuration to Manage DEC

Safety design principles are applied in the IMSR design to enhance defence in depth, including safety classification, equipment qualification, redundancy, diversity, segregation, physical separation, independence, single failure criterion, fail safe design. IMSR design has incorporated five levels of defence in depth aimed at preventing accidents and ensuring additional protection if prevention fails. As a Generation IV reactor, the IMSR is designed to eliminate the need for offsite emergency response.

(d) Containment System

The IMSR Containment system consists of a leak-tight fully metallic envelope that houses radioactive materials. This includes radioactive materials (irradiated fuel salt and off-gases) in the Core-unit, Fuel Salt Storage Tanks, Gas Holding Tanks, Hot Cell, and connecting piping.

(e) Spent Fuel Cooling Safety Approach / System

The irradiated fuel salt will reside in the Fuel Salt Storage Tanks, on-site within the containment, which are cooled by a combination of passive and active design features. After the decay heat reaches a sufficiently low temperature, the fuel salt is transferred to spent fuel storage tanks which are also within Containment and cooled passively through heat transfer to the ambient.

6. Plant Safety and Operational Performances

The IMSR technology meets the safety principles established by the Global International Forum for Generation IV nuclear energy – "*operations will excel in safety and reliability*." The quantitative safety goals are aligned with the Canadian Regulator's (the CNSC) expectations. The concept of severe accident with core melting does not apply to the IMSR since the fuel is already in a molten state for normal operation. The large release frequency is less than 10^{-8} per reactor year. The plant design incorporates automation features to maximum extent possible and the relative simplicity reduces need for large staff complements. The plant layout is optimized to accommodate nuclear safety features such as fire protection, radiation protection, security, and safeguards. The IMSR400 capacity factor is 95%, with 42 outage days planned for Core-unit replacement (every 7 years), and other maintenance (planned and unplanned) expected to require 63 outage days every 7 years. The IMSR is designed to meet the latest Canadian and international nuclear safety requirements and standards. It achieves excellence in nuclear safety through its inherent safety characteristics and passive design features. Defence in depth is applied throughout the design process of the IMSR to provide a series of levels of defence aimed at preventing accidents and ensuring additional protection in the event that prevention fails. The IMSR design has implemented the necessary provisions to prevent accidents and to mitigate the consequences of any accidents that may occur.

7. Instrumentation and Control Systems

The I&C for the IMSR plant is not as challenging in terms of complexity and performance due to the passive and inherent safety design features of the IMSR design. The IMSR is designed to have a strong negative temperature reactivity coefficient. This design feature allows reactor power to be inherently controlled to demand power, thus it makes the system easier to control and eliminates the need for in-core reactivity control devices. The IMSR does not have any control rods for power control, as the core temperature acts like a control device, which inherently adjusts itself so that reactor power matches demand power in a relatively short period of time. The I&C system's main functions deal fundamentally with integrated control of production, interlocks for safety coordination, and monitoring system status. The control system is segregated into seismically qualified and non-seismically qualified partitions. The control system employs redundancy to achieve high reliability and fault tolerance throughout the system.

8. Plant Layout Arrangement

The generic design of the dual IMSR nuclear facility consists of two, 442MWth operational reactors serviced by one Control Building, but each with their own Turbine Building. The design considers site characteristics with a wide spectrum of the foundation media ranging from soil sites to hard rock sides. It is expected that site characteristics for the generic IMSR design will envelope and be acceptable for site-specific conditions.

Typically, two transmission lines are connected to the switchyard with two connections to supply the house loads of each unit via transformers. The System Service Transformer (SST) is one supply path and the Main Output Transformer (MOT) and Unit Service Transformer (UST) form the second supply path. A thermal output only plant is similar except that there is no MOT and the UST is replaced with a second SST.

Figure 3: Dual IMSR Nuclear Facility Representation

9. Testing Conducted for Design Verification and Validation

Terrestrial Energy has completed the first phase of testing to confirm Fuel Salt thermo-physical properties. The tests accounted for Fuel Salt aging - build-up of fission products. Verification of the experimental findings' reproducibility will follow. Testing of Fuel Salt redox potential and respective interfacing material response is in-progress. Solubility limits of trifluorides and iodides inside the salt are in progress. Salt chemical and mechanical interactions with non-irradiated graphite is completed. Analogous test using the irradiated graphite specimens will follow. Irradiation of selected graphite grades is in progress and will conclude with the property-testing of the irradiated graphite. The test devoted to the irradiated alloy property testing entered its detailed design phase. Alloy corrosion testing is in progress. Waste management feasibility study is completed, and experimental phase is being planned. Reactor physics, Thermal-hydraulics and instrumentation tests are being designed and planned.

10. Design and Licensing Status

The successful completion of the IMSR Vendor Design Review (VDR) Phase 1 and Phase 2 (the first for an advanced reactor in Canada), with a conclusion of no fundamental barriers to licensing the IMSR plant, has provided Terrestrial Energy solid evidence for the argument of licensing readiness in Canada. The IMSR VDR Phase 1 and Phase 2 are good and available practice to reduce the licensing risks and uncertainties within the Canadian regulatory framework. The IMSR pre-licensing regulatory engagement with the US NRC provides a great opportunity to reduce the risks and uncertainties for advanced reactor licensing application in the US.

11. Fuel Cycle Approach

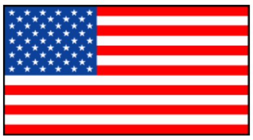
The IMSR, as a molten salt reactor, is fuelled by a proprietary fused mixture of low enriched UF_4 and other molten fluorides, operating on a once-through fuel cycle with no reprocessing, but partial recycling of fuel between successive Core-units. Lifetime reactivity control is maintained in the IMSR by means of negative temperature feedback and the regular addition of makeup fuel salt to compensate for burnup. Excess positive reactivity is not introduced during initial fuelling as is typically done for light-water reactors, thus avoiding the use of burnable absorbers.

12. Waste Management and Disposal Plan

The generation of spent fuel salt is minimized by the design of the IMSR, which allows for recycling of the fuel salt, where the initial fuel salt load of one Core-unit is provided by the irradiated fuel salt from the previous Core-unit. The only requirement for fresh Fuel Salt is to maintain criticality during operation, which results in a significant reduction in the overall spent fuel salt volume over the IMSR's plant life. Spent Fuel Salt is to be conditioned into a glassy-ceramic wasteform using the Synroc process for intended storage in a deep geological repository. By offering a conditioned waste form that can offer improved performance while still being compatible with proposed SSCs for handling existing waste stockpiles, waste disposal costs are expected to be minimized by bypassing a need for developing waste-specific handling equipment and procedures for geological storage and/or allowing for a more space-efficient design of any geological repository.

13. Development Milestones

2015	Conceptual design completed
2016	Start of basic engineering phase
2017	Completion of CNSC pre-licensing Phase 1 Vendor Design Review
2017	US DOE LPO for up to $890Mn project financing support for first US IMSR Plant
2018	Commenced CNSC pre-licensing Phase 2 Vendor Design Review
2019	Commenced irradiation test program on graphite
2019	US NRC and CNSC conduct joint review of key safety analysis
2020	IMSR Plant short-listed to final three for final evaluations at DNNP
2023	CNSC VDR Phase 2 completed with no fundamental barriers to licensing identified
2025	Planned – Commence licensing at first site in North America
2027	Planned - Commence construction of a first full-scale IMSR
2033	Planned - First IMSR plant in-service



KP-FHR (Kairos Power, United States of America)

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Architect's rendering of the Hermes demonstration reactor facility to be constructed in Oak Ridge, TN

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 (Flibe) / 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 (°C)	550 / 650
Fuel type/assembly array	TRISO particles in graphite pebble matrix / pebble bed
Power conversion process	Superheated steam cycle
Fuel enrichment (%)	19.75
Refuelling cycle (months)	Online refueling
Reactivity control mechanism	Control elements, boron
Approach to safety systems	Passive
Design life (years)	20 (vessel), 80 (plant)
RPV height/diameter (m)	7.2 / 3.9
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

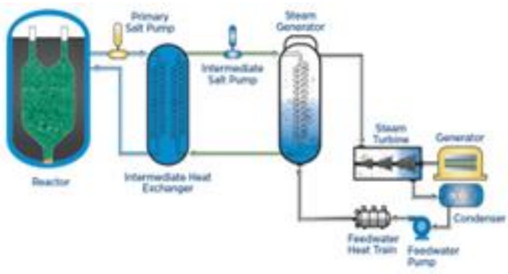
1. Introduction

Kairos Power is a mission-driven company singularly focused on its effort to commercialize advanced reactor technology in time to play a significant role in the fight against climate change. Kairos Power is disrupting the industry with rapid iterative development and vertical integration strategies to deliver a clean energy solution with robust safety at an affordable cost.

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.

2. Target Application

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. The KP-FHR aims to be cost competitive with natural gas.



KP-FHR Power conversion system diagram

3. 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.

4. Main Design Features

(a) Power Conversion

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.

(b) Reactor Core

4.0-cm diameter 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) 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. The number and the location of the shutdown elements are selected to provide sufficient shutdown margin at zero-power conditions. The shutdown system is designed with sufficient reactivity worth to shut down the core from hot full power assuming the failure of the highest worth control element. The reactivity control elements can be inserted for shutdown at a rate sufficient to assure that the design limits for the fission product barriers are not exceeded.

(d) Reactor Vessel and Internals

The KP-FHR reactor vessel operates at low pressure, so has thin-walled construction using 316H stainless steel that complies with ASME Section III, Division 5 requirements. The reactor internals include a core barrel and reflector support structure that position and retain the graphite reflector structure which maintains alignment of the structure to form the major flow paths and reactor core volume. As the pebbles are positively buoyant in the coolant, defueling occurs from the top of the core and online fuel addition from the bottom. The reactivity control and other components interface through the top head.

(e) Fuel Characteristics

Kairos Power’s reactor uses fully ceramic fuel, which maintains structural integrity even at extremely high temperatures and will be undamaged to well above melting temperatures of conventional metallic reactor fuels.

(f) Fission Product Management

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.

5. Safety Features

(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 coolant 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 evaporates water to provide heat removal and does not require inventory makeup for a minimum of 72 hours. System valves are fail-safe and do not rely on safety related electrical power or operator action.

(c) Emergency Core Cooling System

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.

(e) Chemical Control System

A Chemistry Control System (CCS) monitors the salt coolant to detect impurities, ensuring it stays within an established operating window.

(f) Spent Fuel Cooling Safety Approach / System

The zirconium cladding used in light-water reactor fuel contains substantial stored energy, which affects the safety significance of spent fuel cooling systems. KP-FHR fuel is fully ceramic, which simplifies cooling requirements for fuel in the pebble handling system and also in canister storage.

6. 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.

7. Instrumentation and Control System

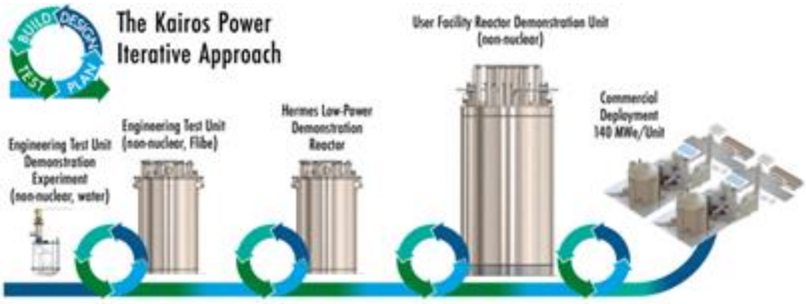
The Kairos Power Instrumentation and Control Systems takes advantage of the inherent safety features of the KP-FHR technology to simplify the reactor protection system configuration and automated actions. The system is sufficiently simplified to allow for deterministic analysis of all design basis events. 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.

8. Testing Conducted for Design Verification and Validation

Kairos Power is committed to a culture that embraces iterative development and facilitates an integrated design philosophy, testing program, and licensing approach to mitigate technical, licensing, manufacturing, and construction risk while establishing cost certainty through iterative hardware demonstrations.

By mirroring SpaceX and its successful rapid spiral development, Kairos Power accelerates test cycles for innovation and optimization. Our rapid iterative approach leverages multiple design-build-test cycles with both nuclear and non-nuclear systems prior to the first commercial reactor. Major hardware iterations include:

Engineering Test Units (ETUs) – non-nuclear systems that test mechanical designs, scale up Flibe production, establish manufacturing infrastructure, and build a sustainable supply chain for raw materials and off-the-shelf components. ETU 1.0 completed 2,400 hours of pumped salt operations in February 2024 and entered decommissioning. ETU 2.0 will begin operation in 2025.



Kairos Power's Rapid iterative approach to hardware development

Hermes – a 35 MWth non-power reactor that will demonstrate Kairos Power's ability to produce affordable nuclear heat. Currently in the design phase, Hermes is scheduled to be operational in 2026.

U-Facility – a non-nuclear demonstration intended to test the manufacturing and construction of primary reactor systems, serve as a training center, and reduce O&M cost uncertainty

KP-X – a first-of-a-kind 140 MWe commercial KP-FHR operating at grid-scale

9. Design and Licensing Status

Kairos Power received its construction permit (CP) for the Hermes low-power demonstration reactor from the U.S. Nuclear Regulatory Commission (NRC) in December 2023. Scheduled to enter operation in 2026, Hermes is a reduced scale (35MWth) demonstration reactor that will prove Kairos Power's capability to deliver low-cost nuclear heat.

10. Fuel Cycle Approach

The KP-FHR operates on a continuous refueling cycle. Fuel pebbles are extracted from the core in the reactor vessel and inspected for burnup and integrity in a pebble handling system. Pebbles are either inserted back into the active core or directed to spent fuel storage. Fuel pebbles pass through the core six times on average before reaching full burnup.

11. 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.

Engineering Test Unit 1.0

Salt Lab located in Alameda, CA

RAPID Lab located in Alameda, CA

In-house manufacturing shop located in KP-Southwest

12. Development Milestones

2018	Pre-Conceptual Design Completed
2018	Commissioned R-Lab
2018	Initiation of Pre-Application Review with NRC
2020	Acquisition of KP-Southwest Facility in Albuquerque, NM
2020	Commissioned S-Lab
2020	Formed strategic partnership with Materion Corporation to produce Flibe
2020	Awarded $629M cost-shared Advanced Reactor Demonstration Program Risk Reduction Award from the U.S. DOE
2021	Cooperative Development Agreement with Tennessee Valley Authority to deploy Hermes low-power demonstration reactor
2021	Testing Facility construction completed in Albuquerque, NM
2021	Hermes Construction Permit Application accepted for review by U.S. Nuclear Regulatory Commission
2021	Construction begins on the first Engineering Test Unit (ETU 1.0)
2022	Kairos Power establishes the KP-OMADA Advanced Nuclear Alliance with leading North American utilities and generation companies
2022	Kairos Power commissions Molten Salt Purification Plant to produce Flibe coolant for the ETU series
2022	30,000th pebble manufactured for ETU 1.0
2023	Kairos Power loads 12 metric tons of Flibe into ETU 1.0 and commences molten salt operations
2023	NRC issues Construction Permit for Hermes test reactor
2024	ETU 1.0 completes 2,400 hours of pumped salt operations and enters decommissioning
2024	Construction begins on the second Engineering Test Unit (ETU 2.0)
2024	Construction begins on the Hermes demonstration reactor in Oak Ridge, Tennessee

Stable Salt Reactor-Wasteburner (Moltex Energy, UK and Canada)

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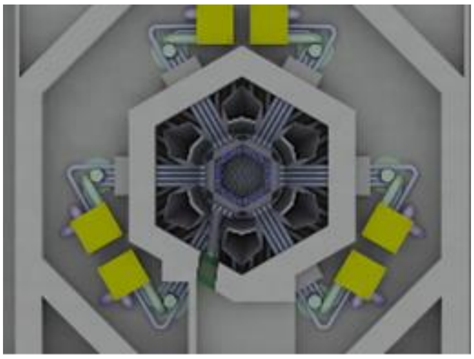


SSR-W Coolant Loop

SSR-W render winter

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Moltex Energy, Canada
Reactor type	Static Fuelled Molten Salt Fast Reactor
Coolant/moderator	Coolant is molten salt MgCl2/NaCl No moderator
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 (primary/secondary), MPa	Atmospheric
Core Inlet/Outlet Coolant Temperature (°C)	575 / 625
Fuel type/assembly array	Molten salt fuel within vented fuel tubes
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	N/A (uses recycled spent fuel)
Core Discharge Burnup (GWd/ton)	~100
Refuelling Cycle (months)	
Reactivity control	Negative temperature reactivity coef, Boron carbide shutdown assemblies
Approach to safety systems	Eliminate hazard, passive engineering
Design life (years)	60
Plant footprint (m^2)	22500
RPV height/diameter (m)	14 / 7
RPV weight (metric ton)	50
Seismic Design (SSE)	Yes
Distinguishing features	Molten salt fuel in conventional fuel assemblies. Thermal energy storage. Very low cost conversion of spent LWR fuel.
Design status	Conceptual Design

1. Introduction

The Stable Salt Reactor - Wasteburner (SSR-W) 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 unseparated from minor actinides and recycled from stocks of spent uranium oxide fuel and produced by a low-cost process called WAste To Stable Salt (WATSS). The reactor outputs its heat as a stream of molten chloride salts, which can be stored in large volumes at low cost, making the facility a high-value peaking plant rather than being restricted to baseload operation. This same system permits the entire steam cycle to be identical to that of coal-fired power stations and for it to be operated completely independently of the nuclear plant. The steam cycle is therefore not subject to nuclear regulations.

2. Target Application

The SSR-W is designed for countries with significant stocks of spent nuclear fuel. The reactor burns all of the higher actinide component of that fuel, leaving a waste stream of relatively short-lived fission products only, as it can recycle actinides from its own spent fuel as well as from spent CANDU or LWR fuel. 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 of national power systems for a low-carbon complement to intermittent renewable energy sources.

3. Design Philosophy

The entire design philosophy is to reduce plant costs by simplifying the design and eliminating hazards instead of merely containing them. This is done by combining the safety and operational benefits of molten salts with those of conventional reactor components. Risks to the public are practically eliminated by design, and not merely contained.

The key features of the design are expected to achieve:

- virtual elimination of the possibility of the release of volatile radiotoxic materials in any conceivable accident, terrorist act, or act of war;
- consequently, approval by regulators for deployment of the SSR-W on smaller sites with smaller emergency planning zones;
- a capital cost as low as $1000/kW when operated as a peaking plant, which is cheaper than fossil fuels, and without relying on subsidies;
- simple manufacture that does not require specialist factories;
- a fuel assembly form that is compatible with IAEA safeguards procedures used in reactors today.

4. Main Design Features

(a) Power Conversion

A molten salt to steam boiler is proposed to generate steam. The turbine is currently being specified for a 300 MWe plant similar to modern fossil fuel plants. There can be a larger turbine of up to 900 MWe installed, to be coupled with the salt storage typically used in large solar facilities, or multiple smaller turbines. This will depend on the local electricity grid needs and economics.

(b) Reactor Core

The reactor core's horizontal cross section is a circle mostly filled concentric hexagonal rings. The innermost rings of hexagons in the core are fuel assemblies, with shutdown assemblies spaced out along the periphery. The gaps between these shutdown assemblies are filled in with permanent reflector and shield assemblies. The gap between the outermost hexagonal ring and the circular tank wall is partially filled by the suction pipes and downcomers of the six coolant loops. The remaining space is filled with a partial hexagonal ring of slots for spent fuel assemblies.

All the hexagonal assemblies are held in position on a diagrid structure at the bottom of the core and by the collective interlocking arrangement of their support structures in the area above the coolant level, to form a tank cover. Reactor coolant flows up the inside of the fuel assemblies, absorbing heat as it rises. Having risen above the fuel assembly to the upper plenum, reactor coolant then flows via six loops to heat exchangers that are outside of the reactor tank, before re-entering the tank in the volume underneath the diagrid.

(c) Reactivity Control

- No significant 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.
- No reactivity shims or control rods are required at any time under normal operating conditions, eliminating the potential for control failures that can lead to an increase in the core reactivity.

- Shutdown is achieved with shutdown assemblies. This is expected to control reactivity through radial neutron leakage.

(d) Reactor Pressure Vessel and Internals

The reactor coolant loops take heat from the fuel assemblies and transfer this heat to the thermal storage system. This is achieved through six coolant salt pumps, which feed six primary heat exchangers by drawing coolant from the tank periphery through the suction pipes. This coolant is then discharged to the lower plenum via the downcomers and jet nozzles. The tube connections to the heat exchangers pass in angled paths through the concrete biological shield, to prevent a shine path (neutron or gamma radiation escaping past shielding). In normal conditions, the flow is forced by pumps to the lower plenum (below the diagrid) and sucked from the upper plenum (coolant volume above the height of fuel), while also having a portion of the coolant bypass the suction pipes and flow freely down the tank wall to the lower plenum.

(e) Fuel Characteristics

The fuel salt is a mixture of sodium and magnesium chloride and actinide and lanthanide trichlorides. It is redox stabilised to render it non-corrosive to steel by inclusion of metal zirconium in each tube, which maintains the salt in a strongly reducing state incapable of dissolving chromium from steel.

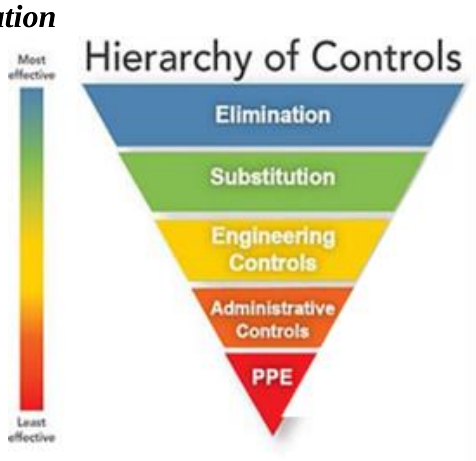
(f) Fission Product Management

The volatile fission products traditionally of concern, such as caesium and iodine, are salts and remain trapped in the fuel. Noble gas radioactive isotopes are held up by the fuel pin vents, which hold it until the gas contains acceptably low radiation levels. Those gases are then filtered by the reactor atmosphere conditioning system.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The design philosophy is to follow the internationally accepted principle of the risk mitigation pyramid. Therefore, the focus is to use inherent characteristics to eliminate hazards and to use engineering safety features to provide additional confidence and backup to the inherent physical characteristics.



Hierarchy of Controls

(b) Decay Heat Removal System

Natural convection of the primary coolant salt would continue in the event of a reactor shutdown or pump failure. Heat would then be transferred to the tank walls, where a finned air duct to atmosphere around the tank walls would take decay heat away indefinitely in an accident scenario.

(c) Emergency Core Cooling System

There is no safety injection system as the coolant has a very high boiling point and there is no credible leakage path to allow the core to be uncovered.

(d) Containment System

There are no significant internal pressures within the SSR-W. The primary containment for the reactor is the tube wall. The secondary containment is the coolant salt, which absorbs fission products in the event of tube failure. The third is the tank itself. The fourth is the concrete structure. Above the tank is an argon space, which has a stainless steel liner surrounded by a ~1 m concrete wall serving as the biological shield. The reactor building walls are ~300 mm thick reinforced concrete. This serves as both a building structure and a shield.

(e) Chemical Control

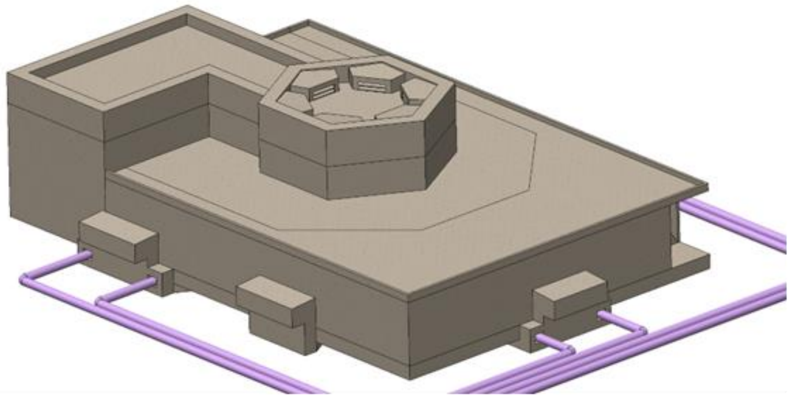
The reactor coolant salt chemistry is maintained in a reducing state through the use of an electrolytic cell (the redox control unit). The flow of coolant salt for the reactor coolant redox control unit is normally provided by the leak-off flow that is generated in the reactor coolant loop vacuum break line when the reactor coolant pumps are in normal operation

(f) Spent Fuel Cooling Safety Approach / System
Several positions along the periphery of the reactor allow the fuel assemblies to cool down after being removed from the core. Afterward, they can be brought back into the WATSS process to allow them to be recycled into new fuel for the reactor.

6. Plant Safety and Operational Performances
The design intent is that the release of radioactivity in excess of the agreed operating discharge authorization is expected to be less than 10^-6 per year. The ramp rates of the plant will be driven by the steam side, not the nuclear side.

7. Instrumentation and Control Systems
There is a separate protection and monitoring system. The protection system is hardwired for a four-channel coding system. The control system is a supervisory control and data acquisition (SCADA) system. The monitoring system is a distributed data acquisition and monitoring system with standard monitoring software.

8. Plant Layout Arrangement
The reactor biological shield (RBS) has a hexagonal cross section. The RBS is built up on the reactor basemat and both can effectively be considered as different parts of the same civil structure. There is a lining on the inside of the RBS Vessel that forms part of the reactor containment. The reactor tank is in the lower volume of the RBS vessel; the upper volume encloses the containment gas space above the reactor coolant level and provides a boundary for this gas space. The reactor tank is enclosed by the concrete RBS vessel, which provides protection against radiation from the reactor core and coolant.



SSR-W Civil Structure

9. Testing Conducted for Design Verification and Validation
The SSR-W and WATSS design is sufficiently developed to support an R&D plan that is now being executed. The novel componentry of the SSR-W includes the primary heat exchanger, the fuel assembly's gas venting facility, emergency heat removal system heat exchangers, and the fuel handling infrastructure. A medium-scale (<1tonne of molten salt) thermal hydraulic loop has been built to demonstrate the phenomena that need to be accurately modelled by the appropriate software.

10. Design and Licensing Status
Moltex was selected by NB Power and the Government of New Brunswick to progress development of the SSR-W in New Brunswick, Canada, with the goal of deploying its first reactor next to the Point Lepreau Nuclear Generating Station. In preparation for this, the SSR-W design is being subjected to the CNSC Vendor Design Review (VDR) process. The design has successfully completed Phase 1 of the VDR in 2021.

11. Fuel Cycle Approach
Current commercial reactors run on the once-through natural uranium fuel cycle, which involves a single pass of the fuel through the reactor. Moltex plans to produce the fuel for the 300 MWe SSR-W (SSR-W300) from used fuel stocks using the WATSS process. WATSS separates out the fission

products into a low-volume high-level waste stream, produces a high-volume low-activity depleted uranium alloy and the SSR-W300 fuel.

12. Waste Management and Disposal Plan

Moltex's WATSS and SSR-W technology consume and destroy a very large proportion of the long-lived higher actinides in spent fuel, while generating useful power. S A very small amount of other radionuclides present in CANDU fuel are not consumed and remain as waste which cannot be destroyed by any nuclear fission process. Most of these radionuclides will be encapsulated and can be safely disposed of in a repository.

13. Development Milestones

2014	UK patent granted for use of unpumped molten salt fuel in any reactor
2014	Independent capital cost estimate complete.
2015	Pre- conceptual design complete and key claims validated by the UK's National Nuclear Laboratories.
2017	Conceptual design completed and CNSC VDR commenced.
2018	Master patent on static fuelled molten salt reactors granted in major geographies. Several other patents progressing through PCT process.
2018	Moltex successful in UK Government Advanced Modular Reactor Competition.
2018	Moltex signs agreement with NB Power to advance the development of the first commercial demonstration SSR-W in Canada.
2020	Moltex signs ONWARDs agreement with Ontario Power Generation.
2021	Moltex receives funding from the Government of Canada's Strategic Innovation Fund.
2021	Moltex completes CNSC VDR Phase 1.
2022	Moltex announces strategic partnership with SNC-Lavalin Group
2023	North Shore Mi'kmaq tribal council and its seven First Nation member communities make financial investment in Moltex
2023	Successful completion of WATSS experiments using simulated fuel
2024	WATSS patents registered
2024	SSR-W integrated salt test loop built, and waste recycling demonstrated with advanced modelling techniques

Stellarium (Stellaria, France)

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Stellaria, France
Reactor type	Molten Salt Fast Reactor
Coolant/moderator	Molten Salt/no moderator
Thermal/electrical capacity, MW(t)/MW(e)	250 / 110
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	0.13 / 0.4 (hydrostatic) /0.4 (hydrostatic) / 22
Core Inlet/Outlet Coolant Temperature (°C)	550 / 750
Fuel type/assembly array	Molten Uranium, Plutonium Chloride (optional minor actinides)
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	< 8% $PuCl_3$ or 16.5% HALEU
Core Discharge Burnup (GWd/ton)	100
Refuelling Cycle (months)	240
Reactivity control	Control rods
Approach to safety systems	Passive and Active
Design life (years)	60
Plant footprint (m^2)	45 000
RPV height/diameter (m)	7 / 2.3
RPV weight (metric ton)	30
Seismic Design (SSE)	0.45 g
Distinguishing features	Replacement of vessel every 10 years. In-core closed fuel cycle.
Design status	Conceptual Design

1. Introduction

The Stellarium is a 250 MWth /110 MWe molten chloride salt fast breeder reactor developed by Stellaria, a spin-off of the Commissariat à l'Energie Atomique (CEA, French nuclear energy state agency) and Schneider Electric. It is designed to meet the significant clean energy demands of big industrial sectors, including mines, chemical plants, refineries, factories, data centers and hydrogen production plants, all of which need to decarbonize their energy supply to achieved their goals. The Stellarium can provide direct electricity and/or steam, drawing on decades of R&D experience from CEA in fast neutrons reactors. It is engineered to accommodate various fissile and fertile materials such as Uranium, Thorium, Plutonium and Minor Actinides, contributing to the closing of the fuel cycle by enabling the continous breeding and burning of these materials.

2. Target Application

The Stellarium reactor targets energy-intensive industries and utilities aiming to utilize their plutonium inventories and reduce waste. Built in pairs for maximum economic efficiency, a Stellarium plant can generate 220 MW of electricity with supercritical steam turbines (~50% efficiency) or 500 MW of 600°C superheated steam. The output can be adjusted to provide a mix of steam and electricity, catering to specific applications, such as foundries that need both steam for pre-heating metals and electricity for melting alloys. Stellarium can use various fuel cycles, always combining fertile and fissile materials, including Thorium/Plutonium, Thorium/Uranium, Uranium/Plutonium. Different grades of plutonium or HALEU can be used, along with the option to consume minor actinides. Utilities that choose reprocessing can employ Stellarium to consume plutonium, breed plutonium or uranium 233, and destroy minor actinides.

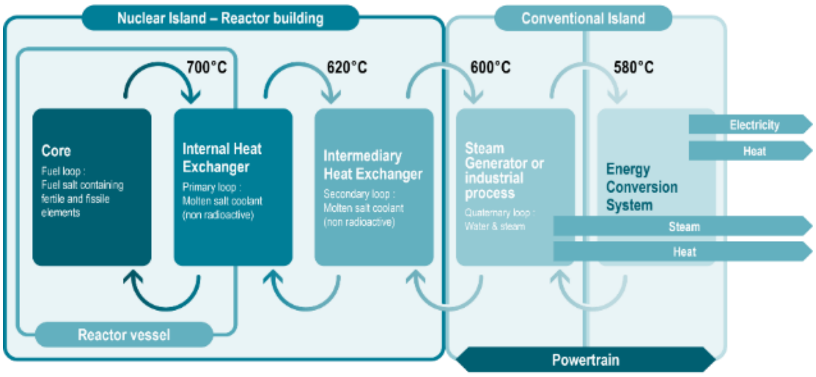
3. Design Philosophy

The Stellarium is designed to satisfy the criteria of a XXIst century GEN IV to ensure high performances and public acceptance. Then the reactor must have : low radioactive rejects in case of severe accident (< 1 mSv a site fence), easy decommissioning even after accident, easy operation (low number of systems, low pressure, complexity…), in-core capacity to close the fuel cycle (iso-generation or breeding), in-core capacity to burn large amounts of minor actinides, capacity to be built almost everywhere (earthquake resistance and size of components allowing road transports), high technological maturity, high dispatchability (equivalent of gas turbine).

4. Main Design Features

(a) Nuclear Steam Supply System

The Stellarium is built with four different loops. By convention the first loop containing the molten salt fuel is labelled “fuel loop”. The primary loop extract the heat form the fuel salt by entering an heat exchanger inside the vessel. As this primary salt will be lightly activated a secondary circuit is needed to avoid any radiological hazard outside the reactor building. Then a Rankine supercritical steam turbine is used to convert heat into electricity with high efficiency (~50%). Steam can be directly used after the steam generator to power industrial processes.



(b) Reactor Core

The reactor core is a critical cavity inside the reactor vessel, the salt is circulated thanks to natural convection produced by the self-heating. The internal heat exchanger is located above the core to facilitate the movement. Around 60 tons of fuel salt are contained inside the vessel, which accounts for ~5 tons of plutonium and ~30 tons of depleted uranium. The breed and burn process allows for 20 years of autonomy, each fission producing energy and neutrons, the later being used to fertilize the uranium.

(c) Reactivity Control

The reactivity control is ensure by control rods. The need for antireactivity being modest because the breeding allows for low excess reactivity.

(d) Reactor Pressure Vessel and Internals

The reactor vessel functions at atmoshperic pressure and is made of Inconel to resist corrosion, irradiation and high temperature. By design its life is limited to 10 years, hence all components are fully exchangeable and easily replaceable. A second, slightly larger vessel encloses the reactor vessel to ensure that any leak is contained.

(e) Reactor Coolant System and Steam Generator

The reactor has a main system for its cooling with the two primary and secondary loops with molten salt coolant (chloride) leading to the steam generators. The steam generators are equivalent to those used in the thermal solar industry. Two specific system are used to remove the decay heat. The first is active and placed on the secondary loops. It uses one small pump on the primary loop and one in the secondary loop. The second is also active and is used to cool the reflector. In case of accident the second system can work in natural circulation and uses water from two big tanks inside the reactor building.

(f) Pressuriser
No pressure inside the different loops except for hydrostatic. A special system allows for the extraction of fission gases from the reactor vessel.

(g) Primary pumps
No pumps inside the reactor vessel and on the fuel loop, the primary and secondary loops have pumps.

5. Safety Features

(a) Engineered Safety System Approach and Configuration
The goal is to reach the 1 mSv at site border in case of severe accident. Thanks to the behavior of molten salt fuels and the absence of pressure the safety approach can be really efficient. In fact only a small number of systems are needed to deal with accidents. Molten salt have really strong negative reactivity coefficient of temperature, which leads to passive shutdown in case of temperature raise. This feedback phenomenon removes all reactivity-induced accidents. For the decay heat removal, the Stellarium has two diversified systems, to cool the salt in the reactor vessel. If the salt has been drained, the subcritical tanks are also cooled. From the containment side, metallic and soluble fission products as well as fertile and fissile materials are trapped in the salt (fusion point at 450°C) and volatile fission products are removed from the reactor plenum thanks to cold traps. The four barriers are at atmospheric pressure.

(b) Safety Approach and Configuration to Manage DBC
The Stellarium exhibits favorable physical responses to most transient. Reactivity insertion accidents (such as the instant withdrawal of all control rods) lead, after a transient power peak, to a controlled salt temperature increase (<100°C) without damage to the main vessel; loss-of flow and loss-of heatsink transients, even unprotected, lead to a similarly reasonable temperature increase, after which the vessel can be cooled for more than 72h by radiation to the passively-cooled reflector. In the case of a main vessel leak, salt flows into the surrounding guard vessel: salt then flows from this vessel to the underlying drain tanks. In all cases the reactor can be restarted after (at most) a vessel change.

(c) Safety Approach and Configuration to Manage DEC
In more severe transients (such as a main vessel leak followed by a guard vessel leak), a majority of the salt can be drained through the connection to the drain tanks. The remaining salt can then be cleaned in the leak-tight reactor pit.

(d) Containment System
The reactor uses 4 barriers to contain the radioactivity inside the reactor building. First is the reactor vessel, second is the guard vessel. The reactor pit is leak-tight and filled with argon to ensure containment and detection of leaks. These three barriers use pressure cascades to avoid passive leaks. The last barrier is the reactor building itself. The reactor is burried and only the hall is above ground.

(e) Spent Fuel Cooling Safety Approach / System
The spent fuel is drained by gravity in the crypts under the reactor pits. Several subcritical heated and cooled tanks are there to wait for the fuel to be able to be transported. After one year the fuel can be safely transported away to the reprocessing plant. A special tool is brought to the reactor site to allow the removal of the subcritical fuel tanks.

6. Plant Safety and Operational Performances
The Stellarium reactor is design to fit inside the existing French Autorité de Sûreté Nucléaire (ASN, French nuclear regulator) framework. Studies are ongoing at the European level to harmonize parts of the differents existing frameworks and Stellaria will include the conclusions to the design. The Stellarium is also design to be built everywhere in Europe and in the majority of the countries. That means that stronger design earthquakes have been taken into account. A standard power plant will include two reactors with a 100-people staff including guards. The plant layout is optimized to accommodate all requirements regarding safety, security, safeguards, and environmental protection. The reactor vessel is replaced every 10 years during a 6-months outage period corresponding to the 10-yearly visit. The reactor is refueled every 20 years thanks to its core breeding ratio of 1.

7. Instrumentation and Control Systems

Thanks to the deep partnership of Scheider Electric the I&C systems of the Stellarium will be state-of-the-art and include the latest innovations. Nuclear instrumentation technologies relevant to fast neutron molten salt reactors have been developped in the past for sodium fast reactors. All nuclear instrumentation is ex-core. The conventional instrumentation is taken from thermal solar technology.

8. Plant Layout Arrangement

The typical power plant includes a reactor building housing two 250 MWth reactors each producing 110 MWe. The steam generators are in an adjacent building, supplying steam to the turbine building. The reactor building is 70 meters long, 30 meters wide and half burried with 15 meters above ground. Cooling options include river cooling, wet cooling tower, dry cooling towers. For industrial steam or heat production the layout may include additional buildings.

9. Testing Conducted for Design Verification and Validation

Stellaria is conducting tests on molten salts loops to verify the thermo-physical properties of selected molten salts and performing chemistry studies in a dedi c ated lab. A critical mock-up will be built by 2027, aiming for criticality by 2028. In the early 2030's, a 50 MWth Firsk-of-a-Kind (FOAK) vessel will be constructed for reactor-scale demonstration, with plans to replace it later with a full-size vessel (250 MWth) connected to the grid.

10. Design and Licensing Status

The consortium made of Stellaria, Orano and the CEA has been awarded by the French governement with a 10-million grant (France 2030 phase 1) and has begun discussion with French ASN in the preliminary screening framework. Preliminary design phase is over as end of september 2024. Preliminary design for the critical experiment and conceptual design for the reactor are due to begin by end of Q3 2024.

11. Fuel Cycle Approach

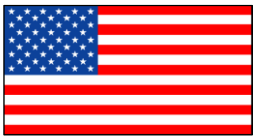
Stellaria's fuel cycle philosophy emphasizes closing the fuel cycle through advanced reprocessing capabilities. The Stellarium reactor is designed for flexibility, allowing it to use various fuel types, including plutonium from PWR/BWR/PHWR spent fuel, depleted uranium, and minor actinides. Stellarium can manage different plutonium qualities and incorporate up to 30% minor actinides in its fuel mix. After 20 years of operation, the reactor is unloaded, and the fuel is sent to a reprocessing plant, where fission products are extracted, leaving the plutonium mass largely unchanged. This allows the cycle to be closed by creating new fuel salt with only depleted uranium make-up. The plutonium can then be reused in Stellarium or other reactor types. For countries without plutonium, Stellarium can be started with HALEU, eventually transitioning to a closed uranium-plutonium cycle after a few cycles. Additionally, the reactor can also be initiated with HALEU and thorium or plutonium and thorium for countries interested in using thorium.

12. Waste Management and Disposal Plan

As the Stellarium burns the minor actinides the only waste management needed concerns : structural materials (vessels…), effluents and fission products. The Stellarium is designed to fit inside the existing French ecosystem with the fuel reprocessing separating useful materials (uranium, plutonium, thorium) from the wastes (fission products and minor actinides). The minor actinides being separated from the fission products in the next plant (La Hague 2, 2045), they will be burnt inside the Stellarium. Then only vitrified fission products will be sent to the Deep Geological Repository (DGR).
The vessel weights approximately 30 t and is replaced every ten years. This metal will be heavily irradiated and will be cut off and compacted into standardized container to fit in a DGR.

13. Development Milestones

2020 – 2023	Preliminary studies and technological innovation (using previously developed patents).	Complete
2023 – 2024	Pre-conceptual design phase and technology validation	Complete
2024 – 2026	Conceptual Design Phase (and preparation for pre-licensing)	On track
2028 – 2030	Basic Design Phase	Planned
2028	First criticality of critical experiment	Planned
2033	Target first criticality for the FOAK plant	Planned
2036	Target first criticality for the first NOAK plant	Planned



Thorcon (Thorcon International, United States of America and Indonesia)

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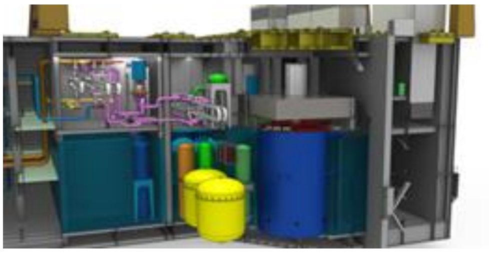
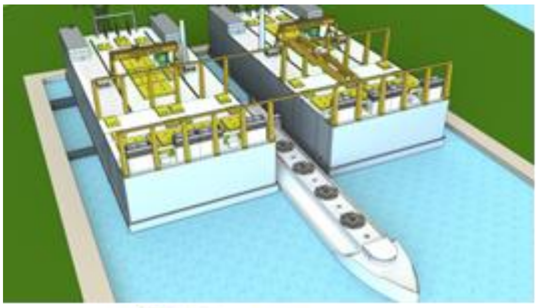
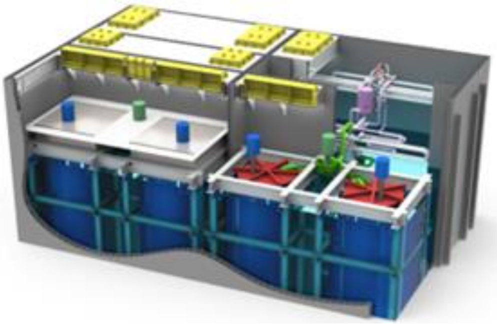






MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Thorcon International, first deployment in Indonesia
Reactor type	Thermal molten salt reactor
Coolant/moderator	Molten salts / graphite
Thermal/electrical capacity, MWt/MWe	557 / 250 (per module)
Primary circulation	Forced (4 pumps), 1 per loop
NSSS operating pressures (4 loops: fuel salt, clean salt, solar salt, steam), MPa absolute	1.06 / 1.30 / 0.56 / 25.7
Core Inlet/Outlet fuel-salt temperature (°C)	560 / 704
Fuel type/assembly array	UF_4 / molten salt
Power conversion process	Supercritical steam turbine
Fuel enrichment (%)	Startup 2.3 / makeup 4.95
Discharge burnup (GWd/ton)	145.8
Refuelling cycle (months)	12 months to adding fuel 48 months to Can changeover
Reactivity control mechanism	Negative temperature coef.
Approach to safety systems	Intrinsic, passive
Design life (years)	80
Plant footprint (m^2)	67x162
Can height/diameter (m)	10.3 / 7.8
Can weight (metric ton)	343
Seismic design (SSE)	1.0
Fuel cycle requirements / approach	Fissile LEU05, or LEU19 with thorium conversion
Distinguishing features	Full passive safety, short construction time, low cost
Design status	Preliminary design

1. Introduction

Thorcon is a molten salt fission reactor. Unlike all current operating reactors, the fuel is in liquid form. The Thorcon reactor operates at near atmospheric pressures and is constructed using automated, ship-style steel plate construction methods.

The top picture shows two pairs of 557 MWe reactor Cans in (blue) silos within the hull wall. Each 250 MWe power module contains two replaceable reactors in sealed Cans, depicted in red, which are inside the silos. One Can of each module produces power 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 second picture above shows two 500 MWe Thorcon power plants and a CanShip exchanging the Can containing the reactor vessel and radioactive primary loop. The yellow rectangles are hatches for access by gantry cranes. The middle graphic shows the air cooling towers, fission island, heat exchangers, steam turbine-generator, and switchgear. The bottom graphic also shows the steam generation cell and secondary heat exchanger cells, flooded in water used as a 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 deploys rapidly.

3. Design Philosophy

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. Operators cannot prevent 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.

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 MWe Thorcon prototype can be operating under test in 2026 and then subjected to failure testing before commercial production can begin.

Rapidly Deployable – The entire Thorcon plant is designed to be manufactured in blocks in a shipyard. The 150 to 500 ton, fully outfitted, pre-tested blocks are assembled into a hull containing the complete power plant, towed to a customer site, and firmly settled in 5-10 m of water. A single large reactor yard can turn out twenty 500 MWe Thorcons per year. Thorcon is more than a power plant; it is a system for building power plants.

Fixable – No complex repairs will be attempted on site. Hatches and cranes permit components of the fission island to be replaced. 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.

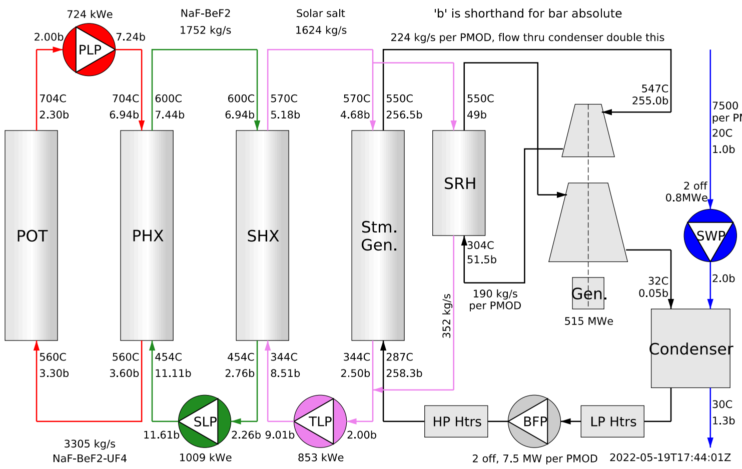
Cheaper than Coal – Thorcon requires far fewer resources than a coal plant. Assuming efficient, evidence-based regulation, Thorcon will produce clean, reliable, carbon-free electricity at less than the cost of coal.

4. Main Design Features

(a) Power Conversion

Each power module employs four loops for converting fission heat to electric power: the primary fuel salt loop inside the Can, the secondary salt loop, a solar salt loop, and a supercritical steam loop.

The solar salt loop captures any tritium that has made it past the secondary loop, and more importantly ensures that a rupture in the steam generator creates no nasty chemicals and harmlessly vents to the Steam Generating Cell via an open standpipe.



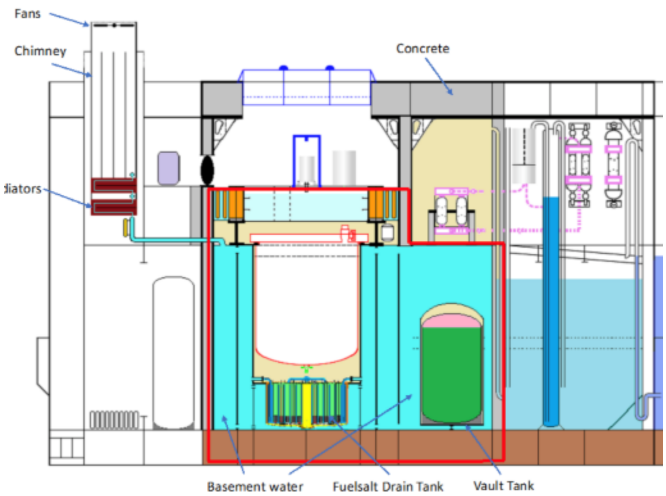
Thorcon is a high temperature reactor, with a thermal efficiency of 46.5% using 30°C seawater compared to about 32% for a standard light water reactor. This reduces capital costs and cuts cooling water requirements by 45%. It also allows Thorcon to use the same steam cycle as a modern coal plant.

The fuel salt is a mixture of sodium, beryllium, and uranium fluorides at 704°C. The (red) Can contains the (orange) reactor called the Pot. The primary loop pump pushes the fuel salt at 3300 kg/s through the (blue) primary loop heat exchanger (PHX).

The PHX transfers heat to secondary salt in (green) piping above the Can. The 560°C fuel salt is piped back to the bottom of the Pot, where the graphite moderator slows neutrons, which fission uranium in the fuel salt as it rises through the Pot, heating the salt. Neutrons also convert some fertile ^{238}U to fissile fuel.

The steel silo cold wall shown in dark blue is cooled by surrounding water. The Can is cooled by thermal radiation to the silo cold wall. The heated water rises by natural circulation to above-deck, air-cooled radiators, where it is cooled and returns to the bottom of the cooling wall via the basement.

A loop of hot secondary salt of mixed NaF and BeF2 is pumped through the PHX in the Can to a Secondary Heat Exchanger (SHX). It transfers heat to a third loop of NaNO3/KNO3, called solar salt because of its use in thermal solar power plants. The solar salt, shown in purple, transfers heat to a supercritical steam generator and steam reheater.



All radioactive fission products and fissile material are contained within the Consolidated Boundary indicated by the red outline. The space is shielded by steel sandwich walls protecting the crew and prevents intrusion. This space is guarded by IAEA seals and monitoring.

(b) Reactor Core

The reactor core is inside the pot. The core is 90% filled with graphite logs that moderate neutron energies. The core is 6.5 m in diameter and 4 m high.

(c) Reactivity Control

Reactivity is controlled by fuel salt temperature, which rises as steam turbine power generation heat demand drops. Fission decreases as salt, graphite, and Pot temperatures rise. Temperature adjustment rods compensate for neutron absorption by xenon during power changes.

(d) Reactor Pressure Vessel and Internals

The Pot reactor vessel pressure is 0.33 MPa, insufficient to spread radioactive material into the environment in an accident, so it does not have the central safety importance that it does in an LWR. The Pot contains circulating fuel salt, graphite moderator, shutdown rod channels, and temperature adjustment rods.

(e) Fuel Characteristics

The fuel salt is NaF-BeF_2-UF_4 72/16/12. As fissile U is consumed fissile Pu_{239} is generated, but not enough to replace the fuel burned. Makeup fuel must be added continuously.

(f) Fission Product Management

Fission product off-gases include xenon and krypton, mixing into helium cover gas flowing over the fuel salt surface in the Pot header tank. The off-gas mix flows through tanks inside the Can to allow most of the radioactive gases to decay to solid fission products which are captured within. The cooled off-gas flows to holdup tanks and charcoal beds in the SHX to allow the longer lived radioactive noble gas to decay except Kr-85 and tritium. The xenon and krypton are removed to tanks and the helium recycled.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The Thorcon negative temperature coefficient provides passive temperature stability. The large margin between the operating temperature of 704°C and the fuel salt boiling temperature of 1430°C exceeds possible temperature excursions, so radioactive material cannot be vaporized.

(b) Decay Heat Removal System

The FDTs radiate fuel salt decay heat to steel jackets cooled by the same naturally circulating water circuit used to cool the cold wall, passively removing the decay heat to the air cooling towers. No operator intervention is needed at any time. No electric power is necessary, though fans keep basement water cooler. Cooling using natural air flow continues indefinitely.

(c) Emergency Core Cooling System

If the air-cooled radiators in both cooling towers become disabled, basement water provides passive cooling for at least 40 days. Cooling time can be extended indefinitely by supplying more water from sources such as the desalination system, domestic water, ballast tank water, or external water via on-deck fittings.

(d) Containment System

The primary loop is totally contained within the removeable Can and FDT. This system constitutes primary containment for fuel salt during power operations and accidents from power operations. The primary containment for spent fuel during transfer and storage are double walled piping and tanks, also within the consolidated boundary secondary containment. These first two containments are gas tight. Tertiary containment is provided by the ship hull structure of three meters of concrete sandwiched between 25 mm steel plates.

(e) Chemical Control

Makeup UF_4 fuel enriched to 5% ^{235}U is added continuously to increase reactivity. Fertile $^{238}UF_4$ can be added to decrease reactivity. Beryllium metal is added to maintain proper redox potential as fluorine is freed via UF_4 fission. No boron additions occur.

(f) Spent Fuel Cooling Safety Approach/System

Thorcon has two gas tight barriers and one containment barrier 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).

(g) Spent Fuel Salt

Operating with uranium fuel in feed-bleed mode, continuous addition of makeup fuel salt causes removal of used fuel salt from the primary loop to an overflow tank in the Can. Periodically, fuel salt in the overflow tank is transferred to a vault tank. The removed, cooling fuel salt is as well protected as the fuel salt being fissioned. Spent fuel salt may later be transferred to a shipping cask and transferred to the visiting CanShip and shipped to a fuel salt handling facility for future recycling.

6. Plant Safety and Operational Performances

Load following relies on reactor physics. As electric power demand decreases, less heat is removed from the fuel salt by the PHX and reactivity and power decrease as temperature rises.

7. 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. Numerous sensors will record and report the condition of power generation.

8. Plant Layout Arrangement

Two 557 MWt power modules drive a single 500 MWe turbine/generator. This allows using competitively priced, efficient supercritical steam turbine-generators.

9. Testing Conducted for Design Verification and Licensing

Measurements of molten salt properties are being conducted at independent laboratories. Soil testing has been completed in the bay of the island site for the Indonesia demonstration plant. The non-fission test platform is being designed to test components such as molten salts, sensors, pumps, valves, pipes, vessels, graphite moderators, and heat exchangers at operating temperatures and pressures, using externally supplied electric power rather than fission power. This facility will be used to verify safety systems including performance for containment and cooling during severe accidents. Neutronics testing will wait for the demonstration plant.

10. Design and Licensing Status

Preliminary design is complete. Some detailed designs are being discussed with specialty component suppliers. License discussions are continuing with the Indonesian regulator, Bapeten, along with university professors and government officials.

11. Fuel Cycle Approach

Fuelled by 5% enriched uranium additions, Thorcon can operate in feed and bleed mode, continually feeding in fresh fuel salt and bleeding excess volume to the onboard storage vault tanks. Once HALEU becomes available in power plant level quantities, a 500 MWe Thorcon can operate as a thorium converter using 5.3 kg of 19.7% enriched uranium and 9.0 kg of thorium per day, on average. After each 8-year fuel cycle, the used fuel salt is transferred to the vault tanks, which have capacity for 80 years of this operation.

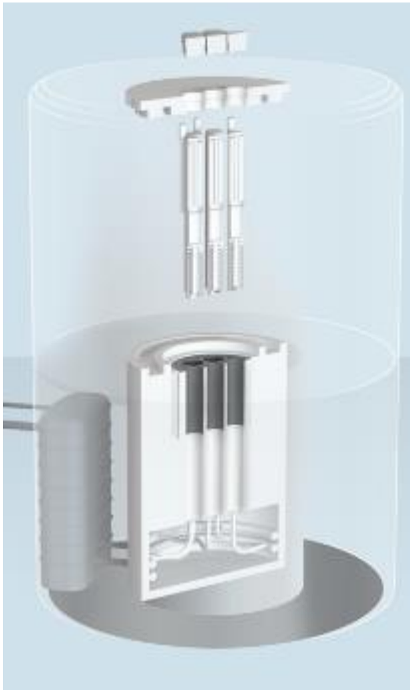
12. Waste Management and Disposal Plan

Spent fuel salt is transferred to passively cooled vault tanks in the SHX cell for storage up to 80 years. Once appropriately cooled, the fuel salt can be transferred to a shipping cask and removed by crane and loaded to a CanShip to transfer it for reconditioning or final storage.

13. Development Milestones

2022	Pre-licensing vendor design review in Indonesia; Preliminary design complete
2023/24	Construction of Non-fission Test Platform; Testing of the Non-fission Test Platform
2025/26	Construction of the demonstration power plant; Begin testing of the demonstration power plant
2027	Complete testing of the demonstration power plant and obtain type license
2028	Begin commercial construction of multiple power plants; Start of commercial operation

THORIZON (THORIZON B.V., Netherlands)



KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	THORIZON B.V., Netherlands
Reactor type	Modular core fast-spectrum reactor with chloride salt
Coolant/moderator	Chloride salt (no moderator)
Thermal/electrical capacity, MW(t)/MW(e)	250 / 100
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	low pressure under all circumstances
Core Inlet/Outlet Coolant Temperature (^{o}C)	500 - 800
Fuel type/assembly array	Fissile-fertile material bearing molten salt in closed containment modules
Number of fuel assemblies in the core	3-7 (current designs, can change)
Fuel enrichment (%)	N/D
Core Discharge Burnup (GWd/ton)	N/D
Refuelling Cycle (months)	5 – 10 years target
Reactivity control	Control rods, burnable poison, core draining, and strong negative temperature feedback
Approach to safety systems	Passive as much as possible
Design life (years)	Full core replacement strategy allows life extension beyond 60 years
Plant footprint (m^2)	N/S
RPV height/diameter (m)	N/A
RPV weight (metric ton)	N/A
Seismic Design (SSE)	N/S
Distinguishing features	Nuclear safe core material replacement strategy, with external core module series production, continuous improvement by module updates, and fuel cycle flexibility
Design status	Conceptual Design

1. Introduction

Thorizon develops a novel molten salt reactor design, based on a modular core approach, in which the core consists of multiple individually contained replaceable modules, named cartridges. When the cartridges are removed, the reactor has no primary circuit and no primary circuit components. The reactor has no fixed large reactor pressure vessel. The technology basis is flexible and allows larger and smaller systems, either by larger or smaller modules, or by more or less modules. The technology offers fuel cycle flexibility, with first goal to burn LWR-plutonium and commercialize, which can be followed by advanced versions on the same technology basis in which fuel cycles can be closed. Thorizon has finalised the conceptual design phase, has a positively evaluated patent, and successfully passed technical due diligences by third parties, and has secured the finances to enter the detailed design phase and execute a core material irradiation program.

2. Target Application

THORIZON offers a technology base for a multitude of different molten salt reactor designs. First goal is to maximize efficient LWR-Plutonium burning within the shortest route to deployment, to reduce the long-lived nuclear waste burden, and turn waste cost into energy generation income. The technology basis combined with the demonstration and commercialisation of Plutonium burners, could allow and support further expansion towards a closed Thorium cycle, or fast spectrum options.

3. Design Philosophy

The largest challenge in molten salt reactors is related to the degradation of core materials. Thorizon's design tackles this challenge by allowing convenient and safe replacement of core materials, while maintaining containment. This approach has led to a design that offers additional advantages, in terms of flexibility, plant lifetime, improvement implementation over the plant lifetime, cost, modularity/series production and practical management of spent fuel and degraded core materials. Another key aspect of the design approach is to avoid complexity, eliminate proliferation risk concerns, and make sure that the design safety and performance can be assessed by basic computational tools, facilitating validation and licensing.

4. Main Design Features

(a) Nuclear Steam Supply System

The NSSS in Thorizon's design is based on a modular core approach where each module contains molten salt with fissile-fertile material. The system operates with forced circulation of the primary molten salt coolant within each module. This coolant transfers heat to a central secondary molten salt system via individual heat exchangers. The secondary molten salt then transfers the heat to a conventional energy conversion system, which can include a molten salt based energy storage systemr, which (flexibly) produces supercritical steam at 550°C. This steam can be utilized directly in industrial processes or for high-efficiency electricity generation. The design eliminates the need for a traditional reactor pressure vessel, instead using a radiation shield and impact barrier to house the modules, each with its own containment and cooling system.

(b) Reactor Core

The reactor core consists of the upper sections of the modules or cartridges, which when placed together form a critical zone. Each individual cartridge is subcritical, unless placed in a critical configuration with other cartridges and moderator/reflector. For IP protection and export control reasons, Thorizon does not disclose information on design geometry and materials.

(c) Reactivity Control

Reactivity control is performed with burnable poison, control rods and passively via large negative temperature feedback. The latter is a specific benefit of molten salt reactors, and utilised in the Thorizon design. To achieve a long cycle, the initial over-reactivity is reduced by burnable absorber in replaceable core materials. Reactivity is quickly reduced in case module pumps stop, as this leads to immediate core draining. Control rod insertion also renders the reactor subcritical.

(d) Reactor Pressure Vessel and Internals

Thorizon has no reactor pressure vessel, but a radiation shield which also serves as external impact barrier, in which modules are placed that each have their own two containments. Each module contains its own pump and heat exchanger, and is closed.

(e) Reactor Coolant System and Steam Generator

The primary circuit consists of individually contained cartridges, in which primary fissile-fertile material bearing molten salt is circulated, for each module individually. The number of cartridges can be varied, as can the size of the cartridge, leading to different system powers. Each of the modules is connected to the same secondary cooling system. The secondary salt transfers the heat to an energy conversion system, providing 550°C steam, which can be used for industrial process directly, or is used for high efficiency electricity production. The steam generator and energy conversion system is off-the-shelf conventional.

(f) Pressuriser
The reactor operates at a low pressure in the primary system, which is secured under all operational and shutdown conditions. This low pressure inherently reduces the need for a pressurizer, as the system's design prevents significant pressure escalation.

(g) Primary pumps
Thorizon's modular core reactor uses forced circulation for primary coolant, with each module equipped with a dedicated pump. This ensures active circulation of molten salt coolant through the closed containment system, enhancing reliability and safety of the reactor's primary coolant system.

5. Safety Features

(a) Engineered Safety System Approach and Configuration
Molten salt reactor technology offers passive safety features, which are maximally exploited by the Thorizon design. In addition, the low pressure in the primary system, which can be secured under all circumstances, reduces the stresses on primary components, and excludes having a significant force to release materials to the outside. Defence in depth is incorporated by adopting multiple containments.

(b) Safety Approach and Configuration to Manage DBC
Thorizon's reactor incorporates a comprehensive safety strategy that leverages its inherent design features and passive safety systems. The reactor's low-pressure primary system minimizes the risk of high-pressure accidents, reducing the potential for material release. The multiple containment barriers, including the molten salt matrix, double containment for each core module, and secondary coolant circuit, work together to manage and contain any incidents effectively.

(c) Safety Approach and Configuration to Manage DEC
The reactor's passive cooling systems, such as the core cavity cooling system that utilizes inert gas for heat removal, provide effective management of heat and prevent temperature escalation. The modular core design also contributes to safety by enabling easy replacement and isolation of affected cartridges. This design approach ensures that even under extreme conditions, the reactor maintains containment and safety.

(d) Containment System
Defence in depth is managed by multiple barriers, taking credit for the absence of primary circuit pressure escalation scenario's under all circumstances

(e) Spent Fuel Cooling Safety Approach / System
Temporary onsite storage of the cartridges is assumed necessary to minimize core replacement outage times. Core section materials are considered medium term waste. Spent fuel salt will be reprocessed after extraction at the reprocessing plant. Non-core section materials can be considered for recycling, economics of which need to be determined.

6. Plant Safety and Operational Performances
Core damage frequency has not been determined yet. It is however to be noted that pressure escalation in the primary containments can be excluded, with a secondary containment to avoid release in case of primary containment failure. Passive safety features, passive reactivity control and external impact barriers, largely reduce core damage and release probability to very low levels. Station black-out scenario is managed by passive cooling as well, providing indefinite grace time.

7. Instrumentation and Control Systems
Plant detailed design is under development. Reactor control and shutdown systems have redundancy. Instrumentation relates to temperatures and pressure. Activity monitoring of secondary and tertiary system, and the core cavity cooling system, and of the space in between primary module containments. Flux measurement in between and outside cartridges.

8. Plant Layout Arrangement
Without cartridges present, the plant has no primary circuit or primary circuit components in the plant, and can therefore ne considered largely conventional. Primary system components are manufactured in series and fuelled outside the plant. At this stage, the purification and conditioning of primary salt

is excluded, hence the plant has no salt reprocessing equipment or facility on site, other than what is included inside cartridges. Current design efforts focus on the primary system, core cavity cooling, reflector-moderator and radiation shielding, to be expanded first to the secondary and tertiary cooling and heat transfer systems. Enveloping building design and control room location and design is under development.

9. Testing Conducted for Design Verification and Validation

An irradiation program of core materials has been initiated. This program will provide material properties as function of irradiation damage and temperature, as a basis for safety and performance analyses, and to determine expected module lifetime. Primary system thermalhydraulic calculations are to be validated with scaled mock-up and a test loop (both under development).
A molten salt irradiation program is also under development. In parallel, salt-material interaction is quantified for all foreseeable salt compositions, both static in laboratory, and by dynamic corrosion testing with a Thorizon designed and built corrosion loops.
The Thorizon concept allows for non-nuclear validation and qualification by a single module mock-up, being representative for the whole reactor system, facilitating the validation, qualification and licensing process. This is foreseen to be established after the current design phase is finalized.
The design has been analysed by thermalhydraulic system code and deterministic and probabilistic core physics codes, reproduced by a qualified external party, confirming no safety showstoppers.

10. Design and Licensing Status

Interaction with Nuclear Safety Authorities in France and Netherlands has started; Site selection is underway, not to be disclosed at this stage. License to build is targeted for 7 years from now, license to operate 10 years from now. It is Thorizon’s aim to accelerate this schedule by co-development with industry as soon as possible, to initiate component manufacture, optimisation and qualification quickly. The licensing process is facilitated by single cartridge non-nuclear demonstration, while the plant build on site is simplified, as the primary circuit is constructed off site, in series, and commence in early stage.
The current phase is intended to be executed by a dedicated in house team of 35 people, supported by external consultants for specific subjects, for which the funding has been secured. Thorizon aims to establish partnerships and to raise additional funding, to support acceleration of the current schedule. Thorizon prefers not to share conceptual technical information in much detail, due to IP protection and export control considerations. The image shown on the first page is a high level, schematic visualisation, showing the module concept and approach, without disclosing technical detail.

11. Fuel Cycle Approach

Modules are replaced as a whole, and salt maximum burn-up is matched by core material maximum lifetime. A 5-10 year cycle is foreseen (with reactor at maximum power 100% of the time). Reactor criticality can match these cycle lengths with the fuel and core design foreseen, with material lifetime confirmation required by material test irradiations.

12. Waste Management and Disposal Plan

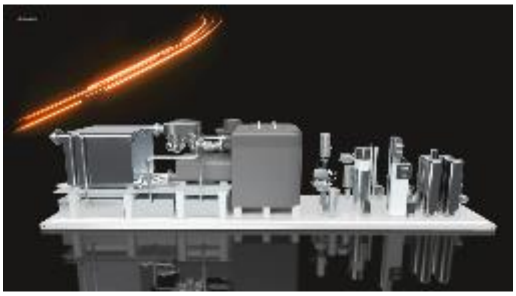
By design, Thorizon systems do not release liquid or solid radioactive waste during normal operation. The cartridges can be transported in shielded casks of manageable dimensions. The salt is extracted for separation and purification and recycling, the core materials are cleaned and prepared for medium term storage. Economics and feasibility of ex-core sections (low-medium level waste) recycling has not yet been determined.

13. Development Milestones

2020	Pre-conceptual design phase, technical due diligence successful, patent positively evaluated.
2022	Investment secured, transition from conceptual design to basic design
2024	Licensing pre-screening successful, preliminary licensing evaluation initiated, salt irradiation project initiated, salt experimental program initiated.
2025	Basic design finalized, site selected, licensing application initiated, transition to final design, demonstration program started

2028-2030	Non-nuclear cartridge prototype operational, validation program finalized, irradiation project finalized, material and component qualification, cartridge off-site series production preparation and execution
2030	Detailed design finalized, license to build, start of on-site construction
2033	Plant construction complete, commissioning of Thorizon One – first of a kind, cartridge introduction, license to operate

XAMR® (NAAREA, France)



KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NAAREA, France
Reactor type	MSR
Coolant/moderator	Molten salt/None
Thermal/electrical capacity, MW(t)/MW(e)	80MWth/40MWe
Primary circulation	Forced
NSSS Operating Pressure (primary/secondary), MPa	Atmospheric pressure
Core Inlet/Outlet Coolant Temperature (°C)	undisclosed
Fuel type/assembly array	Molten salt
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	undisclosed
Core Discharge Burnup (GWd/ton)	undisclosed
Refuelling Cycle (months)	38
Reactivity control	B_4C
Approach to safety systems	Redundant passive systems for decay heat removal, static containment and active reactivity control means.
Design life (years)	40
Plant footprint (m^2)	undisclosed
RPV height/diameter (m)	2,5 / 12
RPV weight (metric ton)	undisclosed
Seismic Design (SSE)	undisclosed
Distinguishing features	The core of the reactor is made up of a plate heat exchanger made of silicon carbide
Design status	Basic Design

1. Introduction

NAAREA (Nuclear Abundant Affordable Resourceful Energy for All) is a French company developing a new energy solution: the XAMR® (eXtrasmall Advanced Modular Reactor), a mass-produced fast modulated micro-reactor fueled with plutonium-chloride (GEN IV reactor). NAAREA's XAMR® will be able to generate electricity and/or heat (80 MWth/40 MWe) from long-lived spent fuel produced by current conventional reactors.

NAAREA’s XAMR® will contribute locally to the development of hybrid energy systems able to secure the energy supply in addition to the electricity grid and massively decarbonizing human activities.

NAAREA's strategy is to sell ”energy as a service” for industrial purposes while ensuring the design, manufacturing, installation, commissioning, operation, maintenance and decommissioning of its fleet of reactors. The innovation of advanced nuclear lies in its ability to attack off-grid electricity markets, medium-temperature (100-400°C) and high-temperature (400-600°C) industrial processes, district heating, electro-fuel production or carbon capture technologies. At first, NAAREA aims to build a prototype of its reactor for validation purposes before deploying an industrial fleet of reactors.

2. Target Application

Some of XAMR®'s technological choices are a direct result of the market analyses conducted earlier, which will allow NAAREA to offer an extremely competitive price to its customers.
Potential markets can be divided into three main categories:

- Sectors linked to the development of "**sustainable cities**", i.e. service activities for populations such as data centers or drinking water supply, but also district heating networks.
- Activities linked to **mobility**, whether light, through the sale of electricity on the electricity grid, or heavy, through the production of electro-fuels.
- **Industrial** sectors, in particular heavy industry, which can be decarbonized directly by supplying heat or electricity, or indirectly by producing hydrogen or implementing carbon capture, utilization and storage technologies.

3. Design Philosophy

The compact design of the generator, the size of a 40-foot container, will facilitate large-scale serial production and will enable easy transport due to its standardized dimensions.
The main features of the reactor design are a molten fuel, which leverages the intrinsic feedback coefficients of liquid fuel; a flushable fuel circuit which allows for limited decay heat removal systems in the core area; two independent sub-critical by design dump tanks including the associated flush trains and passive decay heat removal; a modular conventional island which allows for heat and/or electricity production.
The reactor operating point is around 750°C for the fuel salt, 670°C for the coolant salt and 650°C tertiary salt. This point may evolve depending on the application design studies.

4. Main Design Features

(a) Nuclear Steam Supply System

The XAMR® operates at atmospheric pressure, with the fuel salt being circulated into a closed circuit.

(b) Reactor Core

In the XAMR®, the fuel salt circulates inside a primary salt loop containing a volume, the core region, in which a chain reaction is maintained.
The core of the reactor is made up of a plate heat exchanger made of silicon carbide, a material chosen for its good resistance to temperature, irradiation and corrosion in a chloride environment.

(c) Reactivity Control

Long-term reactivity compensation is done using control drums located inside the core reflector. Short-term reactivity management is achieved mainly through the strong thermal expansion coefficient of the fuel salt, with additional shutdown rods located inside the core reflector.

(d) Reactor Pressure Vessel and Internals

The reactor has no pressure vessel as it operates at atmospheric pressure. The entire fuel loop is located inside a guard vessel made of stainless steel which is designed to funnel large salt leaks to the dump tanks. Reactor Coolant System and Steam Generator
The reactor is cooled using a chloride salt loop which transfers heat from the core to a tertiary circuit connected to the turbine. These systems operates at atmospheric pressure.

(e) Pressuriser

The whole system operates close to atmospheric pressure.

(f) Primary pumps

The fuel salt pump is an centrifugal pump with axial suction and discharge. Containement is ensured through the flushing along the pump shaft. A similar design is used for intermediate coolant.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

As NAAREA aims to deploy small reactors with site constraints being as limited as possible, the design choice made by NAAREA is to implement control drums inside the XAMR® reflector to compensate for the initial excess reactivity. The choice of control drums has been done to limit the axial deformation of the flux profile at the beginning of operation.

It is also necessary to ensure that adequate reactor shutdown can be achieved in any core condition. To achieve this, NAAREA has chosen the same approach as historical molten salt reactors, a combination of shim rods to rapidly stop the chain reaction, followed by flushing of the fuel salt into sub-critical drain tanks.

Decay heat removal (DHR) must be available in any nuclear reactor to ensure that the containment barriers are not mechanically stressed by an unwanted temperature or pressure increase. This is also true for an MSR, and the historical approach is to achieve DHR by natural convection of the drain tank cell atmosphere through a tube lattice containing the fuel salt. NAAREA has elected to pursue this approach, with stringent requirements placed on the drain function and dedicated DHR systems located inside the drain tanks with adequate redundancy. Each drain tank is equipped with one passive DHR system.

(b) Safety Approach and Configuration to Manage DBC

NAAREA aims at no-release in case of DBC through a combination of static containment and delayed filtering in case of airbone fission products release inside the reactor guard vessel. The design goal is to use the flushing approach mostly for DBC3 and 4 and to keep the salt inside the fuel loop during DBC2 to maximize the reactor disponibility.

(c) Safety Approach and Configuration to Manage DEC

DEC are handled through static containement and delayed filtering, with the objective of no evacuation or confinement out of the plant. No release of mechanical energy is expected in case of DEC.

(d) Containment System

Three barriers (the fuel loop, the guard vessel and the reactor building) are engineered for the XAMR®, with a dynamic containement during normal operation and static containement in DBC or DEC.

(e) Spent Fuel Cooling Safety Approach / System

Spent salt is cooled through gas circulation around a closed vessel, combined with a passive decay heat system with an atmostpheric heat sink. Spent salt can be stored in either liquid or solid form in a dedicated tank or solidified in small ingots in a dedicated storage room.

6. Plant Safety and Operational Performances

Undisclosed

7. Instrumentation and Control Systems

Standard instrumentation (nuclear, temperature, pressure, flow) is considered for the reactor, with the objective of limiting the number of on-site workers. The reactor is designed for complete remote routine operation with on-site people only for maintenance purpose.

8. Plant Layout Arrangement

Undisclosed

9. Testing Conducted for Design Verification and Validation

NAAREA is building several natural and forced convection loops to test the equipment currently being designed. NAAREA’s strategy is to carry out full-scale validation of its main equipment (pumps, sensors, DHR systems, control rods).

10. Design and Licensing Status

Currently, NAAREA is in the pre-licensing phase.

No molten chloride salt reactor has been designed or operated in the past, the available experience feedback on molten salt reactors being limited to fluoride-fueled thermal reactors built at ORNL in the fifties and sixties. The current safety framework which has capitalized on several decades of PWR operation is not readily suited to molten-salt reactors, and the solutions which were implemented in the sixties are not adequate today. Furthermore, molten salt reactor design is strongly versatile due to the very nature of the fuel used and the maturity of the various designs currently under development is not sufficient to outline a general safety approach.

11. Fuel Cycle Approach

This reactor is designed for complete integration inside a closed fuel cycle featuring pyro-processing of the spent fuel.

12. Waste Management and Disposal Plan

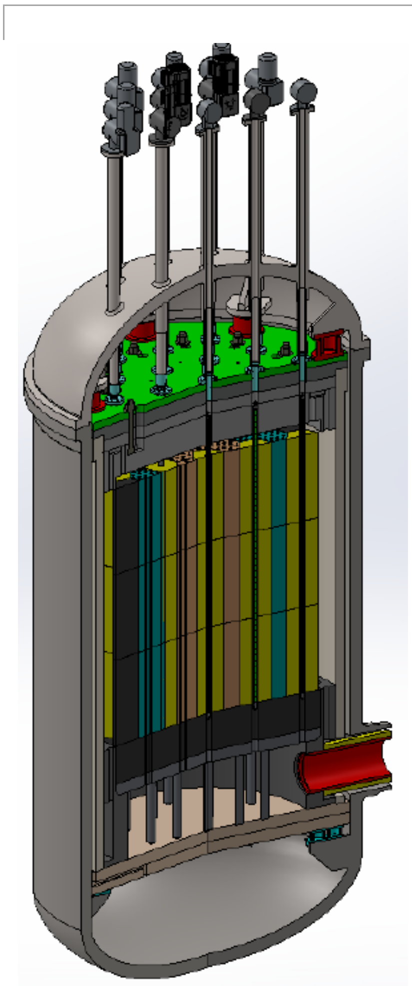
NAAREA is working on developing appropriate dechlorination and cementation processes to adapt its waste production to the current waste management streams.

13. Development Milestones

2021	Start conceptual design phase	Completed
2024	Start prototype basic design and pre-licensing phases	On track
2028	Prototype commissioning	On track
By 2030	Begin commercial operation	Planned

MICROREACTORS

Advanced Micro Reactor – AMR (STL Nuclear (Pty) Ltd, South Africa)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	STL Nuclear (Pty) Ltd., South Africa
Reactor type	HTR, Advanced Prismatic Type
Coolant/moderator	Helium, graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 / 3 for single module plant
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	4 / 1
Core inlet/outlet coolant temperature (ºC)	450 / 750
Fuel type/assembly array	TRISO particles/ LBE eutectic/ SiC
Number of fuel assemblies in the core	420
Fuel enrichment (%)	10 to 20
Refuelling cycle (months)	96
Core Discharge Burnup (GWd/ton)	80-90
Reactivity control mechanism	Control and shutdown rods in core, and the reflector
Approach to safety systems	Passive / inherent decay heat
Design life (years)	40
Plant footprint (m^2)	2 500
RPV height/diameter (m)	5.96 / 2.78
RPV weight (metric ton)	115
Seismic design (SSE)	0.3g (generic site) 0.5g under consideration
Fuel cycle requirements/approach	LEU UO_2
Distinguishing features	Additional SiC barrier; Lead Bismuth Eutectic (LBE) heat transfer
Design status	Pre-conceptual design

1. Introduction

STL Nuclear (Pty) Ltd., the University of Pretoria, the North-West University in conjunction with the South African Nuclear Energy Corporation (NECSA) are developing a 10 MW(t) Small Modular Reactor (SMR) called the Advanced Micro Reactor (AMR). This reactor falls in the category of the High-Temperature, Gas Cooled Reactors (HTGRs). The AMR uses helium as the coolant and is graphite moderated. It uses graphite hexagonal blocks as the moderator and these blocks are arranged to form a cylindrical configuration. The individual SiC fuel assemblies contain TRISO coated particles with either uranium dioxide (UO_2) or uranium oxycarbide (UCO) ceramic fuel kernels of between 10 wt% to 19.9 wt% enriched uranium. The voids between the coated particles are filled with a Lead Bismuth Eutectic (LBE) alloy to provide good heat transfer from the fuel particles to the fuel assembly wall.

2. Target Application

The AMR 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 process heat applications.

3. Design Philosophy

AMR design utilizes proven HTR technologies albeit in a different configuration The reactor is designed to have excess reactivity to operate for several years before refuelling is required. The design is factory assembled to enable road transportation.

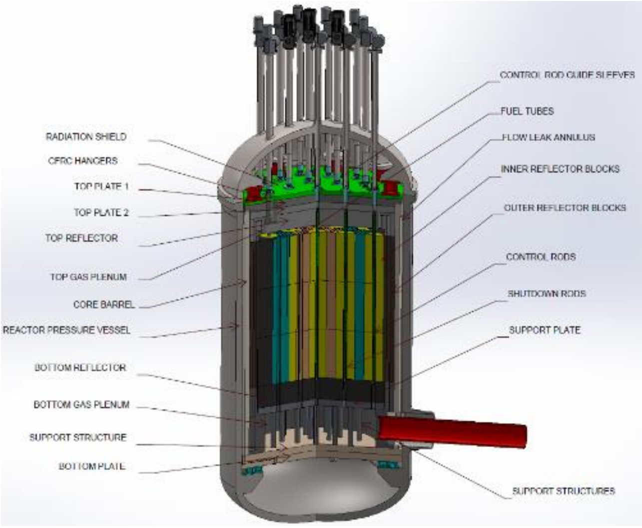
4. Main Design Features

(a) Power Conversion

A Heat Pipe Heat Exchanger (HPHE) is used to transfer heat from the reactor core to the secondary power conversion loop containing air. The LBE alloy is used to absorb heat from the primary helium coolant bundle (located in the centre) and surrounded by a flow-directing shroud. This heat is then exchanged by means of natural convection with the secondary power loop (in the form of a multiplicity of U-tubes) located in the annulus between the flow-directing shroud and the inner wall of the outer shell of the heat exchanger. The LBE returns to the central hot helium bundle at the bottom through several portals in the flow-directing shroud. This flow of LBE in this design can be described as "toroidal natural convection flow".

(b) Reactor Core

The AMR is a high temperature helium-gas cooled reactor with a power of 10 MW(t). The helium coolant is circulated through the reactor core by an electric blower located within the pressure boundary of the heat exchanger. As is the custom for thermal neutron-spectrum high temperature reactors, graphite is used as moderator, which in this design, is in the form of hexagonal graphite blocks packed to form an approximate cylindrical configuration with a diameter and height (including radial reflectors as well as top, bottom and) respectively of ~2.2 m (~1.72 m active diameter), and 2.4 m (active length 2.2 m) with a volume of 5.103 m^3 which results in a core power density of ~2 MW/m^3. The fuel assemblies are silicon carbide tubes that contain the fuel in the form of low-enriched uranium (LEU) dioxide (UO2) or uranium oxycarbide (UCO) TRIstructural-ISOtropic (TRISO) coated particles, immersed in an LBE (45% Pb and 55% Bi) alloy.



Reactor Core

(c) Fuel Characteristics

In typical coated particle. The SiC is the main layer for the retention of fission products. The AMR is also not only limited to kernel and coated particle diameters of 425 microns (μm) and 855 μm respectively but can also utilize kernel diameters of 500 μm and coated particle diameters of 920 μm. The AMR is not restricted to using only high assay low enriched uranium (HALEU) it can utilize enrichments of ~10 wt% to 15 wt%. Approximately ~463 775 coated particles will be contained in a typical fuel assembly.

(d) Fuel Handling System

The AMR is designed to operate between 8 to 10 years without the need for refuelling. Burnable absorbers are used to supress the flux at the beginning of life and extend the life of the reactor as they deplete. The reactor at time of refueling, will be transported back under IAEA seals to the NECSA site where the fuel will be unloaded under continuous IAEA containment and surveillance measures. The spent fuel will be removed and stored during the initial post-operational cooling period on-site in shielded vessels under IAEA seals. After this initial cooldown period and depending on the spent fuel storage strategy in force at the time, it will be transported to its final storage location. A new core will be loaded in the reactor vessel using the same process as described, and the newly loaded vessel will be transported its operational site.

(e) Reactivity Control

As a defence-in-depth, two diverse shutdown systems, each capable of shutting the reactor down with fresh fuel when the reactor is cold and with one member (of highest reactivity worth) stuck in the full extracted position. The shutdown margin is 8%. Thirteen neutron absorber rods are provided in graphite sleeves; 7 within the core and 6 within the side graphite side reflector blocks. There are also 6 shutdown rods located within the core. 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. If all control and shutdown systems are accidentally withdrawn, it will not lead to fuel damage. This is normally defined as a "start-up accident".

(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 the vessel body. The reactor vessel body consists of several forged ring-components circumferentially welded together.

The core structures consist of the metallic parts and the graphite structures. The function of these internal structures is to provide stable core geometry, neutron reflection, cold and hot gas channelling, fuel element flow, shielding, thermal insulation and support of the control and shutdown systems guide tubes and the neutron source. The functional design of the structural core internals is such that they can withstand the steady state and transient loadings during normal operation, anticipated operational occurrences and design basis accidents. The loads borne by the ceramic internals are transferred to the stainless-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.

(g) Reactor Coolant System

The helium coolant enters the reactor vessel at 450°C through the annulus of the co-axial duct. The helium then flows upwards in the helium risers located in the outer graphite reflector. Helium leak flow also enters the annular space between the Core Barrel (CB) and the inside of the Reactor Pressure Vessel (RPV). The helium flow enters the top of the reactor core where it is evenly directed to the 420 borings containing the fuel assemblies as well as between the annuli of the control rod guide-tubes. The helium leak flow also enters the top and bottom RPV domes. The helium from the bottom dome re-joins the leak annulus between the CB and RPV, while the helium entering the upper dome flows past the control rod guides and metallic components and connects to the upper gas plenum and is routed downwards to join the major coolant flow past the fuel assemblies. Some helium from the upper dome is also forced into coolant holes within the control rod guide sleeves which flows in the inner anulus of the control rods, this then exits the sleeve and re-joins the major downward flow in the core. The helium then flows downward through the annulus in the borings past the fuel assemblies to remove heat and exits the core at 750°C. It is then collected in a lower core hot gas plenum that is part of the lower core support structures and flows back through a hot duct (connected to the hot gas plenum) to the HPHE.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The plant is designed to perform its safety functions without reliance on the automated plant control system, or the operator. The plant has no engineered safety systems in terms of active human or machine intervention to assure nuclear safety.

<u>Nuclear stability</u>: If all control and shutdown systems are accidentally withdrawn, it will not lead to fuel damage or a radionuclide release. There is no requirement for active safety systems or operator action to prevent fuel damage. This is achieved with a relatively large negative temperature coefficient of reactivity over the entire operational range, a low core power density, a core geometry that will ensure passive decay heat removal and the radionuclide retention capacity of the TRISO particle fuel as well as the SiC structure of the fuel assembly. Xenon oscillations is damped due to the H/D ratio of the core of less than 3 which is normally used as guideline for inherent stability.

<u>Thermal stability</u>: The low power density is ensured in the core design as well as a high thermal capacity and height to diameter ratio (H/D) greater than 0.97 to ensure that the decay heat removal can solely be achieved through conduction, natural convection, and radiation through the reactor structures. It was determined that increasing the height/diameter ratio beyond 0.97 in a trade-off between neutron losses versus the advantage of gaining passive decay heat removal via the walls of a steel pressure vessel in the event of a DLOFC.

<u>Mechanical stability</u>: The design also ensures that the materials of construction remain below the structural design limits and the maximum fuel temperatures in an accident condition remain below the set fuel damage limits

(b) Decay Heat Removal System / 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 independent cooling trains and is designed for all postulated design basis conditions.

(c) Spent Fuel Cooling Safety Approach / System

The spent fuel will be stored in special tanks of which the height/diameter is in excess of 4. Thermosiphons (heat pipes) are attached to the outside walls of these tanks which removes heat to the outside of the spent fuel area. The condenser ends of these heat pipes are then fitted with fins to dissipate the heat to atmosphere in an entirely passive way. There is also the possibility to use Stirling engine/generators on the condenser ends to generate electricity for charging batteries and in so doing provide a measure of energy conservation.

(d) Containment System

The reactor has two barriers of SiC, that being the SiC layer surrounding each fuel kernel as well as the SiC fuel tube of the fuel assembly. This introduces an additional barrier which will further reduce the probability of any radioactivity being released into the environment. In addition to the two abovementioned barriers, the pressure boundary serves as a third barrier against release of radioactivity, and then as a final barrier the reactor building can also be accredited as a final enclosure.

(e) Chemical Control

Chemical stability: The design of the core and its coolant routing is such that in an event that could allow air to leak into the pressure boundary, there is no possibility that a sustained corrosion of core components by air can take place. The reactor also does away with the possibility of a water or steam ingress scenario as the helium coolant will transfer heat to a HPHE which is a single-phase natural convection heat pipe heat exchanger using LBE as working fluid. This heat exchanger is then coupled to a Brayton power conversion cycle. The use of a HPHE also introduces another important safety feature by eliminating the possibility of tritium, produced in the primary helium cooling circuit to contaminate the air in the secondary circuit by diffusing through a single tube wall.

6. Plant Safety and Operational Performances

The thermal-hydraulic calculations show the maximum fuel temperatures for the calculations was 1101.5°C and the maximum central fuel temperatures determined using a validated computer code being 1084°C (a difference of only ~17.5°C) which is 1.6%. LBE is used as filler material in the fuel assembly due to the fact that it reduces the central fuel temperature by ~130°C compared to a fuel assembly only containing helium as the filler material. The fuel will remain below the normal operational temperature guideline chosen of 1130°C and in the case of a DLOFC event will remain below the Germany set temperature limit of 1600°C.

7. Instrumentation and Control System

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

8. Plant Layout Arrangement

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 (RB) is partially submerged below ground level such that the reactor and heat exchanger cavities are completely protected against postulated external threats. The reactor building, electrical building and auxiliary buildings are connected by means of underground tunnels, providing protection for interlinked services. The RB is seismically designed to withstand a design basis earthquake (DBE) and is the only safety related building structure of the AMR.

9. Testing Conducted for Design Verification and Validation

Experimental activities are being conducted to determine heat transfer enhancement of the Lead Bismuth Eutectic (LBE) in SiC fuel tubes. It is anticipated that the core neutronics will be verified in a future critical facility at the NECSA site. The Helium Test Facility on the NECSA site will be used to test the operational performance of key components under non-nuclear operational conditions of temperature, flow and pressure.

10. Design and Licensing Status

The AMR is at pre-conceptual phase. The neutronics, thermo-hydraulics and heat transfer analyses are being conducted to verify the safety analysis. Mechanical design and material selection are also underway.

11. Fuel Cycle Approach

The AMR is designed for UO_2 and UCO. However, it is not limited to the use these types of fuel. The AMR while maintaining safety characteristics, can use alternate fuels without modifications. Advanced fuel cycles for later investigation may range from a (Th, U)O_2 using both LEU and HEU, to a UC_2 fuel cycle.

12. Waste Management and Disposal Plan

Disposal of spent fuel elements from the AMR is performed 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; (iii) Once cooled down, the flask filled with fuel is transferred from the Hi-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.

13. Development Milestones

2020	Project Started
2021	Pre-conceptual design
2022	To begin with conceptual design

Aurora Powerhouse Product Line (Oklo Inc., United States of America)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Oklo Inc., United States of America
Reactor type	Liquid Metal Fast Reactor
Coolant/moderator	Liquid metal / no moderator
Thermal/electrical capacity, MW(t)/MW(e)	50 – 150 / 15, 50
NSSS Operating Pressure (primary/secondary), MPa	Not pressurized
Fuel type/assembly array	Metal fuel
Refuelling Cycle (months)	120 – 240 months
Design life (years)	40+ years
Plant footprint (m^2)	2 acres
Design status	Actively engaging the U.S. NRC from 2016. Safeguards information protection and handling plan approved, awarded a Site Use Permit by the U.S. DOE as well as fuel material by the Idaho National Lab and the U.S. DOE in 2019. Submitted first-ever combined license application for an advanced fission power plant, and an approved Quality Assurance Program document in 2020.

1. Introduction

Oklo Inc. (NYSE: OKLO) a fast fission clean power technology and nuclear fuel recycling company developing fast fission power plants to provide clean, reliable, and affordable energy at scale.

2. Target Applications

The Oklo powerhouse is designed to provide affordable, reliable, emission-free electricity and heat. Oklo's business model is to provide power on a power purchase agreement basis. Oklo is in active discussion with companies across the following sectors: data centers, defense, industrial and manufacturing, real estate, and oil and gas.

3. Main Design Features

Safety is fundamentally accomplished in Oklo's design by its inherent characteristics, including:

- Low decay heat term, removed by inherent and passive means
- Inherent reactivity feedbacks ensure reactor power is controlled during overpower or overtemperature events
- Multiple barriers to fission product release
- Ambient pressure system removes sources of pressure and minimizes driving forces for release
- Water not required for safety-related cooling
- <18 months installation
- Offsite fabrication

- Operate at atmospheric pressure
- Power output license is 40+ years

4. Initial Proposed Site

Oklo is working with a range of customers related to the initial sites. It is anticipated that the first powerhouse will be sited at the Idaho National Laboratory (INL) Site in southeast Idaho, 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. Additionally, the DOE provisionally approved a specific site within the INL Site for the location of the powerhouse in late 2021.

5. Plant Layout Arrangement

The Aurora site has a primary building with some supporting structures. The Aurora powerhouse has two floors. 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. There is a parking lot for the site as well as landscaping surrounding the site. Due to the inherent safety characteristics, an evacuation zone outside of the site boundary is not generally required.

6. Design and Licensing Status

The design is at the detailed design stage. Oklo has completed fuel prototyping activities, heat transport testing programs, and is now developing experimental advanced fuel recycling and fabrication capabilities. Oklo was awarded a Site Use Permit by the U.S. DOE and was awarded fuel material by the Idaho National Lab and the U.S. DOE. Oklo is the only company to have had a custom combined license accepted for review by the U.S. NRC. Oklo has actively engaged the U.S. NRC from 2016 to the present in pre-application activities, as well as application review activities.

7. Development Milestones

2016	Began pre-application interactions with the U.S. NRC and ongoing pre-application process for Aurora powerhouses
2017	Demonstrated prototype fuel fabrication
2018	Conducted thermal testing
2019	Safeguards Information Protection and Handling Plan approved
2019	Granted a site use permit from the U.S. Department of Energy
2019	Awarded recovered fuel material from the Idaho National Laboratory
2020	Combined license application submittal to the U.S. Nuclear Regulatory Commission and acceptance for review (currently being revised)
2020	Quality Assurance Program Document approved
2021	Began pre-application for fuel recycling
2021	Partnership with Centrus announced
2021	Awarded U.S. Department of Energy Technology Commercialization (TCF) Fund award related to fuel recycling
2016-2022	Awarded five awards from the Gateway for Accelerated Innovation in Nuclear for work on fuel data, fuel modeling, fuel fabrication, thermal hydraulic testing, licensing support, and more
2022	Awarded Advanced Research Projects Agency-Energy (ARPA-E) OPEN program award related to fuel recycling
2022	Awarded ARPA-E ONWARDS program award related to fuel recycling
2023	Began pre-application process for fuel fabrication
2024	Safety Design Strategy for the Aurora Fuel Fabrication Facility approved by DOE
2024	Entered into a land rights agreement with the non-profit Southern Ohio Diversification
2024	Initiative to advance the deployment of two powerhouses in Southern Ohio
2024	In partnership with Argonne National Laboratory, Oklo successfully completed the second phase of the high-fidelity testing campaign at the Thermal Hydraulic Experimental Test Article

2024	Signed non-binding letter of intent to supply 50 megawatts to Diamondback Energy, a
2024	Texas-based independent oil and natural gas company, over a 20-year power purchase
2024	agreement
2024	Began trading on the New York Stock Exchange under the new ticker symbol “OKLO
2024	Signed non-binding memorandum of understanding with Atomic Alchemy to collaborate on isotope production
2024	Partnered with computer networking company Wyoming Hyperscale to deliver 100 megawatts to its data centers over 1 20-year power purchase agreement
2024	Completed successful end-to-end demonstration of advanced fuel recycling process, advancing commercial-scale recycling facility
2024	Established preferred supplier agreement with Siemens Energy for steam turbine generator products and services

ELENA (NRC "Kurchatov Institute", Russian Federation)

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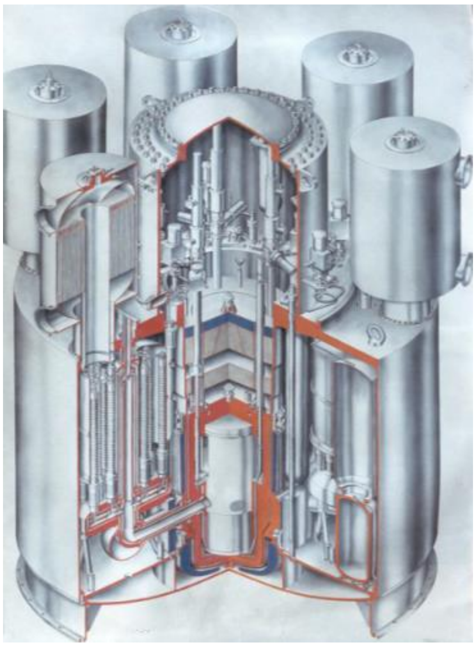


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	National Research Centre 'Kurchatov Institute' (RRC KI), Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	3.3 / 0.068
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	19.6 / 0.36
Core Inlet/Outlet Coolant Temperature (ºC)	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. 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.

4. Main Design Features

(a) 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.

(b) Reactor Core

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

(c) 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.

(d) 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.

(e) 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 100ºC. 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.

(f) 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.

5. 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.

6. 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.

7. Instrumentation and Control Systems

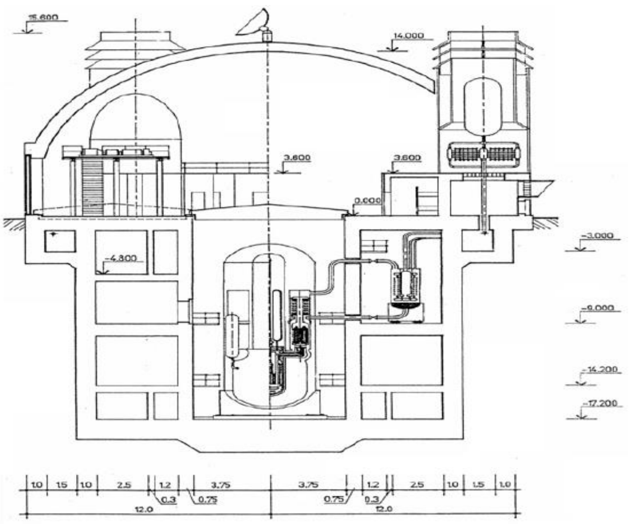
The instrumentation and control (I&C) system of the ELENA reactor is specially arranged to register parameter deviations at early stages of the accidental conditions to predict their further progression.

8. 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.

9. Design and Licensing Status

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

10. 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.

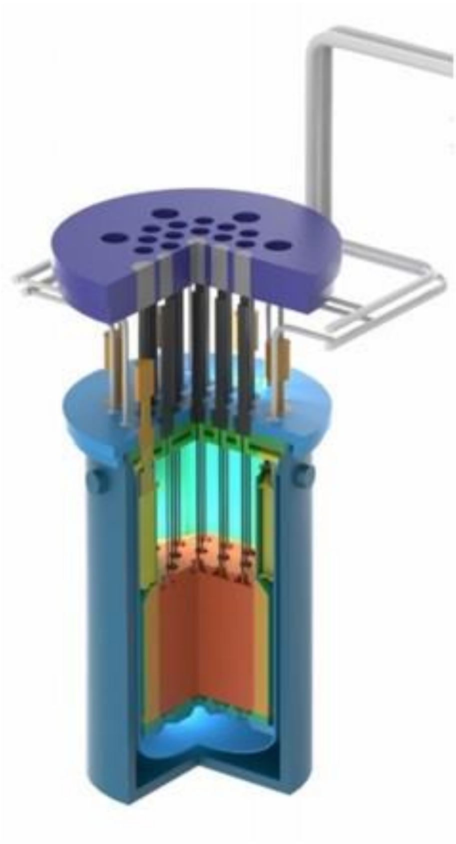
11. Waste Management and Disposal Plan

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.

12. Development Milestones

Not determined.

Energy Well (Centrum výzkumu Řež s.r.o., Czech Republic)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Centrum výzkumu Řež Czech Republic
Reactor type	Fluoride high temperature reactor
Coolant/moderator	Molten salt FLiBe
Thermal/electrical capacity, MW(t)/MW(e)	20 / 8
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	Atmospheric
Core inlet/outlet coolant temperature (°C)	650 / 700
Fuel type/assembly array	TRISO
Number of fuel assemblies in the core	25
Fuel enrichment (%)	15
Refuelling cycle (months)	84
Core discharge burnup (GWd/ton)	70
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)	7 / 3.5
RPV weight (metric ton)	< 100
Seismic design (SSE)	Yes
Fuel cycle requirements/approach	Once through
Distinguishing features	High passive safety features
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; 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 with the large-scale nuclear power reactors, heating plants and the renewable sources of energy.

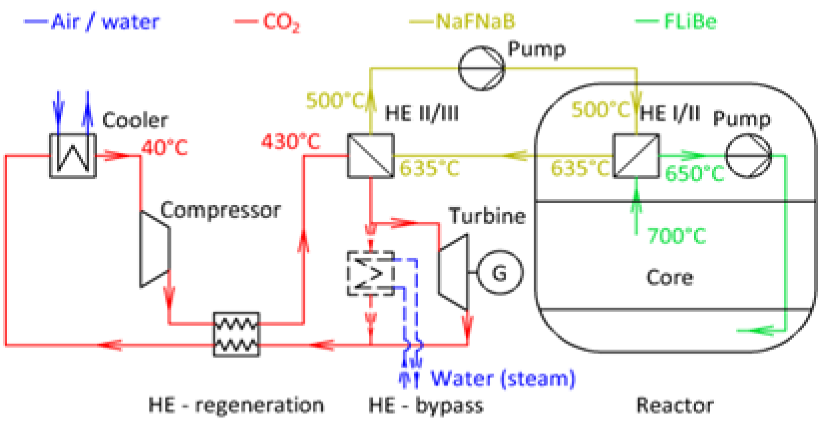
3. Design Philosophy

The 20 MW(t) Energy Well adopts a 7-year fuel cycle, low power density, high use of passive safety and simplicity. The design allows a modular approach and requires minimum on-site welding operations.

4. Main Design Features

(a) Power Conversion

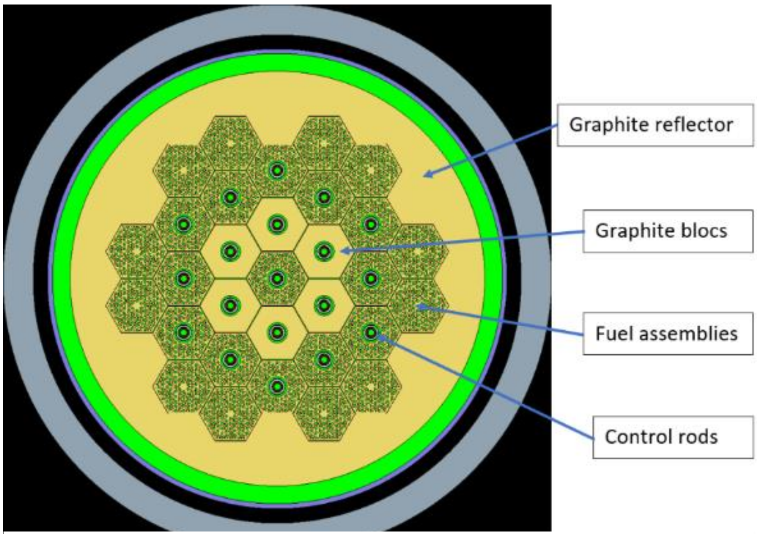
The power plant includes three cooling circuits. Liquid fluoride salts are used as a heat transfer medium (FLiBe, NaBF4) in the primary and secondary circuits. Carbon dioxide in a supercritical state (sCO2) is used in the tertiary circuit. The tertiary circuit considers an Ericsson–Braytonbased cycle 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 highpressure tertiary circuit while ensuring the heat transfer.



Process Flow Diagram of Energy Well reactor with simple heat recovery system

(b) Reactor Core

The reactor core includes, 25 hexagonal fuel assemblies, 6 central graphite blocs, an external graphite reflector and 19 control rods. The core is 2 m high and has a diameter of 3 m with the reflector. The current core arrangement is shown in the figure on the right.



Core Arrangement

(c) Fuel Characteristics

The Energy Well core fuel assemblies are made of TRISO spherical fuel particles that are blended together to form fuel compacts. TRISO coated fuel particles are composed of a uranium kernel coated in successive layers of pyrolytic carbon, silicon carbide, and an outer layer of pyrolytic carbon. The fuel compacts are stacked in a drilled hexagonal graphite block that also contains holes for molten salt coolant flow. A fuel enrichment of 15% has been selected.

(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. A secondary independent shut-down system is currently being studied and shall be implemented in the design.

(e) Reactor Pressure Vessel and Internals

Energy Well is a pool type reactor with molten salt FLiBe as the primary coolant. The primary cooling circuit includes the following main parts: a reactor core with top mounted control rods, a reactor vessel with cover lid, a graphite reflector, a core supporting plate, a flow skirting, six primary heat exchangers molten salt/molten salt and main circulation pumps.

The molten FLiBe flows upward through the core and then, guided by the flow skirting enters the main circulation pumps that circulate the molten salt to the six primary heat exchangers located on the periphery of the reactor vessel above the core. At the outlet of the heat exchangers the molten salt flows in a annular space between the flow skirt and the reactor vessel downward through the core supporting plate.After passing through the core supporting plate, the FLiBe is directed upward to enter the bottom of the core.

(f) Reactor Coolant System

The primary coolant is a molten fluoride salt containing lithium and beryllium (Li_2BeF_4— referred to as “FLiBe”). It has a melting point of 459°C, a boiling point of 1430°C, and a density of 1.94 g/cm3 . The heat capacity of Flibe is 4540 kJ/m3 , which is similar to that of water, more than four times that of sodium, and more than 200 times that of helium (at typical reactor conditions). There is also considerable experience with FLiBe in nuclear systems since it was used in both the primary and secondary loops of the Molten Salt Reactor Experiment (MSRE) and related test loops. The relatively high melting point of FLiBe will require special design features. Experience feedback from sodium-cooled fast reactor, lead cooled submarine reactor and MSRE is being considered during the Energy Well development. The use a pool-type reactor vessel shall reduce some of the challenges related to molten salt freezing in the primary circuit. Molten salts are also transparent which is an advantage during refuelling, maintenance operations and inspection.

(g) Secondary System

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 includes 3 main components: a salt/salt heat exchanger; a salt/sCO_2 heat exchanger; and a circulation pump. The heat exchangers create the interface of circuits, and the pump ensures the required mass flow of secondary salt 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 shall include 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 and FLiNaK with a solidification temperature of 454°C, could be foreseen.

(h) Steam Generator

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 sCO_2 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.

5. 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; low power density; and use of natural circulation and passive safety systems.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

Passive residual heat removal from the primary circuit through the reactor vessel 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) Spent Fuel Cooling Safety Approach / System

The spent fuel cooling system is still to be defined.

(d) 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 containment vessel, the pit (together with the maintenance room shielding ceiling) and the reactor building are additional barriers of the containment system.

(e) Chemical Control

The chemical control of the molten salts in the primary and secondary circuit is still to be defined. Feedback experience from MSRE operation shall be considered.

6. 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 with a refuelling performed on-site every 7 years The refuelling is performed by a fuel handling machine similar to the one used at the Fort Saint Vrain reactor. Fuel handling equipment is designed to be portable and will not be left on site when not needed to prevent access to fissile material.

The main circulation pumps and the primary heat exchangers connecting flanges are bolted to the reactor vessel lid. The limited size of the primary circuit equipment and the location of the primary circuit connecting flanges above the molten salt level shall facilitate the maintenance operations.

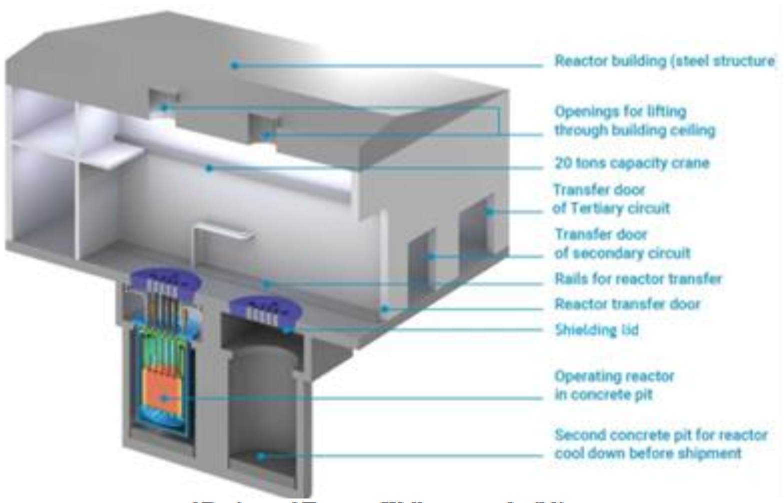
To target markets with a demanding power up to 100/200 MW(t) a multiple unit approach is foreseen.

7. Instrumentation and Control System

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.

8. Plant Layout Arrangement

The Energy Well reactor building layout is shown in the figure on the right. The primary, secondary and tertiary circuit are located in a common steel building. The three circuits are located in separate rooms with separate transfer doors. The reactor vessel with the primary circuit is located in a concrete pit in the reactor room. Additional pits can be added in the reactor room to implement more reactor units and increase the delivered power. The reactor room is equipped with an overhead crane to handle shielding lids, transfer casks and equipment. Control rods actuators and



3D view of Energy Well reactor building main circulation pumps motors are located in a maintenance room above the reactor vessel and below the ground level. This room can be inerted during maintenance operations and refuelling to prevent air ingress in the reactor vessel.

9. Testing Conducted for Design Verification and Validation

Material corrosion test facilities are built and currently in operation in Centrum výzkumu Řež. A forced circulation loop with molten FLiBe is currently being built to verify thermohydraulic parameters.

10. Design and Licensing Status

As of 2022 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.

11. Fuel Cycle Approach

A once-through scheme approach is foreseen for the fuel. Once the 84 months fuel cycle accomplished, the spent fuel is removed from the core and transfer to the on-site spent fuel storage area. Fresh fuel is shipped to the site, inspected and then installed in the reactor vessel with the fuel handling machine.

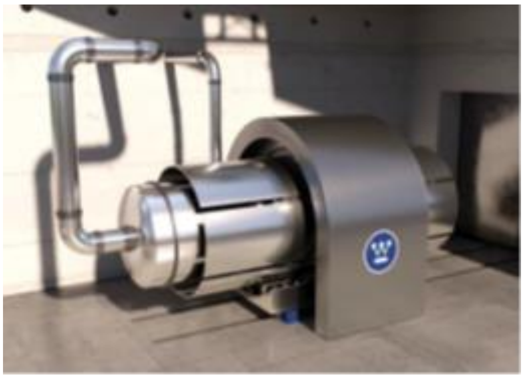
12. Waste Management and Disposal Plan

The waste management and disposal plan are under development.

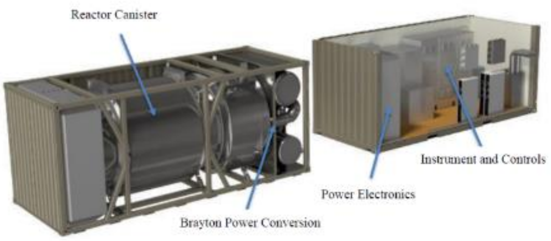
13. Development Milestones

2010 2017	Basic research in the neutronics, thermohydraulics and material compatibility in regard to the FLiBe salt
2017 - 2022	Pre-conceptual design phase and technology validation
2022 - 2027	Basic design
2027 - 2032	Experimental verification using an integral test facility
2032 - 2040	Finalization for manufacturing

Westinghouse eVinci™ Microreactor (Westinghouse Electric Company LLC, USA)



Westinghouse's eVinci microreactor schematic



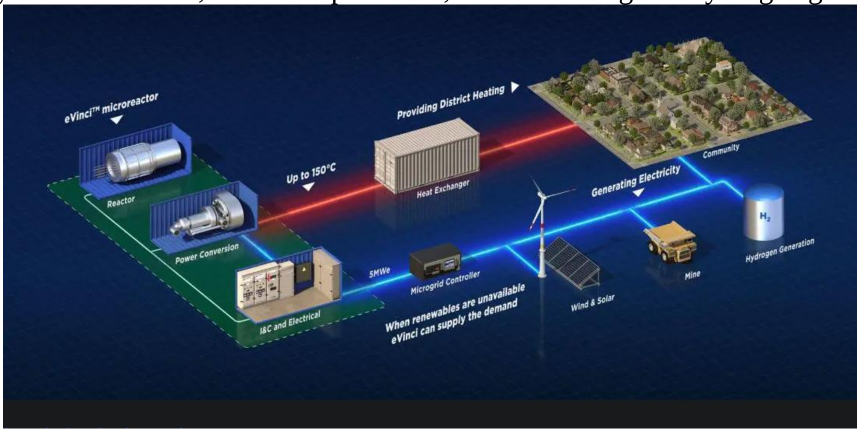
KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse Electric Company LLC, USA.
Reactor type	Micro-modular reactor
Coolant/moderator	Heat pipes (Na) / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	15 / 5
Primary circulation	Natural
NSSS Operating Pressure (primary/secondary), MPa	0.12 (primary)
Core Inlet/Outlet Coolant Temperature (°C)	N/A
Fuel type/assembly array	TRISO in fuel compacts
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	19.75
Core Discharge Burnup (GWd/ton)	N/A
Refuelling Cycle (months)	8 full-power years
Reactivity control	Control drums (Primary) and rods (Defense-in-Depth)
Approach to safety systems	Inherent and passive safety for shutdown and heat removal
Design life (years)	8
Plant footprint (m^2)	< 2 000
RPV height/diameter (m)	N/A
RPV weight (metric ton)	N/A
Seismic Design (SSE)	1g pga
Distinguishing features	Transportable reactor that can operate autonomously
Design status	Detailed Design

1. Introduction

The Westinghouse eVinci™ Microreactor is a 5MWe/15MWth reactor capable of operating for 8 or more years prior to refuelling/replacement and 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. A key attribute of the design is its transportability within standard shipping containers. The design is based on heat-pipe reactor technology which passively transfers heat generated by the nuclear fuel from the core to the Power Conversion System (PCS), which relies on an open-air Brayton cycle to convert heat into electricity, making the reactor appropriate to deploy in a large variety of environments. The reactor core consists of a monolithic, hexagonal-shaped, graphite structure containing a pattern of channels for heat pipes and fuel compacts, containing 19.75% U-235 enriched (HALEU) TRISO fuel. Because of its compact and simple design, the eVinci microreactor will be manufactured and fuelled in a factory, and then transported to an end user site. The figure below shows how the eVinci reactor and the PCS can be packaged into two standard transport containers: one of the containers houses the reactor and the PCS, the other container includes power electronics and the instrumentation & controls (I&C) system.

2. Target Application

The eVinci microreactor 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 decentralized and off-grid markets, with the flexibility and resilience required to provide energy for data centers, industrial processes, district heating and hydrogen generation.



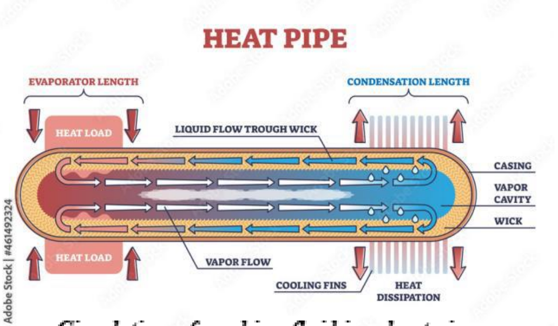
3. Design Philosophy

The design of the eVinci reactor leverages proven heat pipe technology developed by the Los Alamos National Laboratory (LANL) for space application. This uranium-fuelled 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 fostering overall plant simplicity. The design utilizes the inherent safety features in the fuel, moderator, and heat pipes to enhance safety and self-regulation capability.

4. Main Design Features

(a) Nuclear Steam Supply System

The eVinci microreactor is a micro-modular reactor that utilizes passive heat pipes to circulate thermal energy from the reactor vessel to the working fluid in the Power Conversion System (PCS). The heat pipes, which are fully enclosed, contain sodium that uses capillary forces and pressure differences to naturally circulate the heat from one end of the pipe to the other. These components foster high reliability and operational simplicity compared, for instance, to forced circulation of a high-pressure coolant which may be featured in other micro reactor designs.



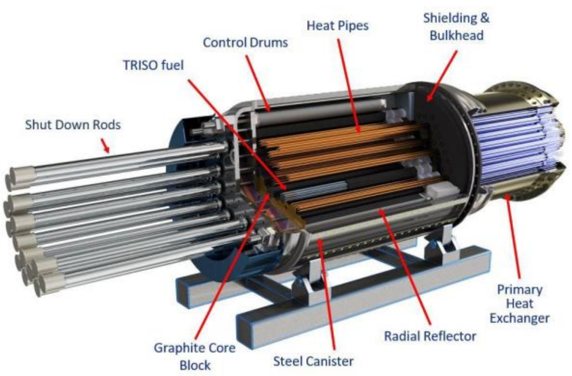
Circulation of working fluid in a heat pipe (source: bajeczneobrazy.pl)

(b) Reactor Core

The reactor core consists of a monolithic, hexagonal-shaped, graphite structure containing a pattern of channels for heat pipes and fuel compacts, containing 19.75% U-235 enriched (HALEU) TRISO fuel.

(c) Reactivity Control

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.



Sectional cut of the eVinci microreactor displaying the key technological components

(d) Reactor Coolant System

The reactor heat transport design leverages the passive heat pipe thermal transfer capabilities of sodium heat pipes, which are located in channels in the core and under normal operation are used to move heat to the heat exchanger and associated PCS using capillary forces and pressure differences that naturally circulate the heat from one end of the pipe to the other. These components foster high reliability and operational simplicity removing the need for coolant pumps in the primary system compared, for instance, to forced circulation of a high-pressure coolant which may be featured in other microreactor designs.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Resiliency of the eVinci microreactor is enabled through packaging within a secure canister, installed in a strong secure structure on site and connected to a micro-grid coordinating both heat and power. A dedicated Micro Grid Interface System, interfaces with the Brayton controller and reactor control system allowing for automated control logic for load following capability, make it ideal to operate together with other energy resources. The integration capability of the eVinci microreactor 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.

(b) Containment System

The reactor containment system is composed of a canister containment subsystem (CCS) and a secure vault subsystem (SVS). The CCS encases the entire core of the microreactor and features a robust design to safely contain and manage the nuclear fuel and fission products within the reactor system. Its structural components will prevent the release of radioactive materials and maintain a safe operating environment within the reactor vessel. The CCS is further fortified by the SVS, which provides multiple layers of protection against external hazards to maintain the integrity of the reactor during accident scenarios.

(c) Spent Fuel Cooling Safety Approach / System

The fuel handling system was designed so that all related activities will be performed off-site at licensed facilities. The spent fuel will be stored at these licensed facilities in casks until a Deep Geological Repository is constructed.

6. Plant Safety and Operational Performances

The eVinci microreactor 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 there are no safety related operator actions and minimal daily activities to perform. 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.

7. Instrumentation and Control Systems

The Autonomous Control System (ACS) is the primary I&C system of the eVinci microreactor. The ACS facilitates autonomous control by utilizing fiber optic sensors for high fidelity temperature measurement of the reactor, neutron flux sensors for monitoring 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 priority between trip, load following and manual system controls. 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.

8. Plant Layout Arrangement

The eVinci microreactor 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 connections 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.

9. Testing Conducted for Design Verification and Validation

The conceptual design for the eVinci microreactor was completed in 2022 and led to Westinghouse initiating licensing engagements with regulators in the U.S. and Canada. In 2023, Westinghouse started procurement for the Nuclear Test Reactor (NTR) design and began prototyping various components of the reactor for testing. Procurement of the NTR was completed for fabrication of its components to begin in 2024, along with criticality, transient, and irradiation testing. The eVinci design for manufacturing was also finalized. In the next few years, Westinghouse plans to test the NTR at the Idaho National Laboratory DOME facility, continue testing various eVinci systems, and initiate manufacturing to prepare for regulatory licensing approvals and commercialization.

10. Design and Licensing Status

The conceptual design for the eVinci microreactor was completed in 2022 and led to Westinghouse initiating licensing engagements with regulators in the U.S. and Canada. In 2023, Westinghouse started procurement for the Nuclear Test Reactor (NTR) design and began prototyping various components of the reactor for testing. Procurement of the NTR was completed for fabrication of its components to begin in 2024, along with criticality, transient, and irradiation testing. The eVinci design for manufacturing was also finalized. In the next few years, Westinghouse plans to test the NTR at the Idaho National Laboratory DOME (Demonstration Of Microreactor Experiments) facility, continue testing various eVinci systems, and initiate manufacturing to prepare for regulatory licensing approvals and commercialization. Westinghouse can leverage its broad, successful experience in licensing new nuclear technologies including the globally operating AP1000 PWR, to ensure global compliance of the eVinci microreactor. This is supported by incorporation of safety standards established by international organizations, such as the International Atomic Energy Agency (IAEA), into the design and performance of comprehensive technical reviews of the reactor with regulatory experts. Westinghouse has also accounted for adaptability to local requirements in different regions in the reactor’s design.

11. Fuel Cycle Approach

The eVinci microreactor uses 19.75% U-235 enriched UCO TRISO fuel arranged into fuel compacts that are loaded into the reactor’s core and can sustain 8+ years of full power operation without refuelling. Refuelling does not take place on the customer site; instead, at the end of the fuel cycle, the spent reactor core will be transported to a licensed facility in its original canister where refuelling and spent fuel storage will occur, with newly fuelled reactor cores replacing spent fuel reactor cores.

12. Development Milestones

2021	Electric demonstration
2027	First nuclear demonstration
2029	Commercial deployment

HOLOS-QUAD (HolosGen LLC, United States of America)

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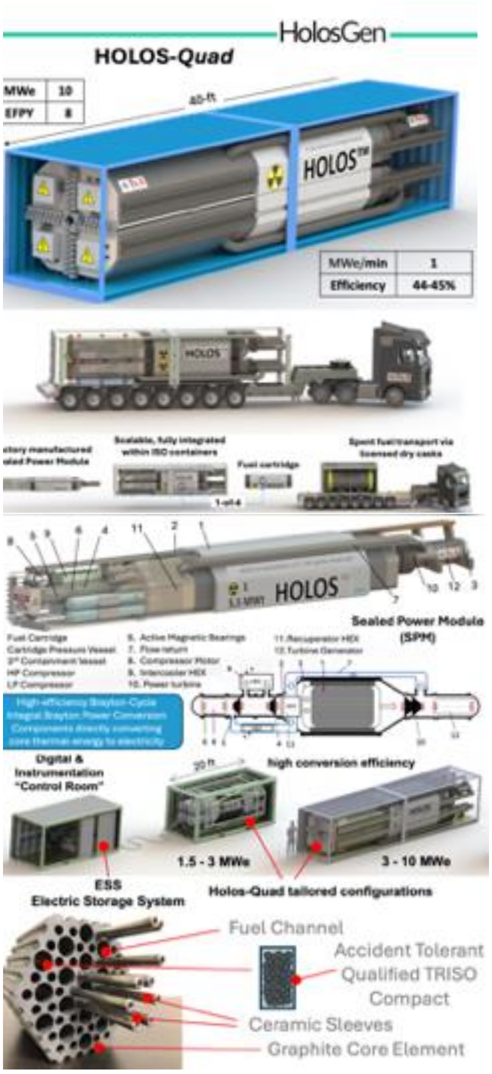


RISO: TRistructural ISOtropic, UCO: Uranium OxyCarbide. 'We: Mega-Watt-electric, ISFSI: Independent Spent Fuel orage Installation, SPM: Sealed Power Module, EFPY: quivalent Full Power Year. TRL: Technology Readiness Level, SS: Energy Storage System, BoP: Balance of Plant, SSC: ystems, Structures and Components. UHS: Ultimate Heat Sink, EX: Heat Exchanger, I-HEX: Intermediary-Heat Exchanger, MB: Active Magnetic Bearings.

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	HolosGen LLC, United States of America
Reactor type	High Temperature Gas Reactor
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	22 /10
Primary circulation	Helium
NSSS Operating Pressure (primary/secondary), MPa	No NSSS. Brayton Power Cycle Max 7MPa, Min 3.5MPa
Core Inlet/Outlet Coolant Temperature (°C)	590 / 855
Fuel type/assembly array	Qualified TRISO-UCO Compacts
Number of fuel assemblies in the core	Multiple layers of hexagonal graphite-fuel elements (number of moderator-fuel elements proportional to power rating and EFPY)
Fuel enrichment (%)	19.95
Core Discharge Burnup (GWd/ton)	61
Refueling Cycle (months)	96 at EFPY operating conditions
Reactivity control	Redundant independent control drums inside the fuel cartridge and redundant shutdown rods outside of the pressure vessel boundary.
Approach to safety systems	Passive decay heat removal
Design life (years)	40 to 60
Plant footprint (m^2)	82 (reactor comprised in ISO container + external radiation shields)
RPV height/diameter (m)	9.5/2.3 (4 coupled SPMs)
RPV weight (metric ton)	40 (mobile operational power plant)
Seismic Design (SSE)	4g max acceleration (rail transport shocks limit)
Distinguishing features	Integral, transportable, rapidly relocatable, high efficiency, automatic, load-follow, high-temperature process heat with independent working fluid, net plant efficiency >40% for power rating above 30% of nominal power, ship propulsion application, 3rd party validated TRL5.
Design status	Detailed Design

1. Introduction

The HOLOS-*Quad* microreactor is a high-performance gas-cooled reactor designed for transport, featuring four Sealed Power Modules (SPMs) that convert thermal energy into electricity. Each SPM uses a closed-loop full Brayton gas thermodynamic cycle, with an efficiency of 42% for air-cooling and 45% for water-cooling. The system has a load-following rate of 10% of the reactor nominal power per minute and is offered in different configurations to meet specific requirements and power rating.

2. Target Application

The HOLOS-*Quad* design is a high-temperature process heat generator suitable for non-carbon fuel production (e.g., hydrogen), steelmaking and other high-temperature industrial applications. It has load-following capability, making it suitable for distribution, powering microgrids and stand-alone electric island. The scalable design can also supply emergency power generation for disaster areas, and large commercial reactors. Scaled-down versions can supply power on demand for space propulsion, deep-space, lunar surface and Mars missions. Traditional applications include providing carbon-free power to remote communities and terrestrial or offshore mining operations currently relying on fossil fuel burning.

3. Design Philosophy

The underlying design philosophy for all HOLOS design configurations relies on unique design features that enhance safety by eliminating or reducing the Balance of Plant (resulting in lowered Loss Of Coolant Accident probability) and reduced power plant footprint. Through high-rate and high-efficiency load-follow capability the deign operates with higher fuel utilization, which directly reduces the volume of spent fuel and thermal pollution per unit of electrical energy generated. From an economic standpoint the HOLOS design philosophy results in substantial reductions of the costs associated with: i) capital construction; ii) operations & maintenance; iii) spent fuel transport and storage; and iv) deactivation and decommissioning activities.

4. Main Design Features

(a) Nuclear Steam Supply System

The HOLOS-*Quad* design configuration integrates the components forming a Brayton cycle with intercooler and recuperator heat exchangers altogether with the fuel cartridge equipping each of the four coupled SPMs. This design architecture approach eliminates the traditional Balance of Plant, reduces the working fluid inventory and the plant footprint, while enabling high-efficiency electricity supply satisfying highly variable power profiles. The design may be equipped with an intermediate heat exchanger to supply thermally coupled high-temperature process heat working fluids to support industrial applications.

(b) Reactor Core

The HOLOS-*Quad* configuration of SPM contains ¼ of the core, which is mechanically coupled to form a pseudo cylindrical core geometry. The core is equipped with four fuel cartridges loaded with accident tolerant qualified TRISO fuel and burnable poison embedded within hexagonal graphite elements, ensuring optimized burnout for approximately 8 Effective Full Power Years (EFPY). The core design is safety compliant and allows safe transport during thermally active, and spent SPMs. The U.S. Argonne National Laboratory conducted and published analyses on the HOLOS-*Quad* SPM core performance.

(c) Reactivity Control

The reactivity controls for the HOLOS-*Quad* design is achieved through independent, diversified and redundant reactivity control mechanisms based on banks of control drums and shutdown rods with assured negative reactivity insertion even under failure of a number of control drum and sbutdown rod banks.

(d) Reactor Pressure Vessel and Internals

The HOLOS-Quad design utilizes four compact reactor pressure vessels forming independent SPMs that replace the traditional Reactor Pressure Vessel (RPV). Each SPM is designed with easy to manufacture power conversion components, integrated altogether with the reactor core. Silicon Carbide (SiC) sleeves represent the pressure boundary within the core, allowing high-pressure circulation of helium coolant at high temperature.

(e) Reactor Coolant System and Steam Generator

Core and auxiliaries are cooled by helium thermally coupled with the Ultimate Heat Sink (UHS) through the Brayton intercooler and cooler heat exchangers. The UHS may be represented by environmental air, or environmental water assumed at 30°C. For space applications the UHS is represented by radiators for radiative heat transfer. The HOLOS-*Quad* design does not utilize steam generators; however, steam can be generated as part of the process heat secondary system for selected applications.

(f) Primary pumps
Each of the four SPMs is equipped with a High-Pressure and Low-Pressure axial compressor forming turbomachinery rotors. Each compressor is driven by a dedicated direct-drive (no gear box) variable speed motor. There is no metal-to-metal or metal-to-ceramic contact between rotary and stationary components and no need for a lubrication system as the design is equipped with redundant active magnetic bearings (AMBs).

5. Safety Features

(a) Engineered Safety System Approach and Configuration
The HOLOS-*Quad* reactor design, featuring a low power density core and passive cooling, is designed to ensure safety under normal and off-normal operating conditions. The sleeves, in each SPM, prevent radionuclide migration from potentially damaged TRISO fuel elements and eliminate the need for helium clean-up system. The Argonne National Laboratory conducted safety and technical performance analyses and the results of the inherent safety of the HOLOS-*Quad* thermal-hydraulic design are documented in multiple technical reports, such as the "Preliminary Thermal-Hydraulics and Selected Safety Analysis for HOLOS-QUAD Reactor Design."

(b) Safety Approach and Configuration to Manage DBC
The core is represented by a relatively large mass of graphite with respect to the total amount of fuel, with a relatively low power density. As a result, the fuel does not experience substantial temperature excursions under total loss of coolant circulation and total loss of coolant scenarios. As part of the design validation activities the results of thermal-hydraulic and heat transfer performance, as well as the effectiveness of passive decay heat removal have been independently peer-reviewed and published by the Argonne National Laboratory.

(c) Safety Approach and Configuration to Manage DEC
The HOLOS-*Quad* design underwent Design Basis Accident and Beyond Design Basis Accident simulations validated by physical data obtained through testing of the Sealed Power Module-Subscale Simulator (SPM-SS) equipped with a graphite core and electrical heaters replacing and mimicking the behavior of TRISO fuel. The results validate the inherent safety of the design under postulated DEC as documented by technical reports published by the Argonne National Laboratory and the University of Michigan Department of Nuclear Engineering.

(d) Containment System
The HOLOS-*Quad* design utilizes accident tolerant TRISO fuel with functional containment trapping radionuclides, eliminating the need for sealed structures. However, additional engineered pressure boundaries are included in all HOLOS design configurations to enhance safety. These boundaries prevent radionuclides from leaking from potentially damaged or defective microspheres and trap air surrounding the SPMs.

(e) Spent Fuel Cooling Safety Approach / System
The HOLOS-*Quad* design allows for the "hot" removal of individual SPMs by mechanically separating and transferring them into shipping containers with shielded casks. If the SPMs are removed before reaching "spent" conditions, they can be repositioned and reassembled into the engineered container, allowing load-follow power generation until they become spent. The irradiated SPMs can be removed and reassembled, or permanently stored within 72 hours of their last shutdown. This is an innovative and unique feature of the HOLOS-*Quad* design.

6. Plant Safety and Operational Performances
Plant safety is ensured by the low core power density, the passive removal of decay heat, the inherent TRISO fuel tolerance to high temperature operations and excursions, and the presence of burnable poison within the graphite-TRISO matrix to ensure that core reactivity remains below safety margins even under water flooding events. As the design is configured to perform load-following power generation and can be seamlessly coupled to a "black-start & ballast" battery bank, or fly-wheel banks as Energy Storage Systems (ESSs), the operational performance is effectively real-time automatic. This approach relaxes the requirements associated with monitoring conducted by operation personnel and further reduces the Operation & Maintenance costs. Analyses conducted by the Argonne National

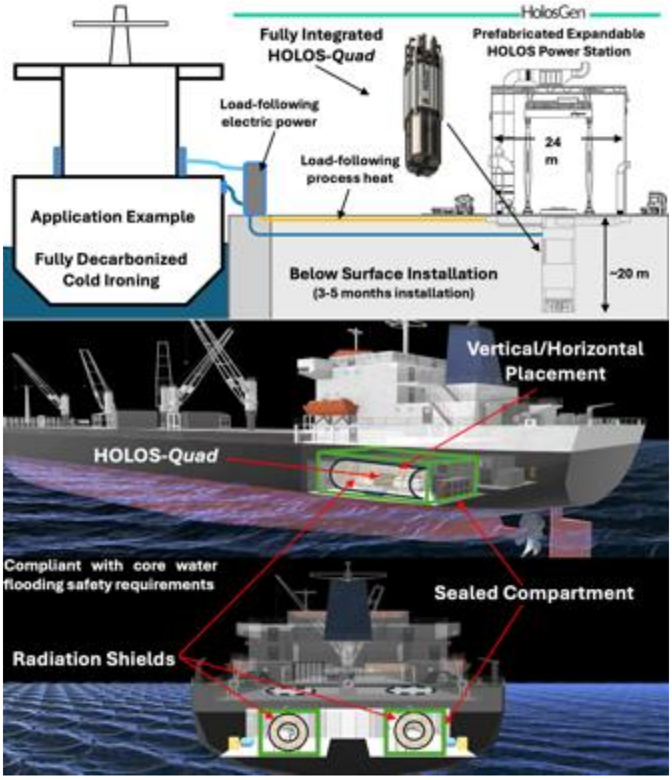
Laboratory indicated that the net power plant efficiency remains higher than 40% for power maneuvering between 30% and 100% of nominal power. Maintaining high net plant efficiency through such a large power spectrum is another innovative and unique feature of the HOLOS designs. High efficiency operations result in a proportionally higher fuel utilization, which translate into additional sellable electricity, and a lowered volume of spent fuel per unit of electrical or process heat energy generated.

7. Instrumentation and Control Systems

All HOLOS design configurations are equipped with integral power conversion components designed to follow the electrical load automatically and at high rate. The control system regulates the power flows between the thermal-to-electric conversion components, the black-start & ballast battery or flywheels (forming the ESS), the electrical loads and the core reactivity rate. The instrumentation and control systems further manage the active magnetic bearings, control drum and shutdown rod drives. During load following maneuvering the control system negotiates the compressor speed and the inventory control system which enables fast-acting power maneuvering. For a 10MWe unit, the digital instrumentation and control equipment, altogether with the ESS, occupies the volume equivalent to a 20-foot "digital I&C" ISO container, electrically coupled to the 40-ft reactor container via shielded, Electro Magnetic Interference- and Electro-Magnetic Pulse-shunted data and power cables. Note that the HOLOS-*Quad* does not require connection to the power grid to operate.

8. Plant Layout Arrangement

The HOLOS-*Quad* is a highly integrated, transportable nuclear power plant that can be positioned horizontally within a prefabricated concrete "shield building", or vertically within a low-cost borehole. The active magnetic bearings enable horizontal and vertical positioning of the SPMs. Each borehole is sealed at the top through a removable shielding cap. In the subsurface configuration, a first reactor unit is housed within a prefabricated building assembled with components transported within shipping containers. This design expedites installation and de-installation activities, reducing capital and decommissioning cost at the end of the licensed life of the "HOLOS Power Station". The prefabricated building can be expanded to house multiple units with reduced upfront cost and may include an Independent Spent Fuel Storage Installation (ISFSI). The design eliminates the Balance of Plant, increases thermal-to-electric conversion efficiency with a reduced power plant footprint per unit of electric energy generated. This configuration is ideal for non-invasive installation on commercial ships and supports decarbonized "cold ironing".



9. Testing Conducted for Design Verification and Validation

The HOLOS-*Quad* design has been validated by scientists and subject matter experts at the U.S. national laboratories and academic institutions through comparisons of high-fidelity computer model projections, sponsored in part by the U.S. DOE ARPA-E MEITNER funding program, with test data obtained through a non-nuclear Sealed Power Module-Subscale Simulator (SPM-SS). An electrically heated surrogate core equipping the SPM-SS was operated to mimic TRISO fuel compacts and thermal-hydraulic performance under normal operations, off-normal and accident conditions. A series of peer-reviewed technical reports, published by the Argonne National Laboratory and by the Nuclear

Engineering Department at the University of Michigan, document and validate the HOLOS-*Quad* tecnnical and safety performance.

10. Design and Licensing Status

A preliminary safety assessment report (PSAR) for a 30MWe HOLOS Power Station equipped with multiple units was completed in 2023. The PSAR included risk assessments during installation and removal and factored onsite and offsite maintenance, refurbishing and refueling operations, power flow and parallel connection with the bulk power grid and ESSs. A risk assessment report for multiple unit installations configured for stationary and for ship propulsion applications with tailored specific power profiles will be completed in 2024. PSARs will be submitted for review by the U.S. Nuclear Regulatory Commission (NRC) pending client’s identification of final power requirements.

11. Fuel Cycle Approach

The 10MWe HOLOS-*Quad* design is currently equipped with a core optimized for approximately 8 Equivalent Full Power Years (EFPY). Spent fuel elements, housed and sealed within each SPM, passively dissipate decay heat and can be placed within dual purpose transport and storage shielded dry casks with the ability to initiate transport 72 hours from the unit last shutdown. The dry casks with spent fuel, or the fuel cartridge separated from the integral power conversion system surrounded by shielding materials, comply with the dimensional constraints of ISO shipping containers. Spent fuel can also be stored at on-site Independent Spent Fuel Storage Installations (ISFSI) with transport to offsite ISFSIs at the end of the Power Station life. This approach reduces shielding requirements and costs associated with transport and storage. The currently adopted fuel cycle is “once through”.

12. Waste Management and Disposal Plan

The great majority of power conversion components and their ancillary components are not exposed to direct neutron irradiation, thus requiring simplified decommissioning procedures. Almost all of the SSCs forming an individual SPM can be classified as low-level radioactive and non-radioactive waste, ready for refurbishment or disposal. Costs associated with spent fuel handling, transport, and storage to onsite or offsite ISFSI are included in the HOLOS-*Quad* costing analysis and factor the reduced mass of spent fuel produced over 40-60 years of licensed life because of high-efficiency thermal-to-electric conversion.

13. Development Milestones

2015 – 2017	Conceptual design developed by Filippone & Associates LLC (F&A)	Complete
2017 – 2018	HolosGen LLC developed and successfully tested the Holos-Test-Assembly-1 (HTA-1) subscale prototype equipped with a surrogate core.	Complete
2018 – 2019	HolosGen awarded by the U.S. DOE ARPA-e MEITNER funding program to demonstrate safety and techno-economic performance.	Complete
2019 – 2022	Completed construction of the Sealed Power Module subscale simulator (SPM-SS) with helium as working fluid at 7 MPa pressure for the full-scale SPM. Completed design validation demonstrating: 45% net plant efficiency; fully fitted within a single 40-foot ISO shipping container; 8 EFPY, passive decay heat removal; load-following capability at a rate of 1MWe/min and safety margins with core water flooded. Key design aspects validated via testing.	Complete
2022 – 2023	Completed the Preliminary Safety Analysis Report (PSAR) for specific use cases involving an expandable Holos Power Station with multiple 10MWe units with undersurface installation at a U.S. federal facility.	Complete
2023 – 2024	Developed detailed step-by-step installation procedure for a 30MWe Holos Power Station equipped with a prefabricated removable building with reduced footprint and deactivation and decommissioning costs.	Complete
2024 – 2027	Complete digital twin and electrical full-scale twin of the *HOLOS-Quad* configuration.	Planned

2027 – 2028	FOAK nuclear testing at U.S. DOE or alternate facility.	
2029	Conversion of the FOAK demonstrator unit into first commercial unit.	Planned

HOLOS-MONO (HolosGen LLC, United States of America)

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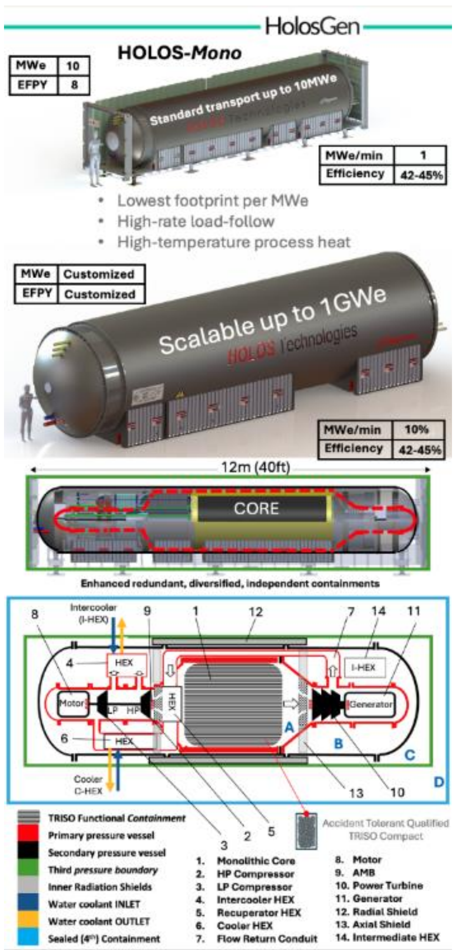


TRISO: TRistructural ISOtropic, UCO: Uranium OxyCarbide. MWe: Mega-Watt-electric, ISFSI: Independent Spent Fuel Storage Installation, SPM: Sealed Power Module, EFPY: Equivalent Full Power Year. TRL: Technology Readiness Level, ESS: Energy Storage System, BoP: Balance of Plant, SSC: Systems, Structures and Components, UHS: Ultimate Heat Sink, I-HEX: Intermediary Heat Exchanger, HEX: Heat Exchanger, AMB: Active Magnetic Bearings.

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	HolosGen, USA
Reactor type	High-Temperature Gas Reactor (HTGR)
Coolant/moderator	Helium/Graphite
Thermal/electrical capacity, MW(t)/MW(e)	22/10
Primary circulation	Helium
NSSS Operating Pressure (primary/secondary), MPa	No NSSS. Brayton power cycle: Max 7, Min 3.5
Core Inlet/Outlet Coolant Temperature (°C)	590/855
Fuel type/assembly array	Qualified TRISO UCO compacts
Number of fuel assemblies in the core	Monolithic graphite-fuel matrix
Fuel enrichment (%)	19.95
Core Discharge Burnup (GWd/ton)	82
Refueling Cycle (months)	96 at EFPY operating conditions
Reactivity control	Drums and shutdown rods
Approach to safety systems	Passive decay heat removal
Design life (years)	40 to 60
Plant footprint (m^2)	5000 (stationary 10MWe underground version) 7000 (30MWe & ISFSI)
RPV height/diameter (m)	9.5/2.3 (individual SPM)
RPV weight (metric ton)	60 all included operational
Seismic Design (SSE)	4g (rail transport shocks limit)
Distinguishing features	Transportable, scalable, integral power conversion system, high-efficiency and high-rate load-follow. High-temperature process heat with decoupled working fluid. Seamless interface with external commercial power conversion components. 3rd party validated TRL6
Design status	Detailed Design

1. Introduction

The HOLOS-*Mono* design is a scalable reactor that offers high-efficiency electricity and high-temperature process heat supply, ensuring compliance with customer and application-specific power profile requirements. It uses a single, scaled-up, SPM housing a traditional monolithic core, reducing capital cost and deployment time. The design optimizes the High Temperature Gas Reactor (HTGR) typology utilizing qualified TRistructural ISOtropic (TRISO) fuel, considered an Accident Tolerant Fuel (ATF). TRISO fuel has high radionuclide retention capability and does not require further containment. The HOLOS-*Mono* design integrates a Chemical and Volume Control System (CVCS) with the helium inventory control system, enhancing containment functions. For power rating up to 10MWe the HOLOS-*Mono* can be transported as a complete reactor system by standard shipping containers, while special transport is required for versions of the design with higher power rating comparable to large and Small Modular Reactor (SMR) designs (e.g., from 45MWe to 1GWe).

2. Target Application

The HOLOS-*Mono* features high-efficiency load-follow electricity maximizing fuel utilization and reducing spent fuel production per unit of electric energy generated, and high-temperature process heat to support non-carbon fuel production, steelmaking and other energy intense industrial applications. The HOLOS-*Mono* system/sub-system model and non-nuclear prototype has been demonstrated in an operational environment qualifying it to TRL6 (Technology Readiness Level as per NASA definition). This represents a major step forward as it retires the high technological risks associated with new and innovative technologies. The design features of scalability, customization and capability to support variable electrical loads make it ideal to support a large variety of applications including, for example, power generation for data centers, ship propulsion for commercial shipping and offshore power production, and on-demand power compensating for the intermittency of renewable energy sources. The design can also be configured to supply high-temperature process heat to decarbonize energy-intensive industrial processes.

3. Design Philosophy

Safety is paramount under all operating, shutdown, transport and storage modes. By design the fuel cannot experience challenging temperatures under both design basis and beyond design basis accident scenarios. Passively cooling provides heat removal under loss of coolant and total loss of electricity. The HOLOS-*Mono* core operates at a relatively low power density with the graphite moderator acting as a "thermal flywheel" that prevents substantial temperature excursions under all scenarios. For example, under worst case scenarios the fuel maximum temperature briefly reaches approximately 1350°C, well below TRISO fuel melting temperature. The ability of the design to supply variable power with high-efficiency and minimum loss in efficiency lowers the volume of spent fuel per unit of electric energy generated, thus reducing the frequency of spent fuel transport as well as risks and costs associated with these activities.

4. Main Design Features

(a) Nuclear Steam Supply System

HOLOS-*Mono* integrates the components forming a Brayton cycle with intercooler and recuperator heat exchangers altogether with the monolithic core, thus eliminating the need for the traditional Balance of Plant, reducing the working fluid inventory and plant footprint. A motor-driven compressor, mechanically decoupled from the expander, enables unique power control features and eliminates the gearbox normally coupling the compressor to the expander turbomachinery. By enabling independent rotary speed control of the compressor and the expander rotors, the design results in high conversion efficiency and high-rate load-follow power regulation. The design supports electricity and process heat supply through thermal coupling of the recuperator heat exchanger with an intermediate heat exchanger.

(b) Reactor Core

HOLOS-*Mono* core is helium cooled, graphite moderated and fuelled with commercially qualified accident tolerant TRISO fuel. The core is further equipped with burnable poison optimizing the core burnout to provide for approximately 8 Effective Full Power Years (EFPY) of operation and ensuring shutdown margins under core water flooding event. Key safety and technical performance parameters have been test-validated.

(c) Reactivity Control

Reactivity is controlled by independent, diversified and redundant reactivity control mechanisms based on banks of control drums and shutdown rods with assured negative reactivity insertion even under failure of a number of control drum and shutdown rod banks and under water flooding of the core.

(d) Reactor Pressure Vessel and Internals

HOLOS-*Mono* is equipped with a non-traditional simpler to manufacture primary pressure vessel with relatively small components forming a scaled-up Sealed Power Module (SPM) comprising the power conversion system and the core. The motor-driven compressor turbomachinery and the turbine-driven generator are housed witihn portions of the primary pressure vessel designed to normally operate at a maximum pressure of 7MPa with the majority of the vessel operating at 3.5MPa. The control drums and shutdown rods are also housed within the primary pressure vessel.

(e) Reactor Coolant System and Steam Generator

Core and auxiliaries are cooled by helium thermally coupled to the Ultimate Heat Sink (UHS) by the cooler and intercooler heat exchangers. The UHS is represented by open- or closed-loop water, or environmental air assumed at 30°C for efficiency calculations. When water-steam is selected as the process heat working fluid, steam can be generated through the intermediate heat exchanger wherein the process heat working fluid is thermally coupled to the helium circulating through the power conversion system. During operations cooling is active (as a result of the compressor operations), while decay-heat removal occurs passively.

(f) Primary Pumps

The scaled up SPM housing the core is equipped with a helium compressor formed by High-Pressure and Low-Pressure axial turbomachinery rotors, driven by a direct-drive (no gear box) variable speed motor. There is no metal-to-metal or metal-to-ceramic contact between rotary and stationary components and no lubrication system is required as the design is equipped with redundant active magnetic bearings (AMBs).

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The design is equipped with the traditional HTGR safety features with enhanced safety through engineered multiple containments capable of storing the whole working fluid inventory. Decay heat is removed passively with multiple natural heat transfer mechanisms and the fuel maximum temperature remains within safety margins under Loss Of Coolant Accident and Station Black Out scenarios.

(b) Safety Approach and Configuration to Manage DBC

The core is represented by a relatively large mass of graphite with respect to the total amount of fuel. As a result the fuel does not experience significant temperature transients under total loss of coolant circulation and total loss of coolant scenarios with decay heat removed by natural heat transfer mechanisms with the air surrounding the SPM.

(c) Safety Approach and Configuration to Manage DEC

The design underwent Design Basis Accident and Beyond Design Basis Accident simulations validated by physical data obtained through testing of the Sealed Power Module-Subscale Simulator (SPM-SS), a reactor simulator equipped with a graphite core and electrical heaters replacing TRISO fuel. The results validate the inherent safety of the design under postulated DEC as documented by numerous techncial publications developed by the Argonne National Laboratory and the Nuclear Engineering Department at the University of Michigan.

(d) Containment System

Although qualified TRISO fuel does not require the classic large volume reinforced concrete containment mandatory for light water reactors, the equivalent primary pressure vessel within the SPM is surrounded by a secondary pressure vessel which operates at a low vacuum to capture potential helium leaks from the primary pressure vessel. As helium is collected within the volume of the second pressure vessel it is compressed back into the primary pressure vessel. A third pressure boundary formed by the volume of the shipping container is designed to capture potential helium leaking from the second pressure vessel. A final fourth containment, formed by the "reactor cavity" for subsurface configurations or "reactor compartment" for above surface, ship or off-shore configurations, ensures that potentially leaked helium and potentially activated Argon gas contained in the air surrounding the reactor remains captured until the activated Argon naturally decays a few hours later and the potentially contaminated helium is treated prior to opening the reactor cavity or compartment (e.g., for refueling, maintenance operation or for deactivation and decommissioning activities.

(e) Spent Fuel Cooling Safety Approach / System

For HOLOS-*Mono* units rated at 10MWe with monolithiv cores lasting 8 years at full power, the whole unit is left within its cavity or compartment for a cooling period prior to transporting it to a specialized refueling, refurbishing or storing facility. Alternatively, the unit can be opened to conduct onsite refueling, refurbishing and maintenance activities. For these units and units with higher power rating the reactor cavity features remotely controlled equipment to robotically conduct these activities with a closed-loop passive air circulation system continuously supporting passive decay heat removal. The reactor cavity remains sealed during these periodic activities and can act as an Independent Spent Fuel Storage Installation (ISFSI) as per U.S. Department of Energy definition. Spent fuel is ultimately removed from the primary pressure vessel and inserted within transport and storage licensed spent fuel casks for onsite or offsite long term storage.

6. Plant Safety and Operational Performances

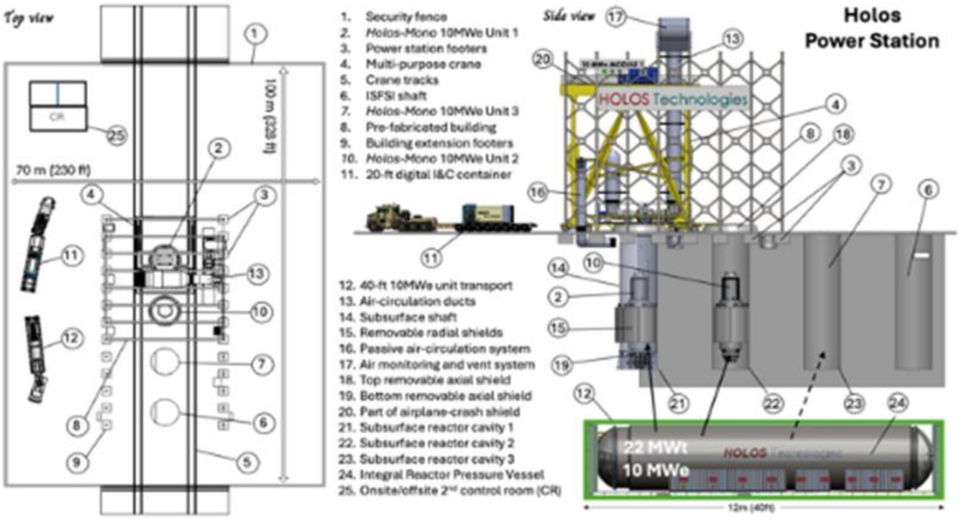
The underlying defense-in-depth offered by the HOLOS-*Mono* design is based on its qualified, accident tolerant, TRISO fuel. This type of fuel also enables high-temperature operations leading to a high thermal-to-electric conversion efficiency. The SPM is further sealed within the container (for units' power rating up to 10MWe) and the container is further sealed within a dedicated compartment (above surface, ship and off-shore installation) or a reactor cavity through a low-cost borehole (below surface installations). These configurations multiple pressure boundaries engineered to trap working fluid potentially contaminated (e.g. by defective TRISO) for treatment and neutralization, thus enhancing the already high safety performance associated with HTGR technologies. The unique integral power conversion system equipping this design further enables high-rate load following maintaining >40% efficiency for electrical load >30% of its nominal power. This translates into high operational performance, higher fuel utilization, and proportionally lowered spent fuel production per unit of electric energy generated. This further enhances safety as lowered volumes of spent fuel translate into reduced risks associated with spent fuel handling, transport and storage per unit of electricity produced.

7. Instrumentation and Control Systems

The design follows the electrical load automatically, with a power flow control system that maximizes the thermal-to-electric conversion efficiency while regulating the black-start/ballast battery or flywheel banks and the electrical loads. (Note that the HOLOS-*Mono* does not require connection to the power grid to operate). The digital instrumentation and control (I&C) system further regulates the active magnetic bearings, control drum and shutdown rod drives, and varies the compressor speed at a high-rate (supporting fast-acting power maneuvering). For a 10MWe unit, the equipment supporting I&C altogether with the power management modules regulating the Electric Storage System (ESS), occupies the volume equivalent to a 20-foot "digital I&C container" electrically coupled to the 40-ft container via shielded electro-magnetic interference and electro-magnetic pulse shunted data and power cables.

8. Plant Layout Arrangement

For power rating up to 10MWe per unit the HOLOS-*Mono* design features a highly integrated, transportable nuclear power plant fully comprised within the dimensional constraints of standard shipping containers. For above surface applications, the container is positioned via tractor trailer inside a prefabricated removable air-crash and fire-engulfment protected building. For underground operations the reactor installation is "verticalized" as the active magnetic bearings enable horizontal, angled or vertical operations. The HOLOS Power Station can also be equipped with HOLOS-*Mono* units with >10MWe load-follow power output without changing the design architecture.



9. Testing Conducted for Design Verification and Validation

The design has been investigated for several years by design and resource teams formed by scientists and subject matter experts at the national laboratories, industry and academic institutions, sponsored in part by the U.S. DOE ARPA-E MEITNER funding program. Validation activities included safety, technical and economic performance quantification via high-fidelity computer modelling developed by the U.S. national laboratories and testing through a subscale simulator equipped with a graphite core and electrical heater to mimic TRISO fuel behavior under normal operations, off-normal and accident conditions.

10. Design and Licensing Status

Preliminary safety performance analyses based on a HOLOS Power Sation formed by 3x 10MWe units with the ability to further uprate the station have been completed and verified through operations of a Sealed Power Module-Subscale Simulator (SPM-SS). Safety assessments were validated through operations of the SPM-SS at relevant full-scale operating and accident conditions. Validation testing was completed in 2022 and continued in subsequent years, qualifying the design to TRL6. A preliminary safety assessment report (PSAR) for the design installed underground at a U.S. federal facility has been completed in 2023. The PSAR will be submitted for review by the U.S. Nuclear Regulatory Commission (NRC) pending client's identification of final power requirements.

11. Fuel Cycle Approach

The design configured to supply 10MWe per unit is currently equipped with a core optimized for 8 EFPY. Spent fuel passively dissipate decay heat, and can be placed within dual purpose transport and storage licensed shielded dry cask. Overall, HOLOS-*Mono* spent fuel will be stored at onsite or offsite ISFSI. The fuel cycle approach currently adopted for this design is *once through*.

12. Waste Management and Disposal Plan

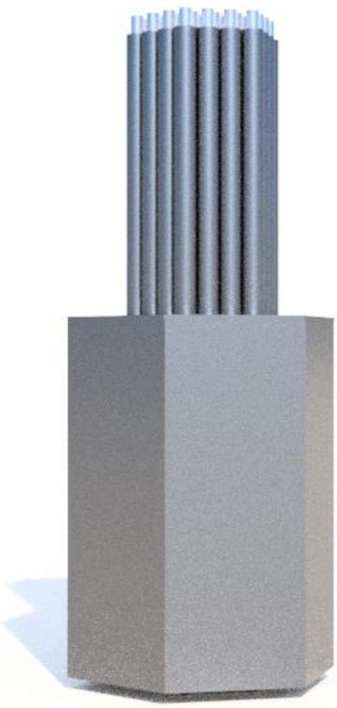
The great majority of the power conversion components and their ancillary components are not exposed to irradiation effects as the design is equipped with internal shields to protect turbomachinery equipment and will not require specialized decommissioning procedures. As the HOLOS-*Mono* power conversion system performs at high efficiency, costs and volumes associated with spent fuel transport, storage and general handling are reduced per unit of electric and process heat energy generated. Equipment subjected to irradiation and classified either as high-level or low-level radiation, as well as spent fuel in dry casks, will be processed and stored at ISFSIs.

13. Development Milestones

2015 – 2017	Conceptual design developed by Filippone & Associates LLC (F&A)	Complete
2017 – 2018	HolosGen LLC developed and successfully tested the HOLOS-Test-Assembly-1 (HTA-1) subscale prototype equipped with a surrogate core.	Complete
2018 – 2019	HolosGen awarded by the U.S. DOE ARPA-e MEITNER funding program to demonstrate safety and techno-economic performance.	Complete

2019 – 2022	Successfully tested the Sealed Power Module-Subscale Simulator (SPM-SS) operated to verify computer model projections developed by the Argonne National Laboratory. The SPM-SS was further operated through multiple shutdown and startup to simulate thermal cycling and accelerated material aging. The SPM-SS further validated safety performance under beyond design basis accident scenarios.	Complete
2022 – 2023	Completed the Preliminary Safety Analysis Report (PSAR) for specific use cases involving an expandable HOLOS Power Station with multiple 10MWe units with undersurface installation at a U.S. federal facility.	Complete
2023 – 2024	Developed detailed step-by-step installation procedure for a 30MWe HOLOS Power station equipped with a prefabricated removable building with reduced footprint and deactivation and decommissioning costs.	Complete
2024 – 2027	Complete digital twin and electrical full-scale twin of the HOLOS-*Mono* configuration.	Planned
2027 – 2028	FOAK nuclear testing at U.S. DOE or alternate facility.	
2029	Conversion of the FOAK demonstrator unit into first commercial unit.	Planned

JIMMY (JIMMY ENERGY SAS, France)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JIMMY ENERGY SAS, France
Reactor type	High temperature gas-cooled reactor
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 to 20 / n.a.
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	1.5 / 3.0
Core inlet/outlet coolant temperature (ºC)	300 / 700
Fuel type/assembly array	UCO TRISO particles packed into pellets
Number of fuel assemblies in the core	N/S
Fuel enrichment (%)	9,8 - 19.75
Refuelling cycle (months)	No refuelling
Core discharge burnup (GWd/ton)	110
Reactivity control mechanism	B_4C Control rods
Approach to safety systems	Passive
Design life (years)	10 to 20
Plant footprint (m^2)	1500 (entire installation)
RPV height/diameter (m)	5 / 5
RPV weight (metric ton)	15
Seismic design (SSE)	Zone 4
Fuel cycle requirements/approach	No refuelling, no on-site storage
Distinguishing features	Compact, low-weight design; simplicity of manufacturing, flexibility of operation; intrinsic passive safety; based on proven technologies which have already been industrialized
Design status	Detailed design

1. Introduction

Jimmy is a high-temperature gas-cooled reactor with a thermal power of 10 to 20MW(t), designed to provide competitive, low-carbon industrial process heat. The reactor is based on UCO TRISO particles, a graphite moderator and a helium coolant (primary loop). A CO_2 secondary loop allows to deliver heat up to 550°C to industrial processes, selecting the secondary heat exchanger according to industrial needs. Valuing the existing HTGR return of experience, the approach consists in delivering a simple, safe and robust heat generator. As a result, the design relies mostly on well-tested solutions, as well on a passive safety demonstration, to optimize the time- and cost-to-market. A first official application has been submitted (April 2024), for a first generator project, with a first client site already identified (France).

2. Target Application

Jimmy targets the industrial heat market, where industrial suffer from both economic and environmental pressure to reduce their use of fossil fuels, while no alternative solution available. Jimmy will allow to replace industrial gas-burners and provide competitive, low-carbon industrial process heat. The design and small reactor footprint and limited capacity (up to 20MWt) allows to integrate into most industrial sites, with one or several generators in a row. The primary target is the French market, especially steam-consumers such as the Agro or Chemistry industry, with needs for steam up to 550°C. After massive deployment to process heat applications in and outside of France, the main applications are district heating (offices, residential buildings…) and hydrogen production.

3. Design Philosophy

The main goal of the design is to provide a simple, safe and financially viable alternative to industrial heat gas-burners. This goal is achieved by using HTGR well-tested solutions (especially regarding safety) and then optimizing the design to optimize safety, reduce cost, and allow easy manufacturing, transportation and assembly. For instance, the use of a secondary CO_2 loop allows to distinctly separate the nuclear perimeter from the industrial zone in terms of safety. Furthermore, the manufacturing and assembly aspects have been taken into account since the beginning of the detailed design: design teams are in contact with presented suppliers to screen commercially available components and integrate them in the design. It ensures that a viable industrial scheme is available and to anticipate the orders of long-lead items, as well as to optimize instruction time by safety authorities.

4. Main Design Features

(a) Power Conversion

The Jimmy generator is designed to provide heat only, i.e., there is no energy conversion into electricity. A primary exchanger allows to transfer heat from the primary helium-based loop to the secondary CO2-based loop. The CO2-based loop pressure is 3.0 MPa and temperature is 600°C. A secondary exchanger is then selected, according to each industrial site's specific needs (fluid, temperature, pressure, etc.).

(b) Reactor Core

The reactor core consists of fuel assemblies inside a graphite reflector. Theses fuel assemblies are designed to be assembled in a factory and transportable inside existing fuel transportation packages.

(c) Fuel Characteristics

The fuel are pellets of HALEU TRISO particles that are conformed to the AGR program envelope to reach the expected burnup.

(d) Reactivity Control

Reactivity control is made with 24 independent B_4C control rods and B4C neutronic poisons that cross the core or the reflector.

(e) Reactor Pressure Vessel and Internals

The Jimmy reactor does not feature a pressure vessel but pressure tubes that constitute the core. This whole core is then contained into a containment vessel. This geometry facilitates the transport and assembly of the reactor on its installation site.

(f) Reactor Coolant System

The primary coolant is a helium pressurized circuit, whose circulation is forced on the cold branch by a helium blower. The circuit divides and reunites before and after the pressure tubes of the core.

(g) Secondary System

The secondary circuit connects to the primary circuit through a plate heat exchanger. It contains pressurized carbon dioxide whose circulation is also forced.

(h) Steam Generator

Jimmy's generator does not directly produce steam, but its industrial client may produce steam based on the heat it gets from the secondary system through a secondary exchanger.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Jimmy's safety is mainly based on the leverage of three properties that the HTR can achieve: passive decay heat removal, TRISO particles robustness and high reactivity coefficients. Based on these intrinsic properties, most of the safety systems are around redundant and highly independent control rods.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

During normal operation, the reactor is cooled thanks to the industrial site that use the transferred heat for its process.

During accidental operation, the graphite plays the role of a thermal absorber. The low power density of the reactor easily ensure that the fuel remains below the limit temperature that it can endure.

(c) Spent Fuel Cooling Safety Approach / System

Calculations and modelling show that spent fuel from a Jimmy's reactor cools passively and does not require temporary on-site storage in a spent fuel pool.

(d) Containment System

The design's containment system is based on the "defence in depth" philosophy and includes a set of technical measures aimed at preventing environmental exposure to radiation and radioactive substances from the reactor.

- The set of barriers includes:
- The coating layers of TRISO particles;
- The graphite of the core
- The primary system
- The containment of the building

(e) Chemical Control

The main chemical control is a helium purification system that supplies a highly pure helium and monitor the health of the core

6. Plant Safety and Operational Performances

The goal of Jimmy is to ensure that large radioactivity release frequency is less than 10^{-7}/reactor year.

7. Instrumentation and Control System

The I&C system is under design and does not contain any major disruption

8. Plant Layout Arrangement

The layout of Jimmy's reactor is a building of w: 25m * l: 25m * h: 25 m. For security reasons, walls are thick to prevent aggression and only few systems go through the wall (mainly the RCCS and the secondary system that reaches the industrial line). All the access for maintenance and installation are on the roof to stay out of an easy reach.

9. Testing Conducted for Design Verification and Validation

Jimmy's philosophy is to use already tested technologies. Thus, most of the required tests are validation tests on the behaviour on well modelled pieces. Jimmy does not plan to conduct a specific test program.

10. Design and Licensing Status

Interaction with Nuclear Safety Authority – Formal licensing process under way;

(Launch of licensing procedure with the first regulatory application (Dossier d'Options de Sûreté) in April 2020, allowing the French Safety Authority to start the instruction)

Site permit for FOAK plant – To be developed.

(Application to be submitted in June 2023 – First site for FOAK has already been identified, with seismic and other environmental studies underway)

11. Fuel Cycle Approach

In order to simplify the design and the operation, and to maximize safety, no refuelling will be done. After the 10-to-20-year life (depending on the nominal power chosen by the client), the entire vessel is extracted from the generator and replaced. Jimmy has a long-term strategy to foster fuel recycling between its systems.

12. Waste Management and Disposal Plan

By design, the Jimmy reactor does not release any waste during normal operation. The only possible source of waste would be the leaks from the helium loop, which is filtered and stored.

At the end of reactor lifetime, the reactor produces waste that can be collected and disposed of in accordance with applicable regulation, relying on existing French waste management infrastructures with minimal adaptations.

13. Development Milestones

2021	Creation of the company
	Seed fundraising allowing to finance Basic Design and first regulatory application
Done so far in 2022	Basic Design completed

	Peer review of Basic Design
	Feasibility study for a first industrial client (Agro industry) completed
	Launch of licensing procedure with the first regulatory application (*Dossier d’Options de Sûreté)* and instruction by the French safety authority (ASN)
Current goals	Preliminary Design completed
	Peer review of Preliminary Design
	First orders placed for long-lead items
	First design team completed (20+ engineers)
2023	Detailed Design completed
	Peer review of Detailed Design
	Manufacturing Design completed
	Peer review of Manufacturing Design
	Orders placed for medium-lead items
	Public announcement of the purchase of a plot of land in Le Creusot, for the installation of Jimmy's industrial platform
2024	Licensing application for an assembly plant in Le Creusot and beginning of the instruction by the French safety authority
	Second licensing application (*Demande de Décret d’Autorisation de Création*) and beginning of the instruction by the French safety authority, for a 1^{st} generator on a 1^{st} customer site
	Communication among local population
	Launch of feasibility studies with several industrial sites in France and signature of two MoUs with foreign players to set up industrial partnerships
	Orders placed for all items
	Equipment qualification
2025-2026	Module pre-assembly (out of site)
	On-site land development
	Obtaining of licensing approval
	On-site reactor assembly & tests
	Fuel loading & tests
	Commissioning

MARVEL Research Microreactor (Idaho National Laboratory, USA)

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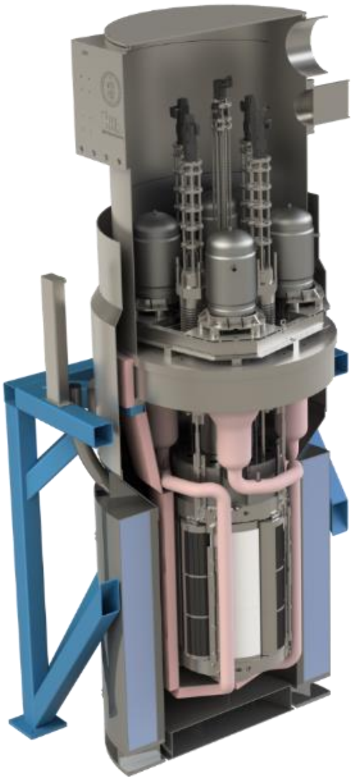

MAJOR TECHNICAL PARAMETERS	
Parameter	Value
Technology developer, country of origin	Idaho National Laboratory, USA
Reactor Type	Liquid Metal Cooled Thermal reactor
Coolant/Moderator	Sodium-Potassium Eutectic, Hydrogen in fuel
Thermal/electrical capacity, MW(t)/MW(e)	0.075-0.1 / 0.015-0.027
Primary Circulation	Natural Convection
NSSS Operating Pressure (primary/secondary)	0.39 / 0.22
Core Inlet/Outlet Coolant temperature (C)	450 / 520
Fuel Type/Assembly array	Uranium Zirconium Hydride
Number of Fuel Assemblies in the core	6
Fuel Enrichment	19.75%
Refueling cycle (months)	> 60
Average Core Discharge Burnup (GWd/ton)	1.50 (current; can be increased if needed)
Reactivity Control Mechanism	Four ex-core, safety-related control drums, One defense-in-depth central shutdown rod
Approach to safety systems	Doppler and moderator-based inherent safety in fuel; Primary coolant, intermediate coolant, and decay heat removal is driven via passive, natural circulation. Loss of on-site power actuates shutdown mechanism passively
Design Life (years)	2-40 (depending on transients)
Plant footprint (m^2)	8.9
RPV height/diameter (m)	2.55 / 1.22
RPV weight (metric ton)	5.94
Seismic Design (SSE)	IBC 2015 Risk Category IV Site Class C
Fuel Cycle Requirements/approach	System life is limited to 2 years
Distinguishing Features	Compact core; naturally driven, adjustable high-grade heat extraction; small footprint
Design Status	Equipment manufacturing in progress

1. Introduction

The Microreactor Applications Research Validation and Evaluation (MARVEL) Project entails the design, development, construction, and startup of an INL test microreactor, funded by the United States Department of Energy (DOE) via the Microreactor Program (MRP). The goal of the project is to establish an operational nuclear applications test bed that can generate combined heat and power to enable integration and R&D with end-user technologies, as well as allow microreactor technologists to test next-generation control systems. The microreactor is a thermal reactor utilizing Uranium Zirconium Hydride (UZrH) fuel with authorization to be provided by the Department of Energy Idaho Operations Office (DOE-ID) for National Environmental Policy Act (NEPA) compliance, safety review, and supplemental readiness assessments for startup and operation. To enable rapid deployment, the MARVEL reactor will reside in the Transient Reactor Test (TREAT) Facility and utilize the existing operating Category B reactor facility, operating crews, and recent re-start experience. The MARVEL team consists of Idaho National Laboratory (INL), Argonne National Laboratory, Los Alamos National Laboratory, Walsh Engineering, Munro & Associates, and Creative Engineers Inc. The primary purpose of investigating a near-term concept is to determine its feasibility and utility for future microreactor research and development, such as a test bed for integrating microreactors with electric and non-electric applications and demonstrating the technology to key end-users. It can also drive infrastructure development and capabilities to support future microreactor demonstrations. The U.S. DOE Microreactor Program supports research and development (R&D) of technologies related to the development, demonstration, and deployment of very small, factory-fabricated, transportable reactors to provide power and heat for decentralized generation in civilian, industrial, and defense energy sectors. Such applications currently face economic and energy security challenges that can be addressed by this new class of innovative nuclear reactors. Led by INL, the program conducts fundamental and applied R&D to reduce the risks associated with new technology performance, manufacturing readiness, and deployment of microreactors. The program aims to verify that microreactor concepts can be licensed and deployed by commercial entities to meet specific use case requirements. Microreactors, often referred to as special-purpose reactors or very small modular reactors, are factory manufacturable, easily transportable, and designed to produce tens of megawatts of thermal (MWth) energy. This power limit allows microreactors to be classified as Hazard Category 2 per the Code of Federal Regulations (CFR) at 10 CFR 830 and DOE-STD-1027. These reactors are decentralized energy sources that can provide sustainable and affordable heat and power to remote communities and industrial users while having self-contained geometry that requires very low maintenance. Microreactors are intended to be self-regulating and do not rely on engineered systems to ensure safe shut down and removal of decay heat.

2. Target Application

MARVEL is targeted at researchers, technology developers, and end-users for enabling the successful deployment and adoption of microreactors through research and education. Some examples of R&D activities are:

<u>Test, demonstrate, and address issues related to installation, startup, and operation</u>

- Simplify siting and environmental review process
- Startup methodology for microreactors
- Normal operating transients such as startup and load management
- Cyber and physical security hardening
- Integration with a net-zero, electrical microgrid
- Demonstration of high and low-grade heat extraction.

<u>Enable Automated Operation Technologies</u>

- Automate operator functions, while maintaining reactor safety
- Demonstrate radiation and temperature-hardened sensors and instrumentation to enable remote monitoring, advanced sensor reliability tests, and online calibration
- Live data can feed a digital twin of the reactor to “train” an artificial intelligence-based control system
- Demonstrate transmission of live data of both electrical and thermal power output during startup, operation, and shut down. This allows real-time feedback on system output, performance, and prediction of any unplanned maintenance needed in an operating microreactor.

<u>Enable Application Integration</u>

- The control systems manage the energy grid demand and reactor power and heat supply. This management requires a carefully designed control system that can predict the interplay of controls, thermal inertia, and reactivity feedback.
- Demonstrate integration approaches for a range of applications investigating both reactor power management and load management approaches.

3. Design Philosophy

The design philosophy is generally to select technologies that can lead to criticality within the shortest time possible. Time is the essential metric of project performance without sacrificing safety and quality. This drives the team to select technologies that are either (i) off-the-shelf, (ii) can be implemented, and/or (iii) can be developed to accelerate technology and manufacturing readiness levels. Innovation and creativity are embedded in the team culture and celebrated. This includes adopting multiple engineering tools and processes from non-nuclear technology industries, e.g. Scrum Agile Product development, and Design for Manufacturability (DFM) modeling from the automotive industry.

4. Main Design Features

(a) Power Conversion

MARVEL is capable of and is originally conceptualized as utilizing commercial, off-the-shelf Stirling Engines from Qnergy. Each Stirling engine can produce 5-6 kilowatt and comes equipped with supporting ancillary equipment for low-grade heat rejection. Qnergy equipment will absorb available high-grade heat and then convert that heat into either using electric power or low-grade heat for rejection into site processes or finally into the ultimate ambient heat sink. An alternate high grade heat extraction system will be available for extracting high grade heat for process heat applications testing, or for testing alternate power conversion systems such as Rankine or Brayton cycle.

(b) Reactor Core

The MARVEL reactor will contain fuel elements sourced from TRIGA International (TI): LEU, 30% uranium, by weight, 19.75 % enriched (TRIGA LEU 30/20). Beryllium and graphite reflected, ZrH moderated, solid fuel, tank type reactor. The MARVEL reactor core comprises 36 LEU fuel elements arranged in a triangular pitch lattice around a central voided location for the central insurance absorber (CIA) rod assembly. Surrounding the core is a thick axial neutron reflector composed of beryllium oxide.

(c) Fuel Characteristics

The major components of the MARVEL fuel system are (1) U-$ZrH_{1.6}$ fuel slugs (annular), (2) Zirconium filler rods (to be inserted into fuel slug annulus), (3) Molybdenum spacer disk placed between lower graphite reflector and bottom of the fuel stack, (4) Stainless steel 304 cladding, (5) Top and bottom graphite axial reflectors, and (6) Top and bottom end plugs.

(d) Reactivity Control

The MARVEL reactor also has a high net negative temperature feedback for prompt reactivity control inherently due to the Doppler broadening of absorption resonances and spectrum hardening from the ZrH moderator with increased temperature. Manual and Passive Reactivity Control and shutdown are achieved with four custom-designed, safety-related Control Drums (CDs) and a defense-in-depth CIA that control reactivity via motor drivers in the control cabinet to position the drums and rods within the reactor structure. Passive actuation functions are built into the design for loss of power and inadvertent energizations of the motors.

(e) Reactor Pressure Vessel and Internals

The Primary Coolant System (PCS) is a high-temperature, low-pressure boundary that houses the core internals, reactor primary coolant, and argon gas headspace. In addition, the PCS passively maintains decay heat removal capability. The boundary is a metal weldment made from 316H stainless steel for high-temperature reactors designed per ASME Section III Division 5.

(f) Reactor Coolant System

The primary coolant system (PCS) is a four-loop hydraulic circuit assembled to transport nuclear fission heat from the nuclear fuel to the intermediate heat exchanger (IHX) using the natural circulation flow of the primary coolant. The PCS also transfers decay heat to the ultimate heat sink. Approximately 150 kg of NaK, liquid metal at room temperature, serves as the primary coolant. The power conversion heat exchangers connect to the reactor vessel and interface with the NaK coolant via the IHX containing eutectic gallium-indium-tin or other suitable, low melting point liquid metal coolant. The Stirling engine coils or high-grade heat exchanger, depending on configuration, extract heat from the primary coolant and reduce the NaK temperature. The cooled and denser NaK then flows outward to the periphery of the core, then downward through four downcomer pipes outside the beryllium oxide side reflector and through in the lower plenum. The NaK then rises back up through the active core under natural circulation forces driven by the heated section of the active core.

(g) Secondary System

Eutectic gallium-indium-tin (eGa-In-Sn) has currently been selected as the coolant for the IHX, serving as the secondary system. An argon gas sweep will be maintained on the IHX overhead gas space to remove any volatile activation products for ease of maintenance and repair. Alternative liquid metal coolants may be selected if the integrated Stirling engines are replaced with a high grade heat extraction system.

MARVEL Integral

5. Safety Features

MARVEL employs rigorous safety standards and expectations for (i) reactivity control, (ii) passive shutdown, and (iii) decay heat removal by relying primarily on physics rather than engineered systems. The fuel has a high prompt negative temperature reactivity feedback due to the strong Doppler broadening of absorption resonances and neutron spectrum hardening from the ZrH moderator. In addition, passive shutdown features enable the use of potential energy to actuate negative reactivity in the core. Both primary and intermediate coolants are driven by natural circulation. Decay heat is removed from the fuel to the outer shell of the reactor primarily by conduction and radiative heat transfer and subsequently rejected to ambient air by convection. Finally, there are three robust fission product barriers- fuel cladding, primary coolant boundary, guard vessel to confine hazardous materials throughout system life in all operational, postulated design-basis, and beyond-design-basis events.

6. Testing Conducted for Design Verification and Validation

The MARVEL team conducts rapid prototyping tests to mature its technologies. So far, more than ten separate effects tests have been conducted, including, but not limited to, the intermediate heat exchangers, control drums, instrumentation control, neutron detection, shutdown rod actuators, and Stirling engines.

Due to MARVEL's novel thermal hydraulic circuit, where a liquid metal natural circulation primary loop is in series with four parallel liquid metal natural circulation loops, an integral effects test (IET) of the system was considered necessary for verifying the transient dynamics of the system before reactor construction. Hence the team has successfully designed and fabricated a full-scale electrically heated prototype of the MARVEL reactor. This is conducted per ASME boiler and pressure vessel code (Section III and VIII) within nine months by employing an agile development process. The test hardware includes (i) a Full-scale mechanical IET test article; (ii) eight electrical control cabinets; (iii) a structural frame; (iv) four IET flow meters; (v) more than 200 thermocouples and pressure transducers; and (vi) four Stirling engines, engine control, and heat rejection units. The goals of this IET are to: (i) validate flow and heat transfer characteristics of MARVEL technology; (ii) benchmark modeling & simulation parameters; (iii) streamline manufacturing methods; (iv) de-risk supply chain; and (v) train operators.

7. Design and Licensing Status

DOE conducted an environmental assessment (EA) as part of the NEPA process, which analyzed the potential environmental impacts of constructing the MARVEL microreactor inside Idaho National Laboratory's (INL's) Transient Reactor Test Facility. At the conclusion of the EA process, DOE issued a final EA with a "finding of no significant impact" (FONSI) for the MARVEL Project (DOE/EA-2146). A safety design strategy (SDS-119, Rev 1) has also been approved by DOE with a Preliminary Documented Safety Analysis (PDSA) submitted in Q3 CY24. The MARVEL reactor is currently in Final Safety Review and preparation for full fabrication phase, with 90% construction planned to conclude in Q2, CY25, and fuel load and initial criticality in Q4, CY27.

MMR (Ultra Safe Nuclear Corporation, United States of America)

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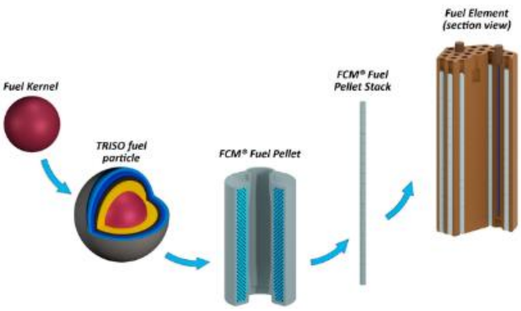
USNC's Micro Modular Reactor



KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Ultra Safe Nuclear Corporation, USA.
Reactor type	High Temperature Gas-cooled Reactor (HTGR)
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	45 / 15
Primary circulation	Forced (1 circulator)
NSSS Operating Pressure (primary/secondary), MPa	Variable, 1.3 - 6 / 0.3
Core Inlet/Outlet Coolant Temperature (ºC)	300 / 660
Fuel type/assembly array	TRISO coated particles inside Fully Ceramic Micro-encapsulated (FCM) matrix / Hexagonal array
Number of fuel assemblies in the core	150 fuel elements
Fuel enrichment (%)	9.9% to 19.75% U-235
Core Discharge Burnup (GWd/ton)	Approximately 50 – 150 GWd/MTU (based on enrichment and refueling approach)
Refuelling Cycle (months)	18 months to 27 years (depending on fuel enrichment and reactor power level)
Reactivity control	Control rod insertion, negative temperature coefficient
Approach to safety systems	Passive safety design, no active decay heat removal required
Design life (years)	40
Plant footprint (m^2)	1,200 - 4,700
RPV height/diameter (m)	6 / 4.5
RPV weight (metric ton)	60
Seismic Design (SSE)	0.3g – 0.6g
Distinguishing features	Micro-reactor, off-grid power potential, modular design, designed for remote/off-grid locations, passive safety systems
Design status	Preliminary Design; Licensing in Canada

1. Introduction

The MMR is a micro-reactor that operates as a flexible nuclear battery and may fulfil nontraditional roles for nuclear power, including service to remotely sited areas, backup power generation, hydrogen production, desalination, process heating, and supporting military and critical national infrastructure facilities.

The energy system consists of two plants:

- The nuclear plant: This is the nuclear facility which contains the MMR High Temperature Gas-cooled Reactor(s) (HTGRs) and includes all the equipment required to transport the heat produced to the adjacent plant.
- The adjacent plant: A non-nuclear facility which harnesses the heat for client specified applications. These applications could be electrical generation, process heat (for example, steam), hydrogen production, district heating etc.

2. Target Application

USNC's MMR is a fission type micro-reactor that provides heat to an MMR Energy System. An MMR Energy System can be configured for a number of applications, such as electricity generation, process heat delivery, hydrogen production, district heating, etc.

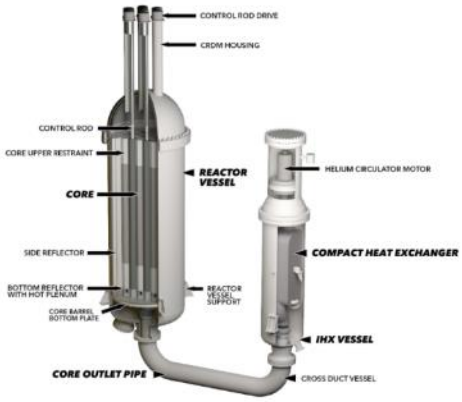
3. Design Philosophy

The design embraces the separation principle, the nuclear plant is physically separate from the adjacent plant, which reduces the nuclear footprint of the overall facility. Figure 2 shows a typical plot plan for an MMR Energy System that is customized for Combined Heat and Power (CHP) delivery. USNC typically provides a package referred to as the "Standard MMR Product" or simply "the Product" for short 3. The Product contains the reactor, its direct supporting services, the security facility and the thermal storage. (The thermal storage can be sized differently if the project requires it). The reactor can be seen in the citadel building, which in turn is anchored in the bedrock. Above grade, some of the supporting buildings are visible with the adjacent plant in the background.

4. Main Design Features

(a) Nuclear Steam Supply System

The Nuclear Heat Supply System (NHSS) is a crucial component in nuclear power plants, converting nuclear potential energy into usable process heat. It involves transferring heat from the reactor core to an intermediate heat exchanger (IHX) and then to the molten salt loop. The helium, a heat transfer fluid, is circulated by a fully submerged circulator, ensuring fuel integrity. The NHSS can operate at various power levels, adjusting the operating pressure of the helium to match project objectives. Some customizations are permanent.



Nuclear Heat Supply System

(b) Reactor Core

The active reactor core consists of hexagonal graphite blocks that contain stacks of the FCM fuel pellets. The MMR core has a low power density and a high heat capacity, which results in very slow and predictable temperature changes. The reactor core provides adequate flow paths for heat removal via the primary heat transfer fluid through and around the annular FCM fuel, and the graphite material of the blocks assists with further heat removal.

(c) 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. The core has strong negative temperature feedback.

(d) Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel (RPV) houses the core. The MMR avoids major water ingress events because there is no steam generator in the primary circuit. The MMR design includes metallic seals to ensure leak tightness. The Reactor Pressure Vessel is designed in accordance with the ASME Boiler & Pressure Vessel Code (BPVC).

(e) Reactor Coolant System and Steam Generator

The Reactor Coolant System in the MMR design utilizes helium as the primary heat transfer fluid. Helium circulates within the system, transferring heat from the reactor core to the Intermediate Heat

Exchanger (IHX), where it passes the heat to the molten salt loop. The system operates at various power levels by adjusting helium pressure, allowing flexibility in power output. The Steam Generator is located on the secondary side of the system, where molten salt from the hot salt tank is directed. The stored heat in the molten salt is used to generate steam, which is then used in a conventional Rankine steam cycle to produce electricity. Steam extraction from the turbine is used for district heating, making the process highly efficient for both power generation and heat supply.

(f) Primary pumps

The Primary Pumps in the MMR reactor system are referred to as helium circulators. These circulators are fully submerged and operate within the primary circuit to circulate helium gas through the reactor core and back to the intermediate heat exchanger. Since helium is used for heat transfer, the primary pumps are designed to handle gas-phase circulation at high temperatures and pressures, ensuring reliable operation under various power conditions. The helium circulators are critical for maintaining the flow of the heat transfer fluid through the reactor core, ensuring consistent heat removal and overall system stability.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The MMR safety concept is built on multiple defense-in-depth features, leveraging the high-temperature stability of the Fully Ceramic Microencapsulated (FCM) fuel, the properties of helium as an inert heat transfer fluid, and the passive decay heat removal system. The MMR reactor is designed to ensure safety without relying on active or powered systems, or operator actions. The geometry of the core, along with its high surface area to volume ratio and the uninsulated reactor vessel, allows heat removal via radiation, conduction, and convection. Additionally, control systems insert upon the trip of the reactor protection system, and no operator action is required during abnormal or accident conditions.

(b) Safety Approach and Configuration to Manage DBC

The MMR has been designed to ensure that the reactor can manage Design Basis Conditions (DBC) without the need for electrical power, engineered control actions, or operator intervention. The inherent characteristics of the helium heat transfer fluid and graphite core allow for safe and predictable temperature management. During DBAs (Design Basis Accidents), heat is passively transferred from the core to the surrounding reactor cavity cooling system, and radioactive fission products are contained within the FCM fuel. The reactor can safely accommodate a loss of helium and forced circulation without exceeding fuel temperature limits.

(c) Safety Approach and Configuration to Manage DEC

The MMR’s high-temperature capability of the FCM fuel, combined with passive safety systems, ensures that the reactor can manage Design Extension Conditions (DEC). The silicon carbide encapsulation of the fuel provides an additional barrier for fission product retention, ensuring that the fuel can withstand temperatures far beyond normal operating conditions. In the event of extreme accidents or beyond design basis conditions (BDBAs), the MMR's passive heat removal system and the inherent stability of the fuel prevent significant fuel particle failure and fission product release.

(d) Containment System

The MMR does not make use of containment structures, i.e., there is no containment building. In terms of functional containment:

- The FCM fuel pellets serve to functionally contain radionuclides for all plant states (both in operating and accident conditions).
- For operating plant states, the Citadel Building provides confinement (and controlled release) of radionuclides. This is achieved using pressure zones and filtering.
- Limited vented release by the Citadel Building is accepted for DBAs

(e) Spent Fuel Cooling Safety Approach / System

Spent fuel assemblies are removed from the reactor and packaged directly into dry storage. The packaged spent fuel can be stored on site or moved to an off-site storage facility. MMR design focuses on passive safety features, and the FCM fuel ensures containment of fission products throughout its lifecycle, reducing the need for complex spent fuel cooling systems.

6. Plant Safety and Operational Performances

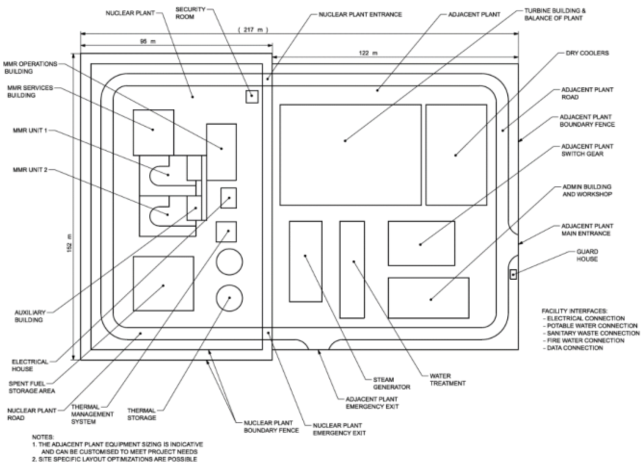
The MMR's design includes passive safety systems that ensure no requirement for electrical power, engineered control actions, or operator intervention during design basis accidents. The operational performance is enhanced by the use of high-temperature stable FCM fuel, helium as a heat transfer fluid, and a robust, low-power density core, which allows slow and predictable temperature changes, ensuring high safety and reliability during operation.

7. Instrumentation and Control Systems

It is noted that control systems will automatically respond to reactor trips by inserting control rods without requiring operator action during abnormal or accident conditions.

8. Plant Layout Arrangement

The MMR Energy System is suitable to be installed on remote, inland sites next to the offtake facility. The MMR Energy System typically includes a Citadel Building, Unit Auxiliary Building, Maintenance Enclosure, Reactor Services Building, Operations Building, and Security Room. The Citadel Building houses the entire NHSS and its supporting services, acting as a barrier against external hazards and radiation exposure. The Maintenance Enclosure provides shelter over the operating floor and facilitates refueling activities. The Reactor Services Building is a steel-clad warehouse-type structure that handles new and spent fuel handling and maintains activated or contaminated reactor components. The Operations Building houses the control room, electronics room, and security services.



Typical MMR Energy System – Plot Plan

9. Testing Conducted for Design Verification and Validation

The MMR has been designed with mature technology to enable early deployment and reduce the amount of technology development work. This has been achieved by following the following principles:

- Select operating temperatures that allow use of qualified materials
- Use of technology that is commercially available in the market today
- Build on the technology developed for operating HTGR
- Use the technology development work done in the US for NGNP and Advanced Reactors
- Avoid novel technology requiring extensive development
- Develop advanced FCM fuel and manufacture on production scale equipment in a pilot facility

10. Design and Licensing Status

From a licensing perspective, the nuclear plant is independent of the adjacent plant, which reduces the nuclear footprint of the overall facility. This reactor design is a new concept with a projected earliest deployment (start of construction) time of 2026. An MMR unit generates approximately 45 MW of thermal energy, which is sufficient to generate 15 MW of electricity.

11. Fuel Cycle Approach

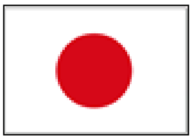
The fuel source for the MMR reactor will be a fully encapsulated, TRISO-coated Uranium Oxycarbide (UCO) kernel. The UCO kernel will be coated with successive layers of buffer, inner pyrolytic carbon, silicon carbide, and outer pyrolytic carbon. This multi-layer structure acts as a fission product retention barrier throughout the fuel lifecycle. The MMR uses USNC's Fully Ceramic Microencapsulated (FCM) fuel, ensuring containment of radioactive byproducts, making the fuel inherently safe across the entire fuel cycle.

12. Waste Management and Disposal Plan

Spent fuel is expected to be packaged into dry storage on-site or moved to an off-site facility, following standard practices for waste management in nuclear plants.

13. Development Milestones

2011	Secured FCM Fuel and MMR Reactor Patents
2016	Established R&D and Fabrication Laboratories
2017	Initiated FCM Fuel Qualification Plan; Started Vendor Design Review Phase I in Canada
2018	GFP submits proposal to CNL, supported by OPG & Ultra Safe Nuclear Company
2019	Started Vendor Design Review Phase II in Canada; Started License to Prepare Site Application in Canada
2021	Start NRC Engagement in USA
2026	Demonstration Project construction



MoveluX (Toshiba Energy Systems & Solutions 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 (°C)	Heat pipe: 680 / 685 Heat Exchanger: 450 / 680
Fuel type/assembly array	Silicide (U_3Si_2) / Hexagonal
Number of fuel assemblies in the core	66 (fuel), 72 (CaH_2)
Fuel enrichment (%)	4.8 – 5.0
Refuelling cycle (months)	Continuous
Core discharge burnup (GWd/ton)	1.0
Reactivity control mechanism	In-Ga Expansion Module (IGEM)
Approach to safety systems	Active / Passive
Design life (years)	10 – 15
Plant footprint (m²)	100
RPV height/diameter (m)	6.0 / 2.0
RPV weight (metric ton)	TBE
Seismic design (SSE)	0.3 g
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
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 800°C is adopted for the moderator material.

2. Target Application

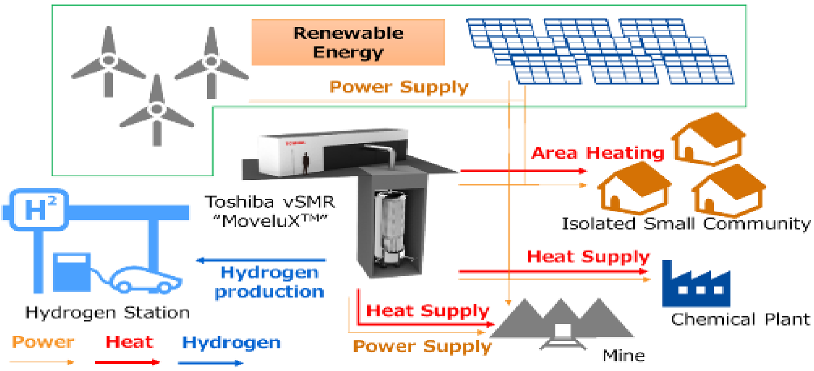
The MoveluX reactor system is a multipurpose energy source that can be used to produce electricity, hydrogen and high temperature heat. Since the reactor system can be installed in remote sites, the heat can be provided for chemical plants and steel mills. When it is used as a power plant it can be used 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. 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.



Utilization image of the MoveluX

4. Main Design Features

(a) Power Conversion

The MoveluX is considering using the gas turbine system for power conversion. The MoveluX can provide a high temperature of around 680°C, therefore, the gas turbine was selected from the viewpoint of the conversion efficiency.

(b) Reactor Core

The MoveluX core consists 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) Fuel Characteristics

The Uranium silicide fuel has high melting point and large thermal conductivity.

(d) 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 In-Gd Expansion Module (IGEM).

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).

(e) Reactor Pressure Vessel and Internals

Reactor vessel with atmospheric pressure is adopted.

(f) Reactor Coolant System

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.

(g) Secondary System

The secondary side of the MoveluX reactor system is currently a helium gas system. This gas system can provide high temperature around 700°C 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.

(h) Steam Generator

The MoveluX not uses the steam generator in the current design, however, this option can be examined.

5. Safety Features

(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) Decay Heat Removal System / Reactor Cooling Philosophy

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.

(c) Containment System

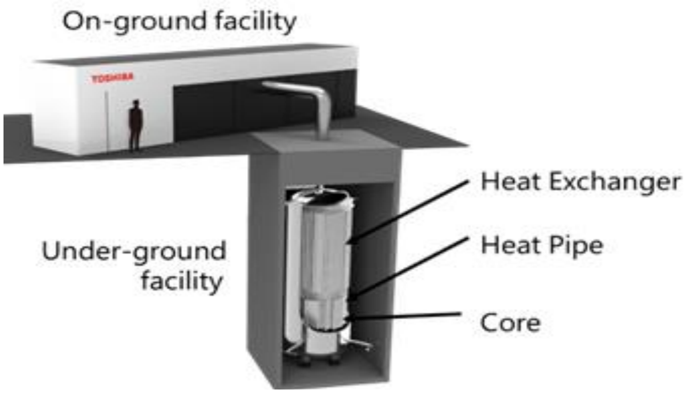
The fuel material is contained in the reactor vessel. The fuel material is separated from primary cooling system, because, heat-pipe is closed heat transportation device. Additionally, the heat-exchanger between heat-pipe and secondary circuit is functioning as one of the boundaries. Therefore, radionuclide will be confined in the reactor vessel unless reactor vessel breaking.

(d) Chemical Control

The MoveluX reactor does not required chemical control.

6. 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.



Schematic view of the MoveluX reactor system

Core Damage Frequency (CDF) : Extremely low.

Refueling : No refueling during the plant life time.

7. Instrumentation and Control System

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.

8. Testing Conducted for Design Verification and Validation

The experiment for the verification and modelling of the thermos-syphon heat-pipe and the compact heat exchanger have been conducting. Additionally, IGEM kinetic behaviour is measured by the neutron radiography. These experiments are scale test. The actual scale test will be conducted step by step.

9. Design and Licensing Status

MoveluX is at the conceptual design stage.

10. 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.

11. 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. On 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 a fresh fuel for recycling use in MoveluX, LWR or FR.

12. Development Milestones

2015	Start fundamental study based on the space reactor design
2017	Complete reactor type decision
2019	Start concept design
2028	Complete concept design and component demonstration
2030	Complete system demonstration without nuclear fuel
2035~	FOAK demonstration

Pylon D1 (Ultra Safe Nuclear Corporation, United States of America)

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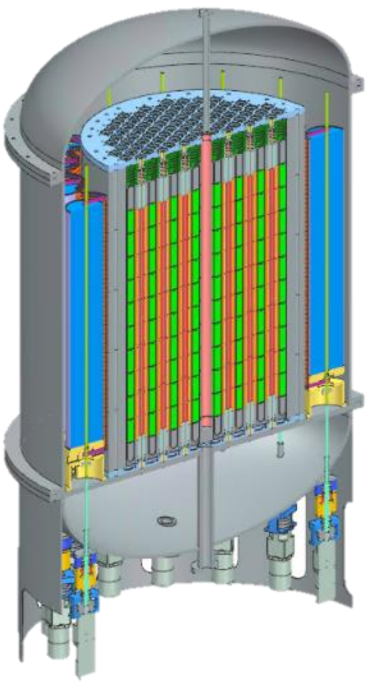

KEY TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Ultra Safe Nuclear Corporation, United States of America
Reactor type	HTGR
Coolant/moderator	Helium (Primary) and sCO2 (Secondary) / Zirconium Hydride
Thermal/electrical capacity, MW(t)/MW(e)	1 / 0 (No substantial electrical output)
Primary circulation	Forced (1 pump)
NSSS Operating Pressure (primary/secondary), MPa	2 / 20
Core Inlet/Outlet Coolant Temperature (°C)	327 / 727
Fuel type/assembly array	FCM™ TRISO / Hexagonal Prismatic
Number of fuel assemblies in the core	150
Fuel enrichment (%)	9.9
Core Discharge Burnup (GWd/ton)	2.5
Refuelling Cycle (months)	30 months
Reactivity control	Drums
Approach to safety systems	Passive
Design life (years)	N/A
Plant footprint (m^2)	500
RPV height/diameter (m)	1.5 / 1.65
RPV weight (metric ton)	2
Seismic Design (SSE)	TBD
Distinguishing features	Compact High Temperature
Design status	Conceptual Design

1. Introduction

The Pylon D1 Demonstration System contains a small, high-temperature gas-cooled reactor (HTGR) that utilizes Fully Ceramic Microencapsulated (FCM) TRISO nuclear fuel, metal hydride neutron moderator, and graphite control drums. Pylon D1 uses helium as the working fluid to transfer the fission generated heat from the reactor via the primary coolant loop to a secondary supercritical CO2 gas loop for rejection to the surrounding environment.

2. Target Application

Ultra Safe Nuclear Corporation (USNC) is developing the Pylon nuclear system architecture, a 10-ton micro reactor designed for research, transportable, and space applications. The Pylon system, based on USNC's Micro-Modular Reactor (MMR) and Fully Ceramic Microencapsulated (FCM) fuel technologies, is easily transportable and capable of safely generating electrical and thermal power anywhere. As part of its development, USNC plans to deploy a version called Pylon D1 for demonstration at the National Reactor Innovation Center (NRIC) in Idaho. This demonstration will validate the system's licensability, performance, safety, and cost-effectiveness.

3. Design Philosophy

The overarching design philosophy maintains a very high degree of conservatism, coupled with the simplest reactor plant design that enables reaching Pylon reactor demonstration goals. The demonstration is focused solely on reaching reactor power and temperature goals, without the expense, design, and safety analysis complexity of a coupled power generation balance of plant with grid connection.

The high-level goals for the Pylon D1 system are:

1. Achieve 1 MWth power,
2. Achieve 1000 K reactor outlet temperature,
3. Be designed to achieve 2.5 MWth-years of operation.

4. Main Design Features

(a) Nuclear Heat Supply System

The Pylon D1 reactor and helium primary loop will operate within the INL DOME containment structure. A shielding structure housed within the DOME containment structure will stop fast neutron and gamma radiation emitted from the reactor during and after operation.

(b) Reactor Core

USNC has the capability to design and additively manufacture nuclear fuel, allowing for rapid optimization in safety, performance, and reliability. Using a method licensed from Battelle Energy Alliance, USNC combines binder jet printing and chemical vapor infiltration to create a high-purity, >93% dense ceramic matrix. This advanced process is essential for producing high-performance nuclear fuel that can retain fission products even under abnormal conditions. Prototype fuel elements are currently being iterated at USNC's Pilot Fuel Manufacturing Facility in Oak Ridge, TN. The high TRISO packing fraction (>58%) achieved allows the Pylon reactor to use 9.9% enriched uranium (LEU+), reducing risks in the fuel supply chain. When commercial HALEU becomes available, USNC can choose to use it for fueling or refueling Pylon reactor cores, providing flexibility in adapting to market availability.

(c) Reactivity Control

For reactor control and protection, Pylon D1 will use control drums placed within the radial graphite reflector. Each control drum contains a layer of boron carbide neutron absorber. To control the reactor, the boron carbide layer is rotated towards or away from the reactor core, changing the amount of neutron reflection occurring in the radial reflector. To provide a diverse and redundant method for reactor shutdown, a safety shutdown rod will be located at the centre of the reactor. The gravity-driven safety shutdown rod is raised using an electromagnetic coupler, which drops the rod into the reactor if power is lost or signal is given by the reactor protection system.

(d) Reactor Pressure Vessel and Internals

The Pylon D1 reactor core assembly consists of a monolithic graphite structure, FCM fuel assemblies, and an upper and lower support plates. The graphite structure functions both as a neutron reflector and the primary structural support for the fuel assemblies. The fuel assemblies are combined in a prismatic hexagonal pattern to form the reactor core.

(e) Reactor Coolant System and Primary Heat Exchanger

The helium primary loop will transfer heat from the reactor through penetrations in the shielding structure to the primary heat exchanger.

(f) Primary pumps

The Pylon D1 system will use a centrifugal helium blower to provide helium circulation in the primary loop.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The safety concept for Pylon D1 is based on the high temperature capability of FCM nuclear fuel, its small size, and its control scheme. FCM fuel leverages the intrinsic properties of silicon carbide, both in the TRISO particle and in the fuel matrix to prevent the release of fission products from the nuclear fuel. With a reactor outlet temperature of 727 °C, the peak fuel temperature can be kept far below the fuel safety limit of 1600 °C.

(b) Safety Approach and Configuration to Manage DBC

Pylon D1's compact size results in a high surface area to volume ratio, allowing efficient management of fission product decay heat. This heat is conducted into the graphite radial reflector and then to the reactor pressure vessel, where it can be convected and radiated to the surrounding shielding and containment structure. For safety, Pylon D1 employs two diverse and redundant shutdown methods: control drums in the radial reflector and a central safety shutdown rod. The safety rod, fully removed during operation, has enough reactivity control to ensure shutdown independently of the control drums, which will also automatically engage during a reactor trip. Either system alone is sufficient to ensure reactor shutdown.

(c) Safety Approach and Configuration to Manage DEC

As a high temperature gas cooled reactor, Pylon D1's approach for managing DEC are the same as the approach to manage DBC. All fission product decay heat can be conducted away from the nuclear fuel, keeping fuel temperatures within safety limits, even in the case of complete loss of all primary coolant.

(d) Containment System

Pylon D1's containment system relies on FCM Fuel with robust silicon carbide radionuclide barriers, based on AGR-derived TRISO particles within a ceramic matrix. Additional safety features include an ASME-rated primary loop pressure boundary, the DOME containment building, and a remote operation site at Idaho National Laboratory. The DOME facility, originally built for the Experimental Breeder Reactor II (EBR-II), provides a containment dome that is 25 meters tall and 21 meters in diameter.

(e) Spent Fuel Cooling Safety Approach / System

As a demonstration reactor, Pylon D1 is not designed for regular refuelling. To simplify handling of the fuel assemblies, assemblies with be tied together by upper and lower support plates. The support plates form six supercells that will can be lifted for shipping, storage, and disposal.

6. Plant Safety and Operational Performances

No operator action is required in abnormal or accident conditions. Control systems will insert upon trip of the reactor protection system. Heat will transfer to the surrounding shielding and containment structures. Radioactive fission products will be contained within the FCM fuel. Fast neutron and gamma radiation will be blocked by the external shielding structure.

7. Instrumentation and Control Systems

Pylon D1 uses independently operated control drums located in the radial reflector, which contain mechanisms to fail-safe in the event of a loss of power, to control reactivity. Pylon D1 also uses a gravity-driven safety shutdown rod, providing a redundant, independent, and fail-safe form of reactor shutdown in the event of loss of power. The fuel, metal hydride moderator, and reflectors all behave differently (in terms of reactivity) as temperature increases. However, the net result is that the system always exhibits a safe negative temperature reactivity behaviour due to the metal content and fuel arrangement.

8. Plant Layout Arrangement

Pylon D1 will be installed in the Demonstration of Microreactor Experiments (DOME) facility at the Idaho National Laboratory (INL) Materials and Fuel Complex (MFC). System components will be skid mounted, allowing for construction and factory acceptance testing at an offsite USNC or vendor facility. System components will be shipped to the DOME facility where they will be inspected and connected. Primary and Secondary Loop working fluids will transfer heat to fluid-to-air heat exchangers outside the DOME facility. No cooling water will be required for operation of D1.

9. Testing Conducted for Design Verification and Validation

A phased acceptance and commissioning approach will be followed to ensure the results will demonstrate that the requirements of the Pylon D1 design meet the design and safety specifications and that it can safely operate as a nuclear demonstrator. Acceptance phases identified: research and development tests; off-site SSC acceptance; non-nuclear system assembly and integration test at USNC facility; and acceptance and commissioning of Pylon D1 in Dome.

10. Design and Licensing Status

To enable operation of Pylon D1, Idaho National Laboratory will submit licensing documentation to the US Department of Energy (DOE) for review and approval. USNC and Idaho National Laboratory have executed a contract to complete the Front-End Engineering and Experiment Design (FEEED), at the end of which will result in the submission of the Safety Design Strategy (SDS) and Conceptual Safety Design Report (CSDR). At this time the SDS has been submitted for review, a conceptual design review meeting has been held, and submission of the CSDR is expected October 2024.

11. Fuel Cycle Approach

The fuel source for the Pylon D1 reactor will be a fully encapsulated, TRISO-coated Uranium Oxycarbide (UCO) kernel. The UCO kernel will be 500 µm in diameter and 9.9% enriched. The UCO kernel will be coated with successive layers of buffer, inner pyrolytic carbon, silicon carbide, outer pyrolytic carbon. The multiple layers around the UCO kernel act as the fission product retention barrier throughout the fuel lifetime. The TRISO particle used in the Pylon D1 reactor is the same design used in USNC's MMR Chalk River Project and will be produced alongside them in the planned fuel production factory operated by the USNC/Framatome JV. UCO was chosen for Pylon ground demonstration because the fuel is available and there is fuel performance data available under the AGR testing program. The fuel performance data in the EPRI report "locks-in existing fuel performance data and results in a form that can increase the efficiency of the safety review process for design certification and licensing applications in the United States and internationally." The TRISO parameters are consistent with the AGR TRISO fuel tested.

12. Waste Management and Disposal Plan

USNC is working with the DOE National Reactor Innovation Center and Idaho National Laboratory on a waste management and disposal plan for the Pylon D1 reactor. Potential end use pathways include reuse of the system in a research capacity, temporary storage at Idaho National Laboratory, or direct transport to the ultimate disposal location.

13. Development Milestones

2018 – 2022	Preliminary studies and technological innovation (using previously developed patents).	Complete
2022 – 2023	Pre-conceptual design phase and technology validation	Complete
2023 – 2024	Conceptual Design Phase	On track
2024 – 2026	Detailed Design Phases	Planned
2026 – 2027	Procurement of Components and Acceptance Testing	Planned
2027 – 2028	Assembly, Commissioning and Operation at NRIC DOME Facility	Planned
2028 – 2030	Disassembly, Decommissioning, and Disposal	Planned

SHELF (NIKIET, Russian Federation)

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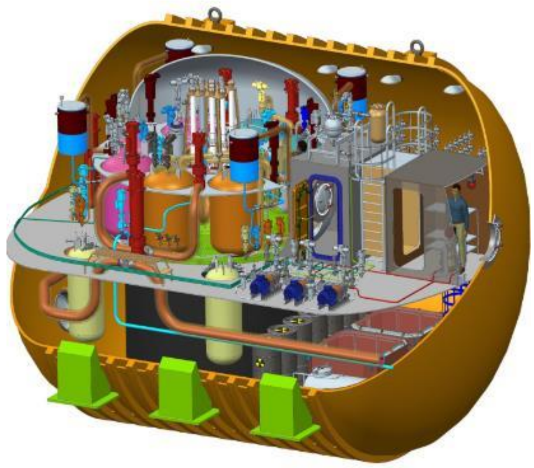
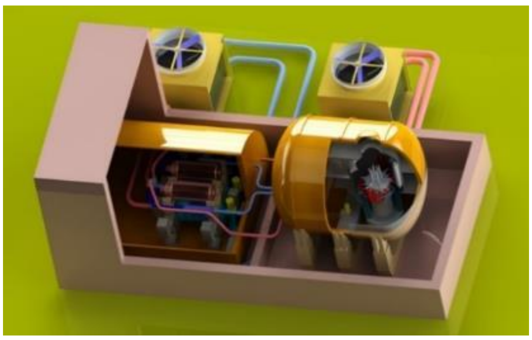




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	35.2 / up to 10
Primary circulation	Natural (for 20 % of rated power)
NSSS operating pressure (primary/secondary), MPa	14.7 / 3.7
Core inlet/outlet coolant temperature (ºC)	271 / 308
Fuel type/assembly array	Cermet (UO_2 + metallic silumin) / hexagonal
Number of fuel assemblies in the core	241
Fuel enrichment (%)	Up to 19.7
Refuelling cycle (months)	96
Core discharge burnup (GWd/ton)	162.4
Reactivity control mechanism	Control rods driving mechanism
Approach to safety	Combined active and passive
Design life (years)	60
Plant footprint (m^2)	16 800
RPV height/diameter (m)	Power capsule: 10 / 8 RPV: 4 / 2.6
RPV weight (metric ton)	Power capsule: 350 RPV: 75
Seismic design (SSE)	VIII by MSK-64 scale
Fuel cycle requirements/approach	Once-through fuel cycle
Distinguishing features	Autonomous passive decay heat removal system, integral transportable containment
Design status	Basic design

1. Introduction

The SHELF-M is a modernized NPP based on SHELF reactor unit with an increased power capacity of 35.2 MW(t) and a rated power output of up to 10 MW(e) developed based on NIKIET's experience in marine nuclear reactors design. The plant modular design and small containment vessel makes it possible to deliver reactor unit on site fully assembled and tested, minimizing construction costs and time. Refuelling and nuclear waste removal will be carried out at dedicated facilities. Optimized core design allowed to increase refuelling cycle for up to 8 years.

2. Target Application

The SHELF-M is a land-based NPP intended to supply electric power to remote areas with decentralized power supply. The plant does not require local water sources - residual and decay heat is removed by external heat exchangers cooled by atmospheric air. The SHELF-M is designed to be operated with low staffing level with only 15 staff members required to be present of site during normal operation.

3. Design Philosophy

The SHELF-M is based on proven technology derived from NIKIET's experience in design and development of marine propulsion PWR with the addition of advanced passive safety systems. Integral RPV layout with a combined forced and natural primary coolant circulation makes it possible to achieve significant reduction in power unit weight and size. RPV and all necessary normal operation and safety systems are assembled inside a cylindrical power capsule with 8 m in inner diameter and 10 m long. Capsule's shell serves as a containment. Compared to SHELF, SHELF-M has optimized core with extended fuel cycle (up to 8 years) and higher thermal capacity (35.2 MW).

4. Main Design Features

(a) Nuclear Steam Supply System

The SHELF-M power unit implements a traditional two-circuit NSSS. SG provides overheated steam to two turbines located in the engine room adjacent to power capsule room and connected to power capsule with 4 independent steam pipelines. Each pipeline is equipped with shut-off valves installed both inside and outside of power capsule and can be independently isolated without reactor shutdown in case of leaks of primary coolant into the secondary coolant.

(b) Reactor Core

The SHELF-M reactor core consists of 241 cylindrical fuel assemblies (FA) of different types with variable fuel enrichment. Fuel assemblies are placed in nodes of a regular hexagonal lattice. Control rods are located between FA and united in 8 groups. Each fuel assembly contains 55 fuel rods with cermet fuel placed in nodes of regular hexagonal lattice and 6 stationary rods of burnable absorber around fuel rods. Emergency protection rods are located inside 18 fuel assemblies in place of 19 central fuel rods. Cermet fuel consists of uranium dioxide fuel particles embedded in silicon-aluminium (silumin) matrix. Fuel rod shell is made of chromium nickel alloy. High thermal conductivity of fuel matrix as well as absence of zirconium in structural elements of fuel rod improves design safety in transients and prevents hydrogen production in steam oxidation of zirconium alloys.

(c) Reactivity Control

The reactor core contains two independent reactor shutdown systems. Control rods with boron carbide absorber at the top and dysprosium titanate at the bottom provide normal operation reactivity control and shutdown. Emergency protection rods with boron carbide ensure reliable shutdown in case of emergency. The SHELF-M is also equipped with ultimate shutdown system — emergency liquid absorber injection system (ELAIS), which is able to shutdown the core in case of severe accident by injecting boron acid in primary coolant.

(d) Reactor Pressure Vessel and Internals

The RPV accommodates the core with reactivity control rods, the primary coolant circuit, the mechanical filter, the thermal shielding, the steam generator and heat exchangers of emergency decay heat removal system. The RPV has an elliptical bottom, cylindrical shells and two rector covers (central and peripheral). All pipelines are connected to the covers at the top of the RPV to prevent the core drainage in case of LOCA.

(e) Reactor Coolant System

Reactor coolant system is based on conventional two-circuit methodology. The SHELF-M primary circuit cooling under normal operating conditions is done using forced circulation by two primary coolant circulation pumps (PCCP). Cooling by natural circulation is also possible with thermal output at the level 20 % of rated thermal power and during shutdown.

(f) Steam Generator

The SHELF-M utilises once-through steam generator with helically coiled tubes located inside the cylindrical annulus between the RPV and the core barrel. The SG comprises a tubing, collection and distribution chambers above and below the tubes, steam and feedwater lines inside the RPV. SG coil-pipes, alongside with emergency decay heat removal system (EDHRS) heat exchanger (HX) coil-pipes form a single coil with 48 independent modules grouped into 8 sections: 4 for SG and 4 for EDHRS HX, which makes it possible to cut off individual section in case of primary coolant leaks.

(g) Pressurizer

The SHELF-M adopts pressure compensation gas system common in Russian marine PWR. It utilizes an external pressurizer with no active components, such as sprinkler system or electrical heating. Pressurizer system consists of 5 separate vessels: gas collector, intermediate pressurizer, end pressurizer and two expansion vessels. Pressurizer is connected to the reactor central cover by a pipeline with a restrictor to reduce coolant leak rate in large-break accidents.

(h) Primary pumps
The SHELF-M utilizes two electrical primary coolant circulation pumps (PCCP) located on peripheral cover of the RPV.

5. Safety Features

(a) Engineered Safety System Approach and Configuration
The SHELF-M is designed with combined passive and active safety systems that comprises emergency decay heat removal system (EDHRS), emergency reactor cooling system (ERCS), emergency liquid absorber injection system (ELAIS) and overpressure protection systems.

(b) Decay Heat Removal System
During normal shutdown, decay heat is removed from the reactor core by forced circulation of primary coolant to SG and then by secondary coolant to the condenser. In case of emergency (primary or secondary circuit pumps failure, secondary circuit loss of coolant etc), emergency decay heat removal system (EDHRS) is used to cool down the core. The system consists of 4 independent loops with two passive circuits in each loop (intermediate circuit and low-boiling circuit) with natural circulation of coolants and stand-alone heat exchangers between circuits. The heat is transferred to the atmospheric air via heat exchangers, located on the roof of the reactor building.

(c) Emergency Core Cooling System
The emergency reactor cooling system is designed to supply the in-vessel circulation circuit with water during accidents with loss of the primary circuit integrity. The system is based on passive principals and doesn’t require any manual or automatic activation. The system contains 4 vessels with coolant under high pressure, which is passively injected into RPV in case of primary coolant pressure drop.

(d) Containment System
The SHELF-M utilizes two containment systems. The first one — safety vessel, located inside power capsule, that encloses RPV and all primary circuit equipment. The second — power capsule shell, which contains safety vessel with RPV, steam and feedwater pipelines, EDHRS intermediate heat exchangers, reactor unit auxiliary equipment and tanks with liquid and solid radioactive waste. Both containment system are designed to withstand pressure and temperature of primary coolant in case of large-break accidents. Both containments are unvisited during normal operation and filled with inert gas (nitrogen) with absolute pressure below atmospheric to prevent minor leaks of radioactive media and reduce the probability and possible consequences of fire inside containment volume. Both containments are equipped with overpressure protection systems.

6. Plant Safety and Operational Performances
The electric power of a single SHELF-M unit is up to 10 MW(e), and the thermal power is 35.2 MW(t). The current supplied to the consumer system is alternate and three-phase with voltage 0.4 kV ± 2 %, frequency 50 Hz ± 1 Hz). The NPP base operation mode is power operation in range from 20 to 100 % of rated power with the capability to vary the consumed power daily and annually. The power increase and decrease rate is 1 % (with forced primary coolant circulation). Design capacity factor - 0.9. Refuelling outage time is 30 days once in 8 years.

7. Instrumentation and Control System
The automated process control system (APCS) of a NPP with SHELF-M reactor unit is to control major and auxilary 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 % of rated power with forced primary coolant circulation, operation at steady power levels at 20 % of rated power with natural coolant circulation, 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 deviation beyond the safe operation limits or in the event of equipment failures leading to the safe operation limits being violated.

8. Plant Layout Arrangement
The SHELF-M plant consists of main building, administrative building, 4 stand-alone fan cooling towers, 4 emergency diesel generators, start-up boiler module, water storage tanks, fresh and spent fuel open storage area and auxiliary modules. The main building is based on lightweight steel frame structure

and houses the reactor hall with power capsule and external concrete protection, and the turbine hall. The building is 48×48×25 m is size and is designed as quick assembly building based on prefabricated structures, that allows to significantly reduce construction costs and time.

The administrative building is assembled from standard containers and houses administrative offices and control room.

9. Testing Conducted for Design Verification and Validation

Experimental activities are built and in operation. The SHELF-M reactor unit fuel rods reactor tests are under way.

10. Design and Licensing Status

Licensing process to be started in 2025.

11. Fuel Cycle Approach

The SHELF-M implements a standard PWR fuel cycle. All fuel assemblies are discharged at the end of refuelling cycle. On-site (for FOAK plant) and remote (for subsequent plants) refuelling strategies are designed. For the FOAK plant, the fresh and spent open air fuel storage is located on site.

12. Waste Management and Disposal Plan

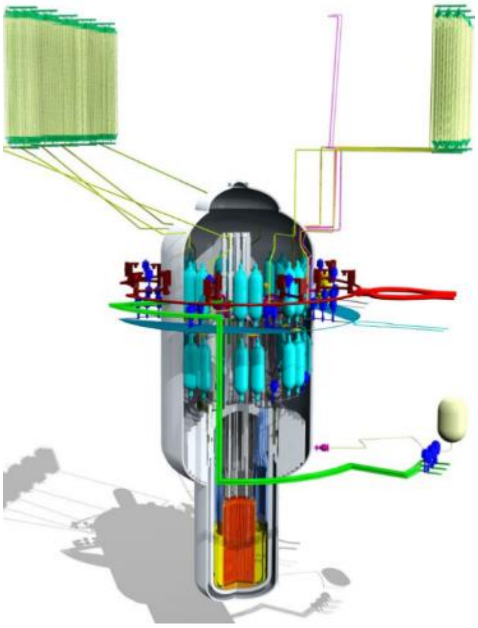
Liquid radioactive waste management system is located inside power capsule. The system is designed to gather radioactive leaks from reactor cooling systems and storage liquid waste inside power capsule.

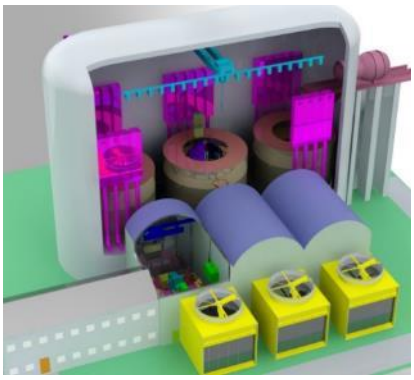
Gaseous radioactive waste system is designed to remove gaseous radioactive waste from within the power capsule during the scheduled maintenance (one a year). Nitrogen from within power capsule passes through filters, where radioactive aerosols are absorbed and stored.

13. Development Milestones

2018	Start of the project SHELF-M (achieved)
2019	Conceptual design (achieved)
2023	Preliminary design (ongoing)
2025	Licensing, detailed design, start of construction
2030	Operation testing

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 (m^2)	~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 customers 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. 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.

4. Main Design Features

(a) 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.

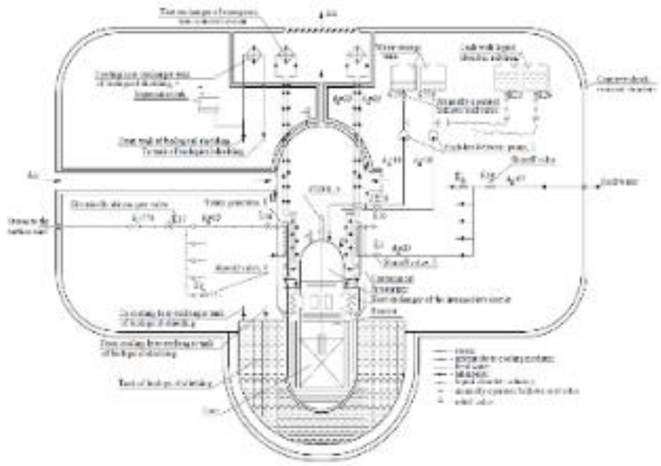
The secondary system (consumer's circuit) is intended to generate a superheated steam from the feedwater (supplied by NPP turbo generator pumps) by means of heat transfer from the intermediate circuit coolant inside the steam generator. Secondary coolant circulates by means of natural convection.

(b) 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.

(c) 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.



(d) 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).

(e) 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).

(f) 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.

(g) Pressurizer

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

5. Safety Features

The UNITHERM safety philosophy is to assure that the radiation impact on personnel, population and the environment under normal and design basis accidents is well below the limits prescribed by the current regulations. The UNITHERM design makes use of passive systems and devices based on natural processes without external energy supply. The design inherently eliminates 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.

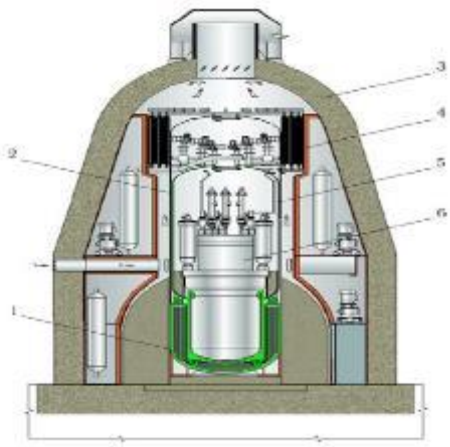
(a) Engineered Safety System Approach and Configuration

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.

6. 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.

7. 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.

8. 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.

9. 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.

10. Fuel Cycle Approach

The duration of the campaign reactor core is 15 years.

11. 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.

12. 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

ANNEX I
Acronyms

AC	Alternating Current
ADS	Automatic Depressurization System
ALARA	As Low As Reasonably Achievable
ARIS	Advanced Reactor Information System
ASEC	Air Heat Sink for Emergency Cooldown
ASME	American Society of Mechanical Engineers
ATWS	Anticipate Transient Without SCRAM
BCR	Back-up Control Room
BDBA	Beyond Design Basis Accident
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCS	Containment Cooling System
CCWS	Component Cooling Water System
CEDM	Control Element Drive Mechanism
CRDM	Control Rod Drive Mechanism
CES	Containment Enclosure Structure
CHR	Containment Heat Removal System
CIS	Containment Isolation System
CMT	Core Make-up Tank
CNPP	Cogeneration Nuclear Power Plant
CNSC	Canadian Nuclear Safety Commission
CPS	Control and Protection System
CRDM	Control Rod Drive Mechanism
CS	Containment Structure
CSS	Control Safety System
CTS	Chemical Technological Sector
CV	Containment Vessel
CVCS	Chemical and Volume Control System
DAS	Diverse Actuation System
DBA	Design Basis Accident
DC	Direct Current
DCS	Distributed Control System
DID	Defence in Depth
DLOFC	Depressurized Loss of Forced Cooling
DRACS	Direct Reactor Auxiliary Cooling System
DVI	Direct Vessel Injection
ECCS	Emergency Core Cooling System
ECDS	Emergency Cooling Down System
ECT	Emergency Cooldown Tank
EDG	Emergency Diesel Generator
EHRS	Emergency Heat Removal System
EPZ	Emergency Planning Zone
ESWS	Essential Service Water System
FA	Fuel Assembly
FBR	Fast Breeder Reactor
FE	Fuel Element
FPU	Floating Power Unit
FRPS	First Reactor Protection System (in CAREM)
FSAR	Final Safety Analysis Report
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 Reator
HEU	High Enriched Uranium
HFE	Human Factors Engineering
HLMC	Heavy Liquid Metal-Cooled
HHTS	Hybrid Heat Transport System
HPCF	High Pressure Core Flooder
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
HX	Heat Exchanger
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
IHX	Intermediate Heat Exchanger
IPIT	Intermediate Pressure Injection Tanks
I&C	Instrumentation and Control
LEU	Low Enriched Uranium
LFR	Lead-cooled Fast Reactor
LFTR	Liquid-Fluoride Thorium Reactor
LLSF	Low Level Safety Functions
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPFL	Low Pressure Core Flooder
LWR	Light Water Reactor
MA	Minor Actinides
MCR	Main Control Room
MCSFR	Molten Chloride Salt Fast Reactor
MHT	Main Heat Transport
MOX	Mixed Oxide
MSA	Moisture Separator Reheater
MSFR	Molten Salt Fast Reactor
MSR	Molten Salt Reactor
MW(e)	Mega Watt electric
MW(t)	Mega Watt thermal
NDHP	Nuclear District Heating Plant
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTEP	Nuclear Thermoelectric Plant
OBE	Operating Basis Earthquake
OCP	Outside Containment Pool
ORNL	Oak Ridge National Laboratory
OTSG	Once-Through Steam Generators
OTTO	Once Through Then Out
PAR	Passive Autocatalytic Re-Combiners
PC	Primary Containment
PCS	Primary Containment System
PCT	Peak Cladding Temperature
PCU	Power Conversion Unit
PCV	Primary Containment Vessel
PCCS	Passive Containment Cooling System
PDHR	Passive Decay Heat Removal
PGA	Peak Ground Acceleration

PHTS	Primary Heat Transport System
PLOFC	Pressurized Loss of Forced Cooling
PLS	Plant Control System
PMS	Protection and safety Monitoring System
PORV	Power-Operated Relieve Valve
PRHRS	Passive Residual Heat Removal System
PSAR	Preliminary Safety Analysis Report
PSIS	Passive Safety Injection System
PWR	Pressurized Water Reactor
RCCS	Reactor Cavity Cooling System
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RCSS	Reactivity Control and Shutdown System
RDP	Reactor automatic Depressurization System
RFA	Robust Fuel Assembly
RHRS	Residual Heat Removal System
RP	Reactor Plant
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RV	Reactor Vessel
SBO	Station Black-Out
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator
SIT	Safety Injection Tanks
SMR	Small Modular Reactor
SNF	Spent Nuclear Fuel
SSC	Systems, Structures and Components
SRPS	Second Reactor Protection System (in CAREM)
SSE	Safe Shutdown Earthquake
TC	Turbo Compressor
T/G	Turbine/Generator
TEG	Thermoelectric Generator
TEU	Thermoelectric Unit
TM	Turbo Machine
TRISO	Triple Coated Isotropic
TRL	Technology Readiness Level
UCO	Uranium Oxy Carbide
UHS	Ultimate Heat Sink
WATSS	Waste to Stable Salt
WDS	Waste Disposal System
WPu	Weapon-Grade Plutonium
WWER	Water Water Energetic Reactor (VVER in Russian)

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