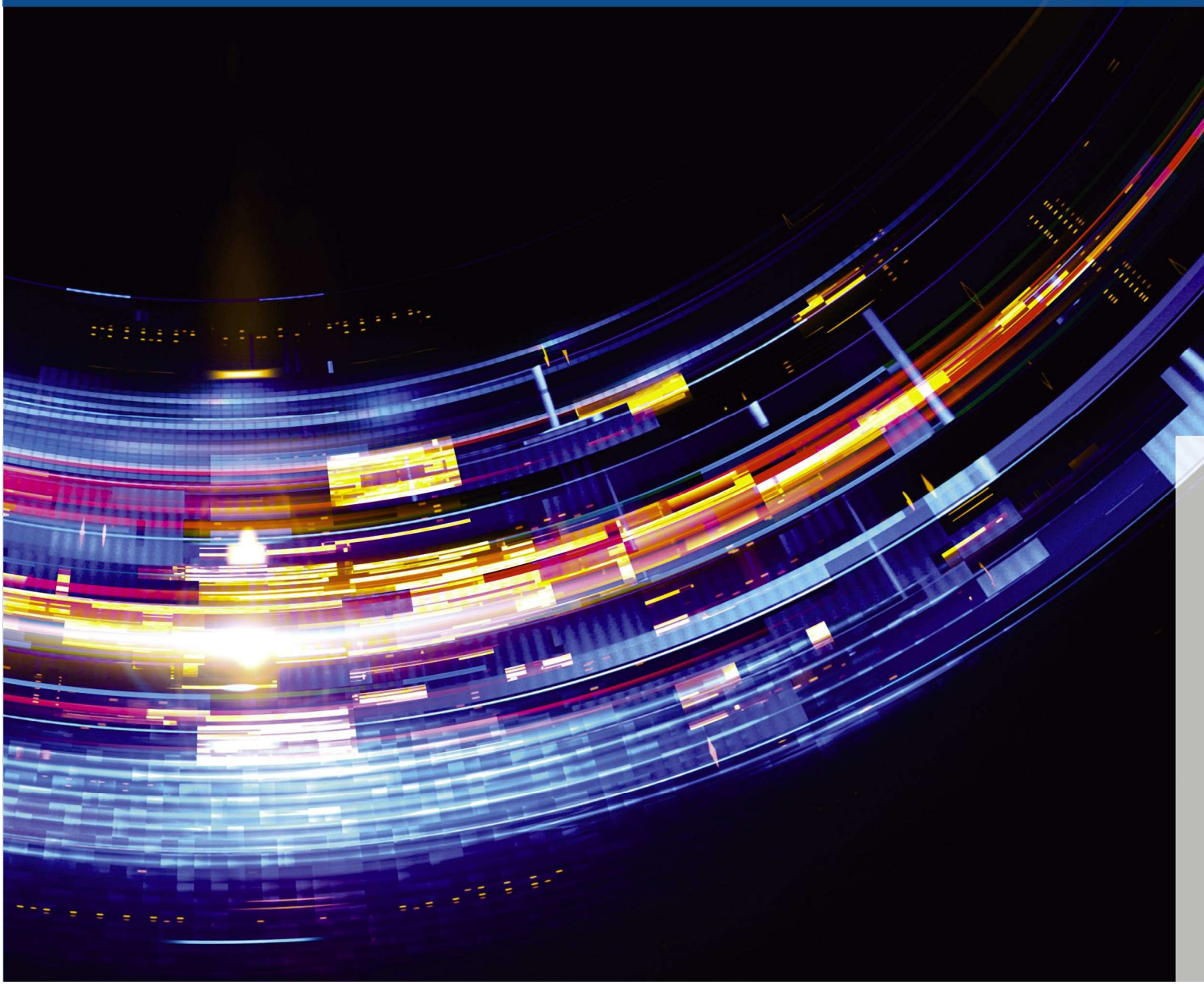
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2026

Small Modular Reactor Technology

Catalogue 2026



SMALL MODULAR REACTOR TECHNOLOGY CATALOGUE

2026 Edition

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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, molten salt 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 upfront 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, district heating or cooling and 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.

Catalogues on the status of SMR technology developments (previously called booklets until 2022) 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 catalogue 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 water-cooled reactors, marine based reactors, high temperature gas cooled reactors, liquid metal and gas-cooled fast neutron spectrum reactors, molten salt reactors, and 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 catalogue.

This SMR catalogue is designed to complete the information found in the IAEA Advanced Reactor Information System (ARIS), which can be accessed at http://aris.iaea.org. The ARIS database includes all SMR designs included in this catalogue as well as large conventional reactors. It has been upgraded in September 2024 to be searchable and incorporate comparison tools. While subsequent revision of this catalogue may occur until the next edition planned for 2028, the ARIS database will also 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 officers responsible for this publication were D. Araujo da Silva, S. Jo and B. Lepouzé of the Division of Nuclear Power.

TABLE OF CONTENTS

INTRODUCTION **1**

WATER-COOLED SMALL MODULAR REACTORS **9**

ACP100 11
AP300 17
BWRX-300 25
CAREM 31
HAPPY200 37
i-SMR 43
NuScale Power Module 51
NUWARD 59
PWR-20 65
RITM-200N 71
Rolls-Royce SMR 79
SAIR 85
SMART 95
SMR-300 103
STAR 109
WL-100 SMR 115
YL-100 121

MARITIME-BASED SMALL MODULAR REACTORS **127**

ABV-6E 129
ACP100S 135
BANDI 141
KLT-40S 147
MARINA 153
RITM-200M 159
SeaMSR-100 165
VBER-300 171

HIGH TEMPERATURE GAS-COOLED SMALL MODULAR REACTORS **177**

EM2 179
FMR 183
GTHTR300 189
HECTAR 195
HTGR 199
HTGR-POLA 205
HTMR-100 211
HTR-10 217
HTR50S 223
HTR-PM 229
HTTR 235
Jimmy 241

PeLUIt-40 247
Xe-100 253
FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS 257
ARC-100 259
AURORA 265
Blue Capsule 269
BREST-OD-300 277
HEXANA 283
LFR-AS-200 289
Natrium 295
OTRERA-300 301
SALUS 100 307
SEALER 55 319
MOLTEN SALT SMALL MODULAR REACTORS 323
CA Waste Burner 325
Flex reactor 331
IMSR400 337
JFHR 343
KP-FHR 349
LFTR 357
Stable Salt Reactor – Wasteburner 363
Stellarium 369
ThorCon 375
Thorizon 381
MICROREACTORS 387
AMR (Power Cell Microreactor) 389
CALOGENA 395
ELENA 401
Energy Well 407
eVinci 413
HARPIA 419
HOLOS-MONO 427
HOLOS-QUAD 435
JMR 445
Kronos MMR 451
Loki MMR 461
Marvel 468
MoveLux 475
SHELF-M 481
Unitherm 487
Zeus 493
ANNEX ACRONYMS 497

INTRODUCTION

Within the Nuclear Power Division of the IAEA Department of Nuclear Energy, the Nuclear Power Technology Development Section (NPTDS) supports Member States in identifying key enabling technologies for advanced reactor concepts and addressing the challenges associated with their near-term deployment. To achieve this, the Section establishes international expert networks, coordinates collaboration among Member State specialists, and develops publications, recommendations, and guidance documents tailored to Member States' needs.

Meeting growing energy demand, strengthening energy security and addressing climate change will require the deployment of all available reliable and affordable energy sources. Renewable energy technologies such as wind and solar power are expected to play an increasingly important role, but they must be complemented by reliable sources of clean electricity capable of operating under a wide range of system conditions. Nuclear power provides low-carbon, dispatchable electricity and can contribute to grid stability, energy security, and economic competitiveness. Alongside hydropower, nuclear power is one of the largest sources of low-carbon electricity worldwide and can support the integration of variable renewable energy while helping maintain a diverse and resilient energy supply.

Small modular reactors (SMRs) may play an important role in the transition to low-carbon energy systems. Their smaller unit size makes them suitable for countries with smaller electricity grids, while their lower upfront capital investment requirements may improve financing prospects and facilitate incremental capacity additions. Their modular design may also enable shorter construction schedules and greater opportunities for factory fabrication. In addition, the diversity of SMR technologies under development expands their potential applications beyond electricity generation, including the provision of industrial process heat and other hard-to-abate energy services. Since 2012, the IAEA has supported Member States by monitoring and documenting SMR developments worldwide through its SMR catalogue issued every two years, which reflects the growing number and diversity of SMR designs.

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 (marine-based SMRs and microreactors) which can rely on any of the abovementioned technology lines.

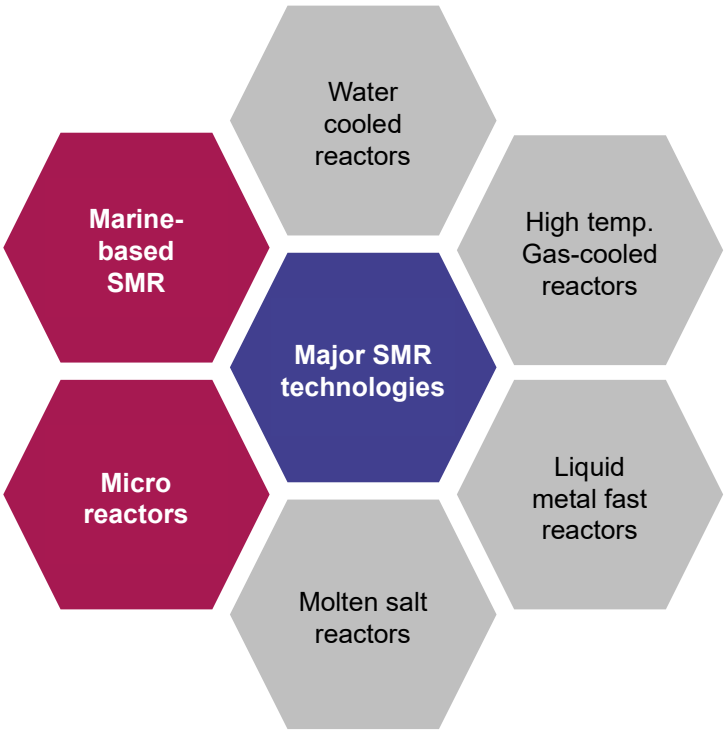


Figure 1: Major technology lines of the SMR technologies

Since 2012, the SMR catalogue has been listing an increasing number of designs, with a peak of eighty-three designs in 2022. Back in 2024, only so-called active designs have been kept, for which the technology developers would agree to provide an update on their development plan. This led to a selection of sixty-nine designs, which was later increased to seventy-one designs in the third edition of the 2024 SMR catalogue (issued in June 2025). 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.

For the 2026 SMR catalogue, it has been decided to provide more guidance to the technology developers and give them concrete milestones to assess their status of development against. This allows the reader to judge by oneself at what stage the design really is. It has also been decided to keep all the designs in the catalogue, but to indicate clearly which ones have been updated since the 2024 edition. As a result, the number of designs will increase again, but it will be easier to identify the progress made by each design. As an added benefit, this will also better synchronize the content of the SMR catalogue with the ARIS database, which also includes legacy designs.

The development phases communicated to the technology developers are shown in Figure 2 below:

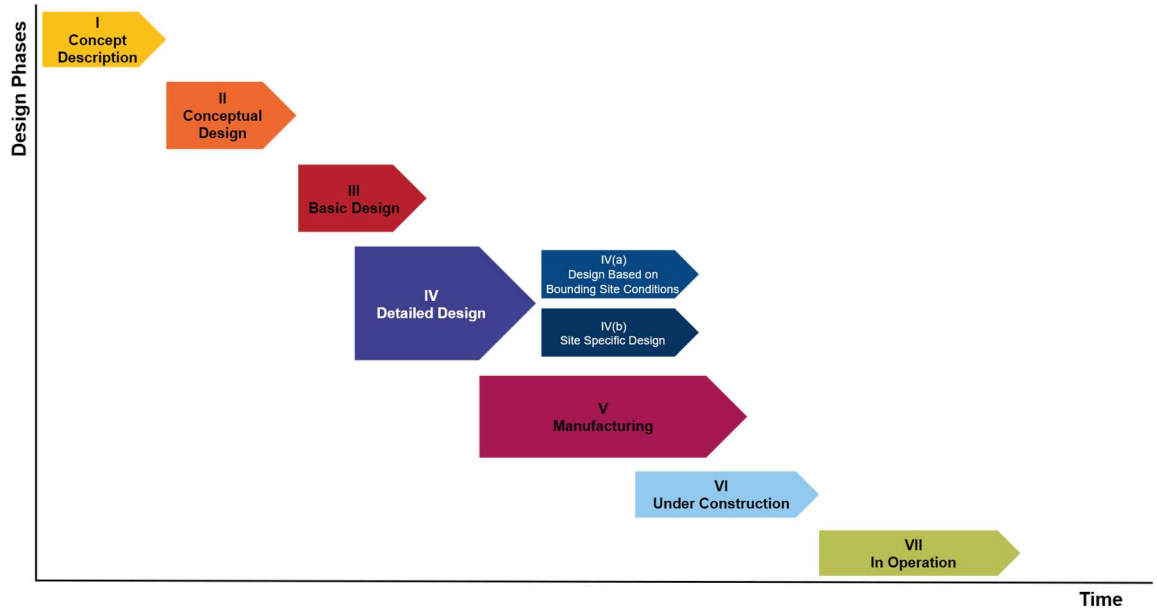


Figure 2: Development phases of an SMR design

The criteria to assess the development phase of a design are listed below:

TABLE 1. CORRESPONDENCE BETWEEN ARIS AND DEVELOPMENT STATUS IN THE SMR CATALOGUE

Status in ARIS	Development status	Criteria
Conceptual design	Concept description	- Basic ideas and goal described - Few calculations/sketches/data
	Conceptual design	- Development and test needs identified - Rough estimations of cost and schedule - Key components and layout drawings - Single line diagrams - Brief description of key components and systems

Basic design	Basic design	- Identification and preliminary analysis of concept relevant incidents and accidents - System description for all NPP systems - Safety analysis needed for a design approval (GDA) completed - Updated estimates of cost and schedule - Technical specifications and documentations for major components and systems
Detailed design	Detailed design (either for boundary site conditions or for a specific site)	- Itemized cost estimation and master schedule prepared - Equipment specifications and equipment drawings - Construction-ready design (Issued-for-construction "IFC" design) - Preliminary Safety Analysis Report (PSAR) - Maintenance plan
Under construction	Manufacturing	- Procurement - In-process inspection and quality control - Factory acceptance tests - End-of-manufacturing report - Packaging, handling and transportation - Prototype module testing
	Under construction	- First concrete (when applicable)
In operation	In operation	- Commissioning - First criticality

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

Part One: Water-cooled SMRs. This part presents the key water-cooled SMR designs from various configurations of light water reactor technology for on-shore grid-connected applications. These designs represent the mature technology considering most of the large power plants in operation today are of water-cooled reactors. There are seventeen (17) water-cooled SMR designs from 12 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 SMRs. This part presents concepts that can be deployed in a maritime environment, as barge-mounted floating power unit or on nuclear-powered ships. This unique application provides many flexible deployment options. This catalogue presents eight (8) maritime-based 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 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 ten (10) 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: Microreactors. 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 or for mission-critical applications that for decades have been served by diesel power plants. sixteen (16) microreactor designs are included and detailed in this catalogue.

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 catalogue for reason of completeness and since designers foresee certain niche markets for their products.

The designs listed in this publication are summarized below. A small blue dot appears next to the designs which have been updated since 2024 catalogue:

TABLE 2: DESIGN AND STATUS OF SMRS INCLUDED IN THIS CATALOGUE

Design	Output MW(e)	Type	Designers	Country	Status
PART 1: WATER COOLED SMALL MODULAR REACTORS					
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	Under construction
CAREM	30	PWR	CNEA	Argentina	Under construction
HAPPY200	200 MWth	PWR	SPIC	China	Detailed Design
i-SMR •	170	PWR	KHNP & KAERI	Republic of Korea	Detailed Design
NuScale Power Module •	77	PWR	NuScale Power Inc.	United States of America	Under construction
NUWARD •	400	PWR	EDF Group, Tractebel, Ansaldo Nucleare	France	Conceptual Design
PWR-20	20	PWR	Last Energy	United States of America	Detailed design
RITM-200N •	55	PWR	JSC "Afrikantov OKBM"	Russian Federation	Under construction
Rolls-Royce SMR •	470	PWR	Rolls-Royce	United Kingdom	Detailed design
SAIR •	45	PWR	Saiheat	Singapore	Conceptual design

SMART •	107	PWR	KAERI and K.A.CARE	Republic of Korea, and Saudi Arabia	Detailed design
SMR-300 •	320	PWR	Holtec International	United States of America	Detailed design
STAR	10	PTWR	Star Energy	Switzerland	Basic design
WL-100 SMR •	100	iPWR	Shanghai Weilan NuEnergy	China	Conceptual design
YL-100 •	80	iPWR	Shanghai Universal Nuclear Energy Technology Co., Ltd.	China	Conceptual design
PART 2: MARITIME-BASED SMALL MODULAR REACTORS					
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	Basic design
KLT-40S	2 × 35	Floating PWR	JSC Afrikantov OKBM	Russian Federation	In Operation
MARINA •	35	Floating MSR	KAERI	Republic of Korea	Conceptual design
RITM-200M •	50	Floating PWR	JSC Afrikantov OKBM	Russian Federation	Under construction
SeaMSR-100 •	110	Floating MSR	SaltFoss Energy	Denmark	Basic 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
HECTAR •	90MWth	HTGR	KAERI	Republic of Korea	Basic design
HTGR •	200	HTGR	JSC Afrikantov OKBM	Russian Federation	Detailed design
HTGR-POLA •	11.5	HTGR	NCBJ	Poland	Basic design
HTMR-100 •	35	HTGR	STL Nuclear	South Africa	Basic design
HTR-10 (test reactor)	2.5	HTGR	INET, Tsinghua University	China	Operational
HTR50S	17.2	HTGR	JAEA	Japan	Conceptual design
HTR-PM	210	HTGR	INET, Tsinghua University	China	In operation
HTTR (test reactor)	30MWth	HTGR	JAEA	Japan	In operation
Jimmy •	60MWth	HTGR	Jimmy Energy	France	Detailed design
PeLUIt-40	10	HTGR	BRIN	Indonesia	Basic design
Xe-100	82.5	HTGR	X-Energy LLC	United States of America	Under construction

PART 4: FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS					
ARC-100 •	100	Sodium-cooled	ARC Nuclear Canada, Inc.	Canada	Basic design
AURORA •	75	Sodium-cooled	Oklo	United States of America	Under construction
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
Natrium	345	Sodium-cooled	Terrapower	United States of America	Under construction
OTRERA 300 •	110	Sodium-cooled	Otrera Energy	France	Basic design
SALUS 100 •	100	Sodium-cooled	KAERI	Republic of Korea	Conceptual design
SEALER-55 •	55	Lead-cooled	BlyKalla	Sweden	Conceptual design
PART 5: MOLTEN SALT SMALL MODULAR REACTORS					
CA Waste Burner	100 MWth	MSR	Copenhagen Atomics	Denmark	Detailed design
Flex Reactor	24	MSR	Moltex Energy	United Kingdom	Conceptual design
IMSR 400	195	MSR	Terrestrial Energy Inc.	Canada	Detailed design
JFHR •	62	Prismatic salt-cooled reactor	Jereh	United Arab Emirates	Conceptual design
KP-FHR •	140	Modular Pebble-bed HTR Salt Cooled Reactor	KAIROS Power, LLC.	United States of America	Under construction
LFTR •	250	MSR	Flibe Energy	United States of America	Conceptual design
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 of America	Conceptual design
Thorizon •	100	MSR	Thorizon BV	Netherlands / France	Basic design
PART 6: MICROREACTORS					
AMR (Power Cell Microreactor)	3	HTGR	Power Cell Micro Reactor Pty	South Africa	Conceptual design
CAL-30 •	30MWth	Pool type	Calogena S.A.	France	Basic design
ELENA	68 kWe	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 construction
HARPIA	5	Heat pipe	Terminus R&D	Brazil	Basic design
HOLOS-MONO •	10	HTGR	HolosGen	United States of America	Basic design
HOLOS-QUAD •	10	HTGR	HolosGen	United States of America	Under construction
JMR •	5	Prismatic salt-cooled reactor	Jereh	United Arab Emirates	Conceptual design
Kronos MMR •	3.5	HTGR	Nano Nuclear Energy	United States of America	Basic design
Loki MMR •	1MWth	HTGR	Nano Nuclear Energy	United States of America	Conceptual design
Marvel*	15 kWe	LM thermal	Idaho National Labs	United States of America	Under construction
MoveluX	3	Heat Pipe	Toshiba Corporation	Japan	Conceptual design
SHELF-M	10	iPWR	NIKIET	Russian Federation	Detailed design
Unitherm	6.6	PWR	NIKIET	Russian Federation	Conceptual design
Zeus •	100kWe	Solid core	Nano Nuclear Energy	United States of America	Conceptual design

* Blue capsule and Marvel designs are metal-cooled but are thermal spectrum reactors.

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

CNNC, China
ACP100



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*CNNC (NPIC), China*
Reactor type	*Integral PWR*
Coolant/moderator	*Light Water / Light Water*
Design status	*Construction in progress*
Thermal/electrical capacity, MW(t)/MW(e)	*358 / 125*
NSSS operating pressure (primary/secondary), MPa	*15 / 4.6*
Core inlet/outlet coolant temperature (°C)	*286.5 / 319.5*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO_2 / 17 X 17 square pitch*
Number of fuel assemblies in the core	*57*
Fuel enrichment (%)	*< 4.95*
Core discharge burnup (GWd/ton)	*< 52*
Refuelling cycle (months)	*24*
Fuel cycle requirements/approach	*Temporarily stored in spent fuel pools*
Reactivity control mechanism	*Control rod drive mechanism (CRDM), Gd2O3 solid burnable poison and soluble boron acid*
Approach to safety systems	*Passive*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*200,000*
RV height/diameter (m)	*10 / 3.35*
RV weight (metric ton)	*300*

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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

☐ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☒HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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 235U 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 Vessel (RV) 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 and Steam Generator

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) Pressuriser

The pressurizer of ACP100 is located outside of the reactor vessel. The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads.

(g) 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.

(h) Distinguishing features

5. Safety Features

(i) 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.

(j) 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.

(k) 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.

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) 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.

(o) 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.

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 Systems

The Instrumentation and Control (I&C) system designed for ACP100 is based on defence in depth concept, compliance with the single failure criterion and diversity. The diversity in the design of I&C system is achieved through: (1) different hardware and software platforms for 1E and N1E I&C, (2) reactor protection system (RPS) with functional diversity, and (3) diverse protection systems to cope with the common mode failure of the RPS. I&C systems of the NSSS include reactor nuclear instrumentation system, RPS, diverse actuation system, reactor control system, rod control and rod position monitoring system, reactor in–core instrumentation system, loose parts and vibration monitoring system and other process control systems.

8. Plant Layout Arrangement

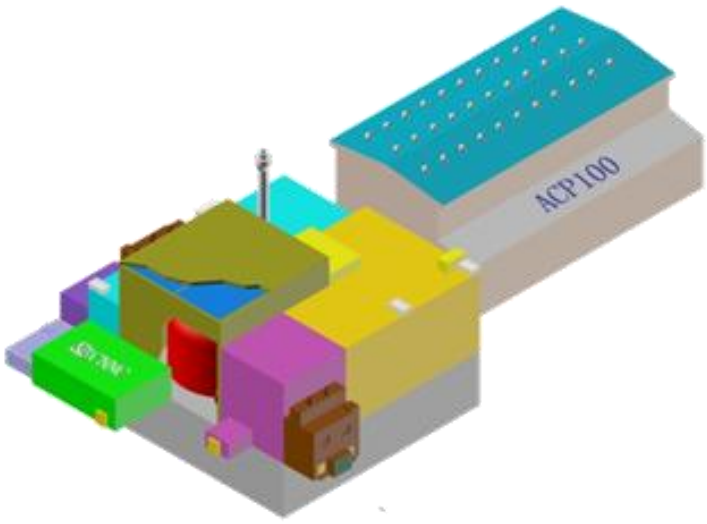


Figure 1. Plant Layout Arrangement of ACP100

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

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

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.



Figure 2. Site Construction 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

Year	Milestone	Status
2016	Generic reactor safety review for ACP100 by IAEA	**Completed**
2017	CNNC signed an agreement with the Changjiang municipal government in Hainan Province to host the first of a kind (FOAK) ACP100 demonstration unit.	**Completed**
2018	Preliminary safety assessment report (PSAR) finished.	**Completed**
2019	PSAR submitted to National Nuclear Safety Authority, Site Preparation started.	**Completed**
2020	Apply for authorize to Changjiang nuclear power site, Hainan, China.	**Completed**
2021	FCD on Changjiang nuclear power site, Hainan, China.	**Completed**
2022	Star nuclear island installation.	**Completed**
2023	The core equipment (RPV and OTSG) hoisted into position	**Completed**
2024	The outer dome of containing vessel hoisted into position, begin cold test.	**Completed**
2026	Target commercial operation.	**Ongoing**

Westinghouse Electric Company LLC, United States of America AP300TM SMR

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Figure 1: AP300 SMR containment vessel peel-away

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Westinghouse, United States of America*
Reactor type	*PWR*
Coolant/moderator	*Water*
Design status	*Basic design*
Thermal/electrical capacity, MW(t)/MW(e)	*990/330*
NSSS operating pressure (primary/secondary), MPa	*15.4/7.06*
Core inlet/outlet coolant temperature (°C)	*297/325*
Primary circulation	*Forced with 2 Reactor Coolant Pumps*
Fuel type/assembly array	*UO_2 fuel / 17 X 17 square lattice*
Number of fuel assemblies in the core	*121*
Fuel enrichment (%)	*< 5 wt. % U-235 in U*
Core discharge burnup (GWd/ton)	*38 GWd/ton*
Refuelling cycle (months)	*36 (Up to 48)*
Fuel cycle requirements/approach	*e.g., open, once-through, take-back*
Reactivity control mechanism	*Rods, soluble boron*
Approach to safety systems	*Passive (as in the AP1000)*
Seismic design (SSE)	*design basis safe shutdown earthquake 0.3g pga*
Design life (years)	*80*
Plant footprint (m^2)	*8300 m2 (Power block buildings, bounding value)*
RV height/diameter (m)	*11/4*
RV weight (metric ton)	*275*

1. Introduction

Westinghouse's 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

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. Plant operability and maintainability is a key focus for the AP300 SMR design, which improves operation while reducing the number of components and associated maintenance requirements. In particular, simplified safety systems reduce surveillance requirements by enabling significantly simplified technical specifications. Selection of proven components ensures a high degree of reliability with low maintenance requirements.

Additionally component standardization reduces spare parts, minimizes maintenance, reduces training requirements and allows shorter off-line maintenance duration.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☒HTSE - electrolysis	☒ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

The AP300 SMR is designed to provide versatile energy output beyond electricity generation, enabling efficient cogeneration applications such as district heating, industrial process heat, desalination, and hydrogen production. Its flexible performance and capability to deliver high quality steam allow integration with district heating networks and industrial facilities requiring consistent thermal energy, supporting both low- and high-temperature heat demands. The plant's inherent load-following capability facilitates simultaneous production of electricity and heat, optimizing overall energy utilization and improving economic viability by supplying multiple energy products from a single asset. This versatility allows deployment across a wide range of sites and use cases, including urban heating systems, industrial decarbonization, and integrated energy systems, while maintaining reliable grid support and operational flexibility.

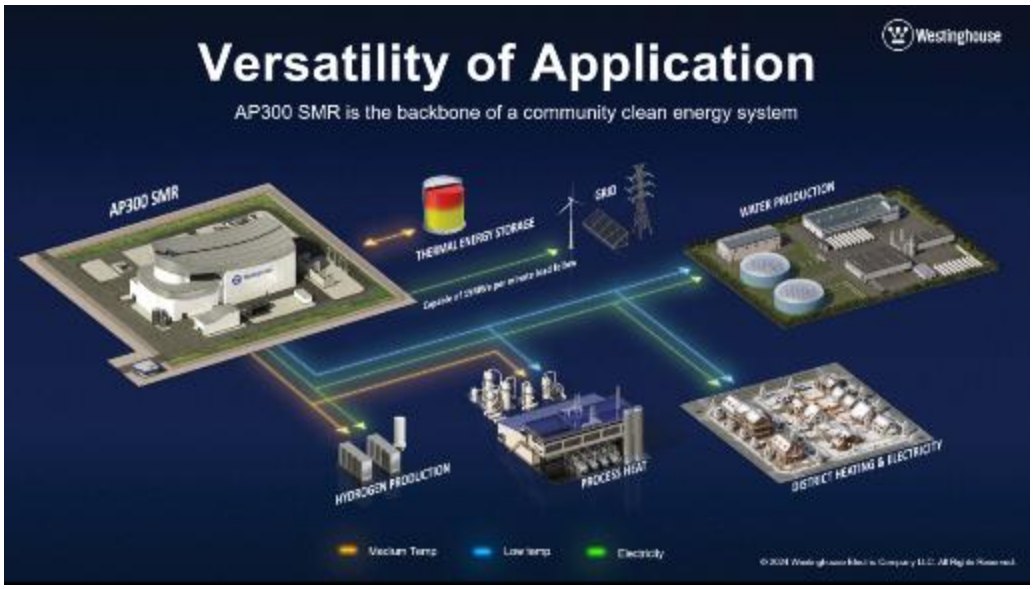


Figure 1: AP300 SMR Applications

3. Design Philosophy

The AP300 SMR has a 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 mid to late 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 plant 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 95%;

- 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 Reactor Coolant Pumps (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 configuration 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. The core is designed to leverage current licensed fuel and has capability to adapt to higher enrichment and higher burnup fuels as technology changes over the life of the plant.

(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, 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 SMR. Additionally, the MSHIM strategy allows to perform load following maneuvers without changes to the soluble boron concentration during the maneuver, which extends load following capabilities and reduces the burden on the Chemical and Volume Control System and related effluents.

(d) Reactor Vessel (RV) and Internals

The AP300 SMR reactor vessel is a robust, cylindrical, high pressure component that houses the reactor core and primary coolant, designed based on proven Westinghouse pressurized water reactor technology with a hemispherical bottom head and a removable flanged upper head to facilitate refueling and maintenance. It contains the nuclear fuel assemblies, control rod cluster assemblies, and associated core support structures, all configured to ensure reliable heat transfer and structural integrity under normal and transient conditions. The vessel and internals are fabricated from materials qualified to established nuclear codes, with stainless steel cladding applied to wetted surfaces for corrosion resistance and are designed to operate at high temperature and pressure while maintaining safety margins. The internals also incorporate features to support efficient coolant flow, neutron moderation, and reactivity control, ensuring stable core performance and long-term durability consistent with the plant's extended design life.

(e) Reactor Coolant System and Steam Generator

The AP300 SMR utilizes a single-loop reactor coolant system design derived from proven pressurized water reactor technology, consisting of a reactor vessel connected to one steam generator through a hot leg, with cooled coolant returning via two cold legs and circulated by reactor coolant pumps mounted at the base of the steam generator. The steam generator is a vertical U-tube design, based on established Westinghouse technology, and is engineered for high reliability and maintainability, incorporating features such as advanced materials and accessible tube configurations to support inspection and servicing.

(f) Pressuriser

The AP300 SMR pressurizer is a separate vessel in the reactor coolant system that maintains stable pressure within the reactor coolant system, ensuring reliable plant operation and minimizing transients. The AP300 SMR pressurizer is identical to the AP1000 plant pressurizer. This relatively large pressurizer enhances operating margins by providing

greater capacity to absorb pressure fluctuations which eliminates the need for fast acting power operated relief valves, reduces the likelihood of reactor trips and improves overall plant reliability.

(g) Primary pumps

The AP300 SMR reactor coolant pumps are sealless motor-type pumps. The pumps used are the same as the proven AP1000 plant pumps. Two reactor coolant pumps are used in the AP300 SMR design, mounted directly on the steam generator channel head.

(h) Distinguishing features

N/A

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The Westinghouse AP300 SMR plant design follows the AP1000 plant approach to safety, which 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.

Similar to the AP1000 plant design, the AP300 SMR has a Passive Core Cooling System (PXS) and Passive Containment Cooling System (PCS), as shown in Figure 3. Like the AP1000 plant, these passive systems 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.

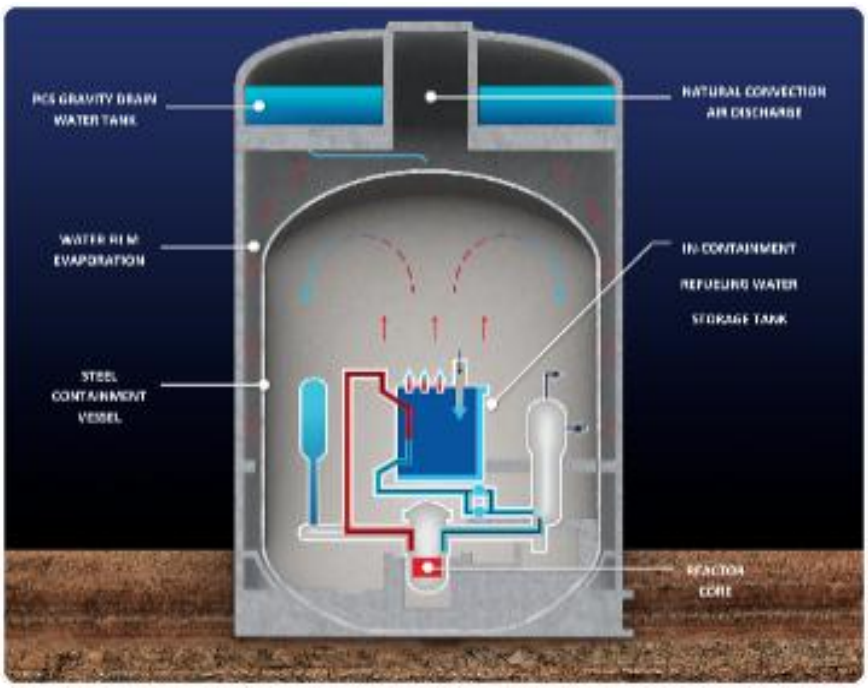


Figure 2: AP300 Containment Overview

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

The AP300 SMR safety approach for design basis conditions, including loss-of-coolant accidents (LOCAs) and station blackout (SBO), is centered on the use of advanced passive safety systems. For design basis accidents such as a LOCA, the PXS provides emergency core cooling and decay heat removal using natural circulation, gravity, and compressed gas, delivering makeup and injection from core makeup tanks, accumulators, and the in-containment refueling water storage tank without reliance on pumps, diesel generators, or AC power, while the passive residual heat removal heat exchanger ensures continued heat removal.

In parallel, the PCS removes heat from containment through gravity-driven water flow and natural air circulation to maintain containment integrity. The design explicitly addresses SBO conditions by assuming complete loss of offsite and onsite AC power in safety analyses, with passive systems automatically establishing and maintaining safe shutdown, core cooling, and containment cooling for at least 72 hours without operator action, supported only by minimal DC power for instrumentation and control. These passive systems are self-actuating, fail-safe, and rely on natural driving forces, significantly reducing dependence on active equipment and human intervention.

Active non-safety systems and diverse actuation systems provide additional layers of defense-in-depth but are not required to achieve or maintain safe shutdown.

(m) Safety Approach and Configuration to Manage DEC

For design extension conditions, the AP300 SMR incorporates a comprehensive severe accident management approach based on passive safety and defense-in-depth principles. The design includes provisions for in-vessel retention to maintain molten core debris within the reactor vessel and preserve containment integrity in the event of core melt, thereby reducing the likelihood of ex-vessel phenomena.

Passive systems such as the PXS and PCS continue to provide decay heat removal and containment heat rejection using natural forces, supporting long-term accident mitigation without reliance on AC power or operator action. Hydrogen generation and related phenomena are addressed through containment design, analysis, and mitigation strategies to ensure that combustion or pressure effects do not compromise containment.

Overall, the design aims to limit and practically eliminate large or early radioactive releases through a combination of inherent safety features, passive systems, and diversified mitigation measures.

(n) Containment System

Notably, in the AP300 SMR, the fuel storage area is located with fuel racks located inside the Containment Building (Figure 3) to leverage the water in the In-Containment Refueling Water Storage Tank (IRWST) for accident mitigation. As a result, the AP300 SMR does not have a spent fuel pool located outside of containment. This provides additional protection from external hazards since stored spent fuel is located within the robust Containment and Shield Building structure.Spent Fuel Cooling Safety Approach / System

(o) Spent Fuel Cooling Safety Approach / System

6. Plant Safety and Operational Performances

The AP300 SMR is designed to achieve high operational reliability with expected plant availability targeting greater than 95%, supported by simplified systems, short refueling outages on a three-year fuel cycle, and reduced reliance on active components, which lowers failure modes and maintenance requirements.

The design also provides strong load-following capability, enabling flexible operation with ramp rates on the order of 5 percent power per minute and routine power maneuvering between full power and reduced output to accommodate grid demand without significant impact on fuel performance.

Safety objectives are based on a robust defense-in-depth philosophy combining passive safety systems, active backup systems, and probabilistic risk optimization to maintain very low core damage and large release frequencies. This is shown with the AP1000 plant probabilistic risk assessment results which show minimized likelihood of radiological consequences due to the passive safety systems and defense-in-depth approach taken by the design.

7. Instrumentation and Control Systems

The 4-division AP300 SMR protection and safety monitoring system (PMS) utilizes the Common QTM platform. The Common Q platform is the standard Westinghouse safety system platform previously approved by the U.S. 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 AP300 SMR also employs a diverse actuation system (DAS) that uses a platform separate from the PMS platforms using independent and dedicated sensors diverse from the Common Q safety system. The non-safety platform is diverse from the Common Q safety platform, and DAS.

8. Plant Layout Arrangement

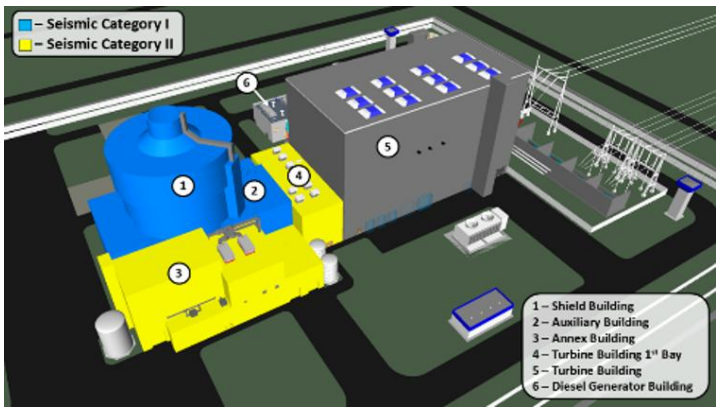


Figure 4: AP300 SMR overall plant arrangement

The AP300 SMR overall plant arrangement (Figure 4) 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 plant 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 AP600TM and AP1000 plant Design Program to provide input for plant design and to demonstrate the feasibility of unique design features. Tests for the AP600/AP1000 plant 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. Nuclear Regulatory Commission (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 with the scale model testing performed during the AP1000 plant development effort, providing further evidence to support the AP300 SMR licensing.

10. Design and Licensing Status

The AP300 SMR is using a proven technology based on the AP1000 plant. So far (in 2026) six AP1000 units are operational resulting in over 35 reactor years of demonstrated operational excellence and their design, construction, testing data, licensing, and operational experience feed into the AP300 SMR program. The AP300 SMR design complies with all international design and licensing regulations including U.S., EUR and URD. Lessons learned from the AP1000 plant licensing process with the U.S. NRC, China National Nuclear Safety Administration (NNSA), Generic Design Assessment in the United Kingdom and Pre-project design review by Canadian Nuclear Safety Commission 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 with submittal of a Standard Design Approval application to the U.S. NRC in 2027. After application submittal, the AP300 SMR will continue plant and site-specific design development to support construction for a first projected COD in the mid-late 2030s.

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 SMR fuel assembly design that improve fuel performance, enhance fuel reliability, enable better fuel cycle economics and provide additional margin.

12. Waste Management and Disposal Plan

The AP300 SMR employs a well-established spent fuel and radioactive waste management approach consistent with the current light water reactor fleet.

Spent fuel is initially stored in a spent fuel pool located inside the Containment Building, providing enhanced safety, shielding, and cooling while limiting fuel handling. After the appropriate cooling period, the spent fuel assemblies are transferred during refueling outages to dry storage canisters and loaded into dry casks using dedicated fuel handling systems. These casks are then placed in an interim spent fuel storage installation (ISFSI) located on-site or off-site for long-term interim storage depending on national policy.

In addition to spent fuel management, the plant incorporates standard gaseous, liquid, and solid radioactive waste processing systems to treat, reduce, and control radioactive effluents in accordance with regulatory requirements.

13. Development Milestones

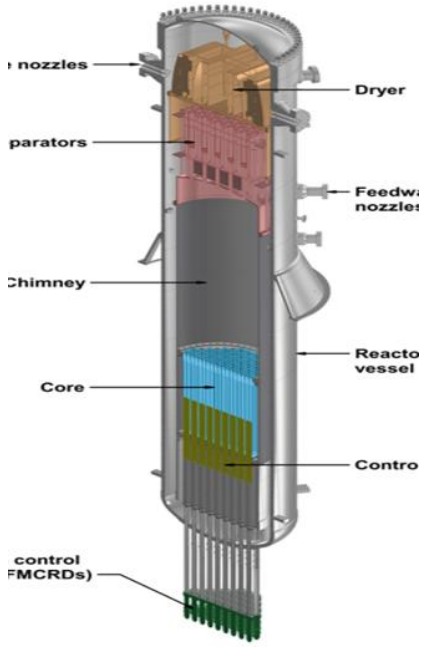
Year	Milestone	Status
2023	Conceptual Design Complete, U.S. NRC pre-application engagement initiated	**Completed**
2024-2026	U.S. NRC pre-application engagement, white papers / topical report planning, and Regulatory Engagement Plan updates	**Ongoing**
2027	Target Standard Design Approval application submittal	**Planned**
2028 through early 2030's	U.S. NRC review; and continued design development; site-specific design, licensing, financing, procurement, execution	**Planned**

Year	Milestone	Status
Mid-late 2030’s	Projected Commercial Operation Date, dependent on site and customer specifics	**Planned**

GE Vernova Hitachi Nuclear Energy, USA and Japan BWRX-300



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*GE Vernova Hitachi Nuclear Energy, USA and Japan*
Reactor type	*BWR*
Coolant/moderator	*Light Water / Light Water*
Design status	*Under construction*
Thermal/electrical capacity, MW(t)/MW(e)	*870/300*
NSSS operating pressure (primary/secondary), MPa	*7.17(Primary Only)*
Core inlet/outlet coolant temperature (°C)	*240 / 288*
Primary circulation	*Natural circulation*
Fuel type/assembly array	*UO_2 / square lattice*
Number of fuel assemblies in the core	*240*
Fuel enrichment (%)	*< 4.95*
Core discharge burnup (GWd/ton)	*< 65.0*
Refuelling cycle (months)	*12-24*
Fuel cycle requirements/approach	
Reactivity control mechanism	*Control rod*
Approach to safety systems	*Passive Defense-in-Depth approach with passive safety systems*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*11,850*
RV height/diameter (m)	*27 / 4*
RV weight (metric ton)	*650*

1. Introduction

The GE Vernova 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 an electricity-production only configuration, approximately 300 Mwe, net. The first BWRX-300 unit is under construction in Canada with a commercial operation target of 2030. The BWRX-300 is a tenth generation Boiling Water Reactor (BWR) and continues the multi-generational journey of simplification in design, 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

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

Target applications for the BWRX-300 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 aims to minimize construction, operation, maintenance, and decommissioning costs. It is designed to provide cleaner, flexible energy generation that is cost-competitive with natural gas fired plants. The BWRX-300 builds on the success and lessons learned from over 60 years of BWR operating history. The important characteristics of the BWRX-300 include:

- 10th generation BWR
- Evolved from USNRC licensed ESBWR
- Design-to-cost approach
- Significant capital cost reduction per MW
- World class safety
- Capable of load following
- Suitable for electricity generation and industrial applications
- Modularized constructability integrated into design
- Reduced on-site staff and security
- Intended to be licensable internationally

The BWRX-300 improves innovation with technology readiness. It relies on fuel that is already licensed, currently in use globally, and used by the majority of the BWR fleet. The design improves material selection and manufacturing techniques while incorporating breakthrough passive and simple concepts. The result is a cost-effective advanced reactor design with world-class safety and economic performance that can be licensed and constructed in the near-term. It offers low risk in comparison to historical large Light Water Reactor (LWR) projects in the U.S., Europe, and Asia, and is highly competitive in the worldwide energy industry.

4. Main Design Features

(a) Nuclear Steam Supply System

The 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

The 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

The 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 Vessel (RV) and Internals

The 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

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

(g) Primary pumps

The BWRX-300 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.

(h) Distinguishing features

The BWRX-300 is a simplified BWR design with natural circulation providing the motive force for coolant flow. It includes passive safety features (i.e., Isolation Condenser System and Passive Containment Cooling System). Its simplified design and modularized construction make it cost-competitive with natural gas plants.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The 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 target core damage frequency of less than 1E-6 per year and a large release frequency of less than 1E-7 per year for the reference site, supported by robust emergency planning and risk assessments.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

The 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.

(m) Safety Approach and Configuration to Manage DEC

The 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.

(n) Containment System

The 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.

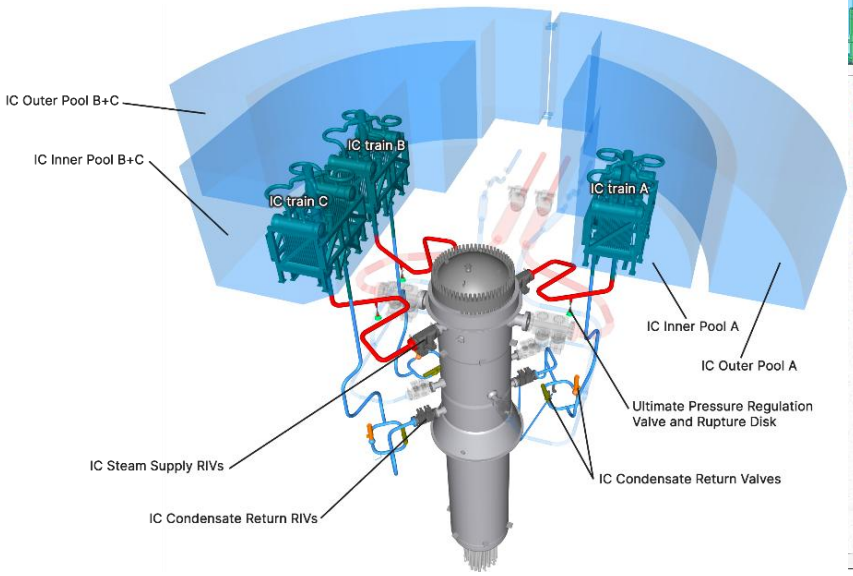


Figure 1. Isolation Condenser System

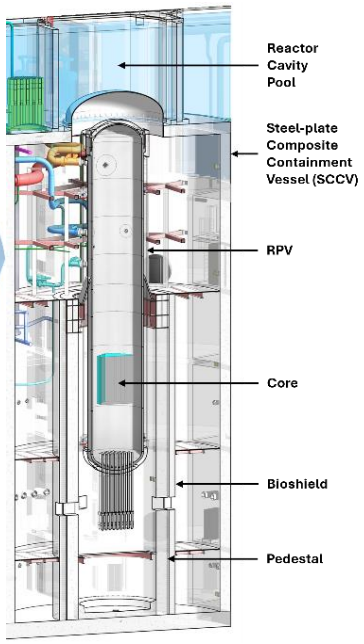


Figure 2. Dry Containment

(o) Spent Fuel Cooling Safety Approach / System

The 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

The 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 target Core Damage Frequency (CDF) of <1E-6 and Large Release Frequency (LRF) of <1E-7 per year for the reference site.

7. Instrumentation and Control Systems

The 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

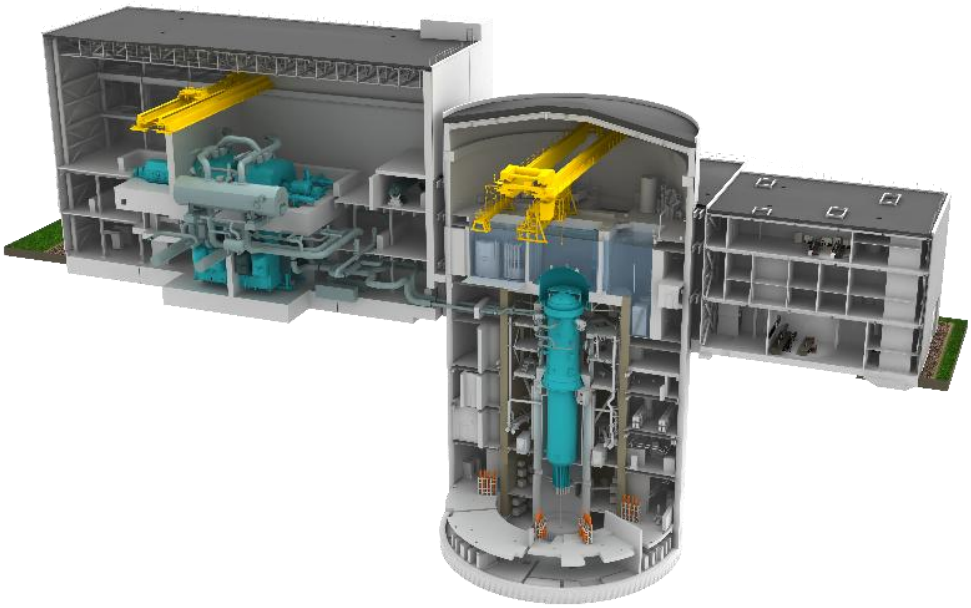


Figure 3. BWRX-300 Plant Layout

The 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.

9. Testing Conducted for Design Verification and Validation

The 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

The BWRX-300 is in various stages of construction, 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 issued in early 2025. Construction in Canada began May 2025. In the U.S., seven Licensing Topical Reports have been approved. The first U.S. Construction Permit Application was submitted in mid-2025 with issuance expected mid-2026. The BWRX-300 also completed the 2-step Generic Design Assessment in the UK in 2025. The Polish Regulator has issued Decision in Principle for six sites to deploy BWRX-300s. Pre-application discussions are ongoing in several other countries.

11. Fuel Cycle Approach

The 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

The 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. The LWM is designed for zero liquid discharge.

13. Development Milestones

Year	Milestone	Status
2017	Initial Conceptual Design	**Completed**
2019	Pre-Application review of Licensing Topical Reports (U.S.A.)	**Completed**
2021	Conceptual Design Phase	**Completed**
2022	Preliminary Design Phase	**Completed**
2022	License to Construct application submitted (Canada)	**Completed**
2023	Vendor Design Review (VDR) completed (Canada)	**Completed**
2024	Generic Design Assessment (U.K.)	**Completed**
2025	License to Construct received (Canada)	**Completed**
2025	Construct first BWRX-300 unit begins (Canada)	**Completed**
2025	Construction Permit application submitted (U.S.A.)	**Completed**
2026	Construction Permit received (U.S.A.)	**Ongoing**
2030	Target commercial operation (Canada)	**Planned**

CNEA, Argentina
CAREM



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*CNEA, Argentina*
Reactor type	*Integral PWR*
Coolant/moderator	*Light Water / Light Water*
Design status	*Under construction (CAREM 25)*
Thermal/electrical capacity, MW(t)/MW(e)	*100/30 (CAREM 25)*
NSSS operating pressure (primary/secondary), MPa	*12.25/4.7 (CAREM 25)*
Core inlet/outlet coolant temperature (°C)	*284 / 326 (CAREM 25)*
Primary circulation	*Natural circulation*
Fuel type/assembly array	*UO_2 pellet/hexagonal*
Number of fuel assemblies in the core	*61 (CAREM 25)*
Fuel enrichment (%)	*3.1% (CAREM 25)*
Core discharge burnup (GWd/ton)	*24 (CAREM 25)*
Refuelling cycle (months)	*14 (CAREM 25)*
Fuel cycle requirements/approach	*390 full-power days and 50% of core replacement (CAREM25)*
Reactivity control mechanism	*Control rod driving mechanism (CRDM) only*
Approach to safety systems	*Passive*
Seismic design (SSE)	*0.25g (CAREM 25)*
Design life (years)	*40*
Plant footprint (m^2)	*36,000 (CAREM 25)*
RV height/diameter (m)	*11 / 3.2 (CAREM 25)*
RV weight (metric ton)	*267 (CAREM 25)*

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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☒ Desalination	☐ >550°C

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 UO2. 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 Gd2O3 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 Vessel (RV) 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.



Figure 1: CAREM reactor pressure vessel

(e) Reactor Coolant System and Steam Generator

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.

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.



Figure 2: CAREM steam generator

(f) Pressuriser

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.

(g) Primary pumps

N/A

(h) Distinguishing features

Core heat removal by natural circulation, pressure suppression containment

5. Safety Features

(i) 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.

(j) 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.

(k) 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.

(l) Safety Approach and Configuration to Manage DBC

N/A

(m) Safety Approach and Configuration to Manage DEC

N/A

(n) 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.

(o) 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.

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

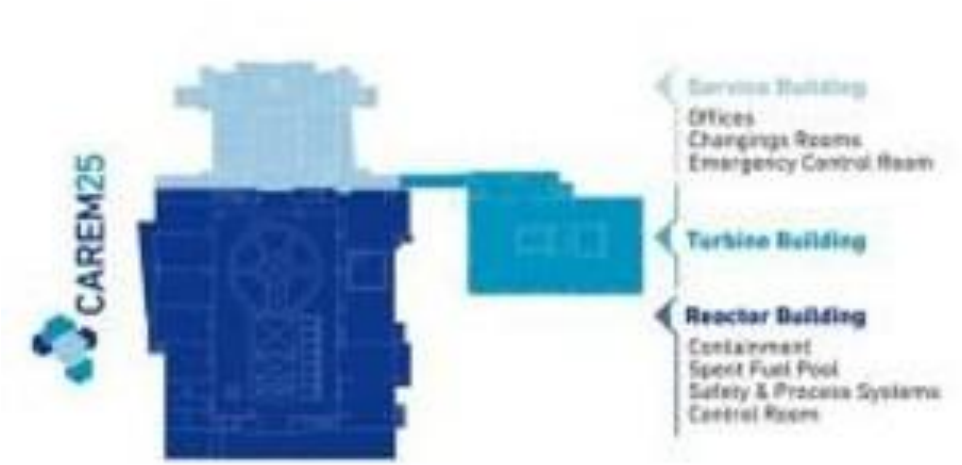


Figure 3: CAREM plant layout



Figure 3: CAREM25 construction site

The 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.

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.

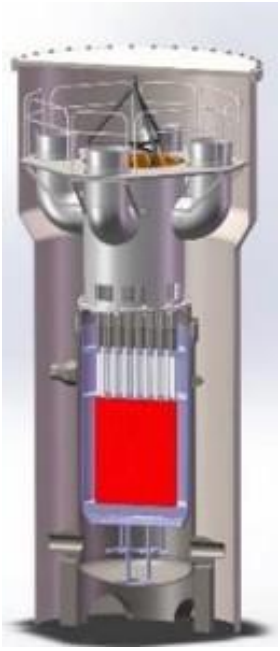
13. Development Milestones

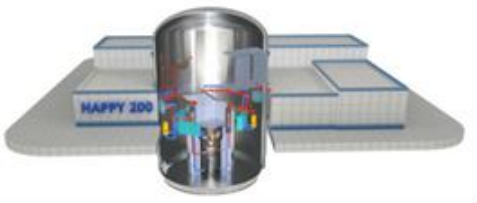
Year	Milestone	Status
1984	CAREM concept was presented in Lima, Peru, during the IAEA Conference on SMRs and was one of the first of the new generation reactor designs. CNEA officially launched the CAREM project	Completed
2006	Argentina Nuclear Reactivation Plan listed the CAREM25 project among priorities of national nuclear development	Completed
2009	CNEA submitted its preliminary safety analysis report (PSAR) for CAREM25 to the ARN	Completed
2011	Start-up of a high pressure and high temperature loop for testing the innovative hydraulic control rod drive mechanism	Completed
2011	Site excavation work beginning	Completed
2012	Civil design beginning	Completed
2013	Construction license	Completed
2014	Civil works started	Ongoing
2024	Electromechanical erection (start)	Ongoing
2028	First criticality	Planned

SPIC, China
HAPPY200



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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*
Design status	*Detailed Design*
Thermal/electrical capacity, MW(t)/MW(e)	*200 / 0 (thermal power only)*
NSSS operating pressure (primary/secondary), MPa	*0.6 / 0.8*
Core inlet/outlet coolant temperature (°C)	*80 / 120*
Primary circulation	*Forced (2 pumps)*
Fuel type/assembly array	*UO2 / 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*
Fuel cycle requirements/approach	*LEU / once through*
Reactivity control mechanism	*Rods*
Approach to safety systems	*Active / Passive*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*1150*
RV height/diameter (m)	*6.0 / 2.25*
RV weight (metric ton)	*26*

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

☐ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

The HAPPY200 is designed for heat generating power source, dedicated to providing 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) Nuclear Steam Supply System

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.

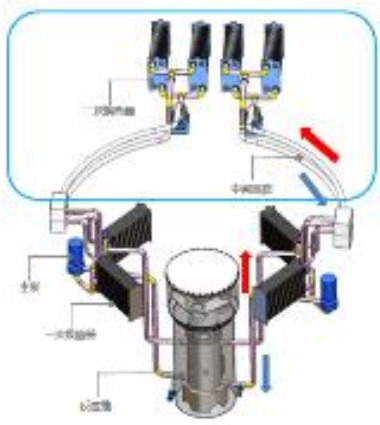


Figure 1

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.

(b) Reactor Core

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 UO2 with Gd as a burnable absorber. The

235U 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.

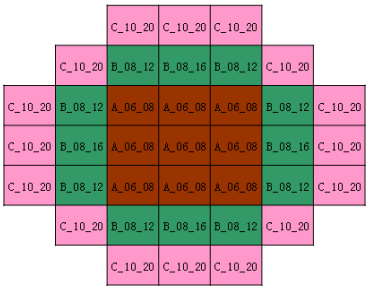


Figure 2

(c) 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.

(d) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

(f) Pressuriser

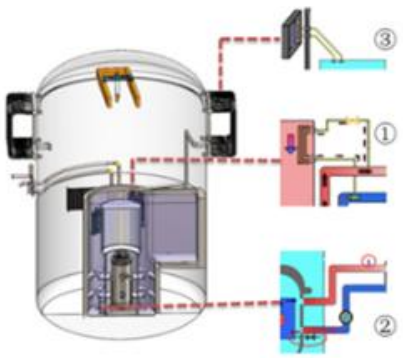
(g) Primary pumps

(h) Distinguishing features

5. Safety Features

(i) 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.



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

Figure 3

(j) 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.

(k) Emergency Core Cooling System

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

(l) Safety Approach and Configuration to Manage DBC

N/A

(m) Safety Approach and Configuration to Manage DEC

N/A

(n) 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.

(o) Spent Fuel Cooling Safety Approach / System

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



Figure 4: HAPPY200 plant layout

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.

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. \

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.

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

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

i-SMRDA, KHNP & KAERI, Republic of Korea i-SMR



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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*
Design status	*Detailed Design*
Thermal/electrical capacity, MW(t)/MW(e)	*520 / 170*
NSSS operating pressure (primary/secondary), MPa	*15.5 / 5.77(Feedwater Header)*
Core inlet/outlet coolant temperature (°C)	*286 / 321*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO2 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*
Fuel cycle requirements/approach	*Open fuel cycle same as operating PWRs*
Reactivity control mechanism	*Control rod, burnable absorber rods, moderator temperature coefficient*
Approach to safety systems	*Passive ≥72h*
Seismic design (SSE)	*0.5g*
Design life (years)	*80*
Plant footprint (m^2)	*undisclosed*
RV height/diameter (m)	*30.4 / 5.5*
RV weight (metric ton)	*650*

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

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☒ Load-following	☒PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

The i-SMR is a multi-purpose reactor designed to support both electricity generation and non-electric applications, including district heating and steam supply, hydrogen production, seawater desalination, and industrial process heat. 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 achieves design simplification by integrating the primary coolant system and eliminating LB-LOCA. The i-SMR adopts the boron-free operation for enhancing safety and economic performance. The fully-passive safety system of i-SMR achieves its safety goals without relying on any safety-grade electrical power or operator action.

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 (LB-LOCA) 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 refuelling 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 using control rods, burnable absorbers, and moderator temperature feedback. The i-SMR adopts boron-free operation, eliminating components and maintenance activities required for boron concentration control. The primary reactivity control system consists of the control rods and burnable absorbers. Burnable poison rods are used to flatten radial and axial power profiles, thereby increasing thermal margin of the core. The i-SMR adopts an In-Vessel Control Rod Drive Mechanism (IV-CRDM), which eliminates the possibility of a control rod ejection accident. A large number of control rods in the i-SMR core provides a relatively high control rod worth. With boron-free operation, the secondary reactivity control system utilizes the large negative MTC of the boron-free core for reactivity control.

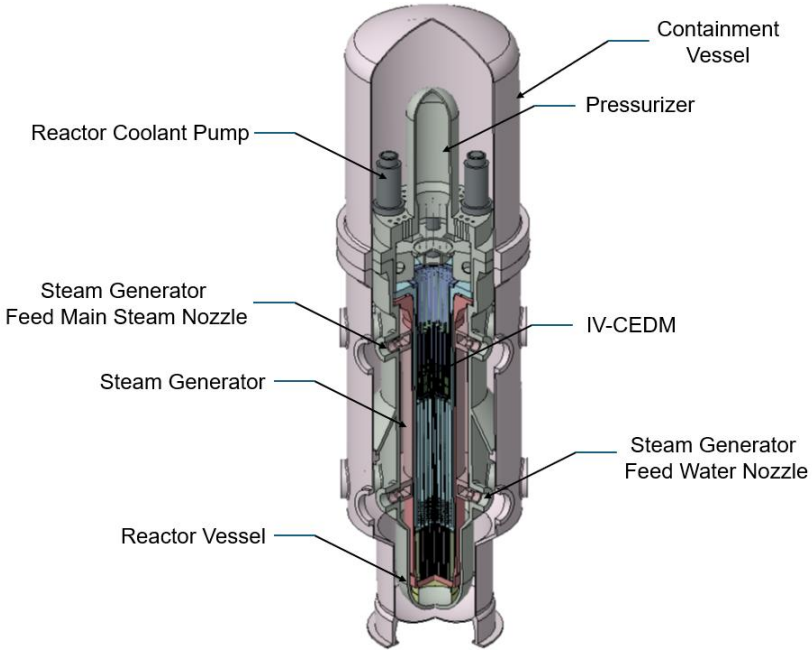
(d) Reactor Vessel (RV) and Internals

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 inner face 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 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.



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. During normal operation, the primary and secondary coolant flows inside the RV follow their respective single flow paths. Under accident conditions, a portion of helical tubes may participate in the heat transfer condition.

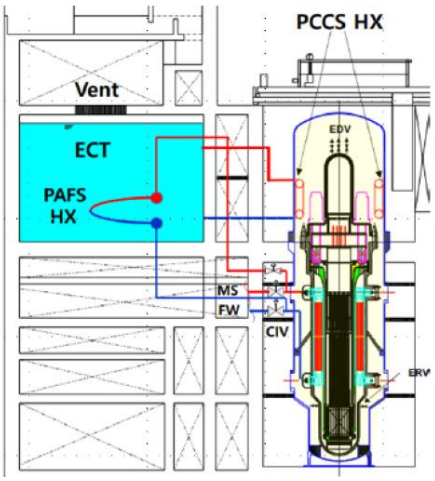
(f) Pressuriser

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 Pressurizer Safety Valves (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 Reactor Closure Head (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 (SB-LOCA) 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.



(h) Distinguishing features

The i-SMR incorporates several unique and innovative design features that significantly enhance inherent safety, operational flexibility, and economic competitiveness:

- Integral NSSS Architecture with Elimination of LB-LOCA: By housing the core, helical once-through steam generator, canned motor pumps, and pressurizer entirely inside a single reactor vessel, all large primary piping is eliminated, inherently preventing Large Break Loss of Coolant Accidents (LB-LOCA).
- Soluble Boron-Free Operation: The reactor operates without soluble boron for reactivity control during normal operation. This simplifies modular makeup and purification system (former CVCS), substantially reduces radioactive liquid waste processing, and ensures a strong negative Moderator Temperature Coefficient (MTC) across the entire fuel cycle.
- In-Vessel Control Rod Drive Mechanism (IV-CRDM): The control rod drive mechanisms are installed completely inside the reactor vessel head/upper internals, inherently eliminating the risk of Control Rod Ejection Accidents.
- Canned Motor Pumps (CMPs): Vertically installed canned motor pumps without mechanical shaft seals completely eliminate the possibility of pump seal LOCA.
- Fully Passive Safety & Modular Deployment: The design relies on fully passive safety systems (such as passive containment cooling and passive decay heat removal) requiring no safety-graded electrical power or operator action. The compact steel containment vessel and modularized nuclear island facilitate high-ratio factory fabrication, shortened construction schedules, and flexible multi-module deployment.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The Engineered Safety Features (ESFs) of i-SMR, which combine proven technologies with advanced design features, are designed to function automatically on demand and mitigate Design Basis Accidents (DBAs) and Design Extension Condition without significant fuel degradation (DEC-A). The 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;

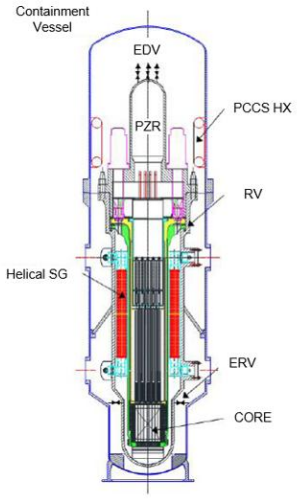


Figure 3

Passive Auxiliary Feedwater System (PAFS)

Passive Emergency Core Cooling System (PECCS)

Passive Containment Cooling System (PCCS)

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, Emergency Depressurization Valves (EDVs), Emergency Recirculation Valves (ERVs). Two (2) ERVs and Three (3) 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 build s 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 the RV and CV. Once the pressures of the RV and CV become sufficiently equalized, the condensed water in the CV annulus is recirculated back to the RV via the ERVs. As such, continuous natural circulation is maintained and thereby ensuring continuous core cooling.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) 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.

(m) Safety Approach and Configuration to Manage DEC

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

For phenomena that could lead to early containment failure, including those occurring within 24 hours after an accident, mitigation systems shall be provided, or the design shall address the phenomena, even if their probability of occurrence is low. For phenomena that could potentially lead to late containment failure if not properly mitigated, the mitigation systems or design measures should be considered in conjunction with the probabilistic safety goals and cost for incorporating such features.

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.

(n) 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.

(o) 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 are 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 plant availability of at least 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, defence-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.

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. The current stage of design development for the i-SMR is the basic design phase. Regarding licensing progress, the i-SMR is actively undergoing regulatory review for Standard Design Approval (SDA) by the Korean regulatory authority. The design is being thoroughly developed and validated in conformity with Korean law, codes and standards for nuclear power plants, as well as global 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

Year	Milestone	Status
2019-2020	Conceptual Design Phase	**Completed**
2021-2023	Basic Design Phase	**Completed**
2024-2025	Standard Design Phase (and preparation for pre-licensing)	**Completed**

Year	Milestone	Status
2026-2028	Standard Design Approval(SDA) Review & Licensing	**Ongoing**
2026-	FOAK Deployment (Site Finalized, Construction Permit Application Preparation)	**Ongoing**

NuScale Power LLC, United States of America NuScale Power Module™



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*NuScale Power LLC., USA*
Reactor type	*Integral PWR*
Coolant/moderator	*Light Water / Light Water*
Design status	*Equipment Manufacturing in Progress*
Thermal/electrical capacity, MW(t)/MW(e)	*250 / 77*
NSSS operating pressure (primary/secondary), MPa	*13.8 / 4.3*
Core inlet/outlet coolant temperature (°C)	*249 / 316*
Primary circulation	*Natural circulation*
Fuel type/assembly array	*UO2 pellet / 17 X 17 square*
Number of fuel assemblies in the core	*37*
Fuel enrichment (%)	*≤ 4.95*
Core discharge burnup (GWd/ton)	*50 – 55 with cycle optimization 60 (max allowable locally)*
Refuelling cycle (months)	*18*
Fuel cycle requirements/approach	*Nominal Three stage in-out refuelling scheme*
Reactivity control mechanism	*Control rod drive, soluble boron*
Approach to safety systems	*Passive unlimited time*
Seismic design (SSE)	*0.5g*
Design life (years)	*60*
Plant footprint (m^2)	*Dependent on plant design options*
RV height/diameter (m)	*17.7 / 2.7*
RV weight (metric ton)	*tbc*

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 ModulesTM (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

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☒HTSE - electrolysis	☒ <200°C
☒ Load-following	☒PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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 UO2 with Gd2O3 as a burnable absorber homogeneously mixed within the fuel for select rod locations. The 235U 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 B4C and their length is 2 m.

(d) Reactor Vessel (RV) 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) Pressuriser

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).

(g) Primary pumps

There are no primary pumps, it relies in natural circulation.

(h) Distinguishing 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. The NPM includes an integral primary system configuration, a containment vessel, passive heat removal systems, and severe accident mitigation features. This design is fully passively safe as embodied by the NuScale Triple Crown for Nuclear Plant Safety™, which ensures that reactors will safely shut down and self-cool, indefinitely, without the need for operator or computer action, AC or DC power, or the addition of water—a first for light water reactor technology. The NuScale design is approved by NRC to operate off-grid, in island mode and it is black start capable.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

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

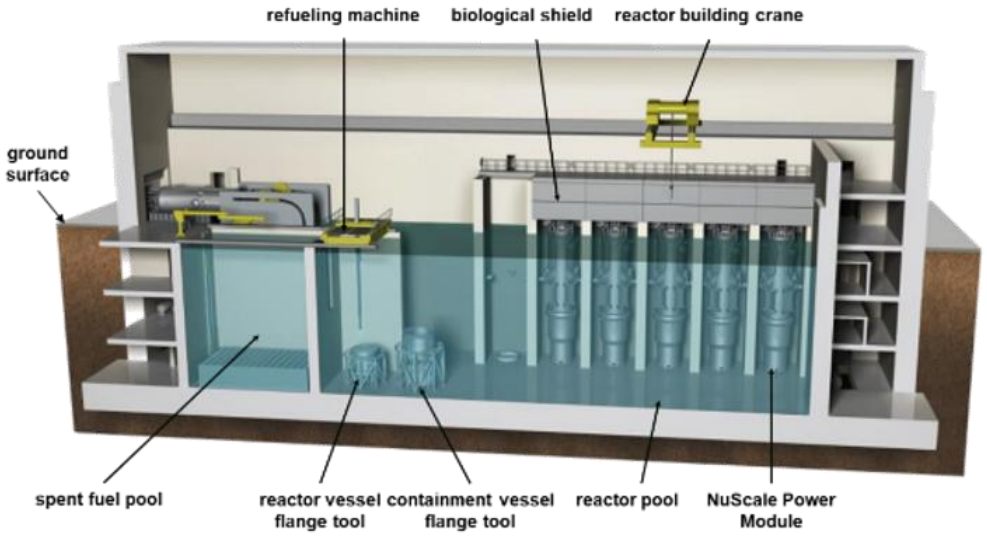


Figure 1: Cut-away view of SMR power plant with NuScale NPMs

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

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.

(m) 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.

(n) 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.

(o) Spent Fuel Cooling Safety Approach / System

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 Systems

The NuScale design includes a fully digital control system based on the use of field programmable gate array (FPGA) technology. The highly integrated protection system (HIPS) platform, approved by the 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

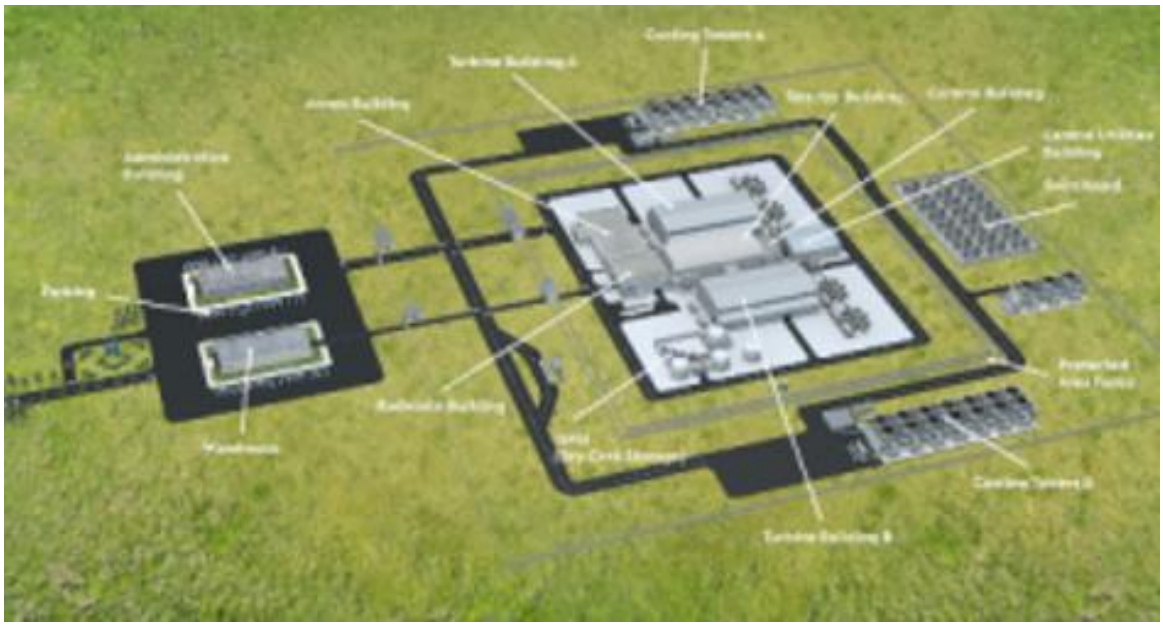


Figure 2: An example of a Site Layout of an SMR plant utilizing 12 NuScale Power Modules

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.

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.

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. The SDA application and topical reports were received by the NRC in January 2023 and approval granted ahead of schedule in May 2025.

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. Development Milestones

Year	Milestone	Status
2003	Initial concept developed and integral test facility operational	**Completed**
2007	NuScale Power, LLC was formed	**Completed**
2008	Initiation of NRC Pre-Application	**Completed**
2011	NuScale Power is acquired by Fluor	**Completed**
2012	Twelve-reactor simulated control room was commissioned	**Completed**
2013	U.S. DOE SMR Cooperative Agreement signed	**Completed**
2016	Design certification application submitted to the U.S. NRC for 160 MWt design	**Completed**
2020	Standard Design Approval (SDA) received from the US NRC in September 2020	**Completed**
2022	NPM production began with the production of forging dies for the Upper Reactor Pressure Vessel	**Completed**
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.	**Completed**
2022	Uprated Standard Design Approval submitted to the NRC to raise nominal power to 250 MWt	**Completed**
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.	**Completed**

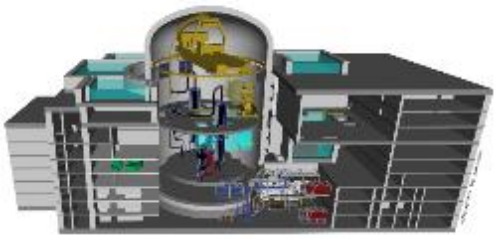
Year	Milestone	Status
2023	Fabrication of 6 Reactor Pressure Vessels and Steam Generator Tube Bundles begins	**Completed**
2023	Uprated Standard Design Approval docketed by US NRC for uprated NuScale Power Module with 250 MWt (77 MWe); US NRC provided a 24-month review schedule for approval.	**Completed**
2025	Design Approval from the NRC received for 250 MWt (77 MWe) NPM design, ahead of schedule.	**Completed**

EDF Group, France
NUWARD



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*NUWARD (EDF subsidiary) in partnership with EDF, Framatome, Arabelle Solutions, Tractebel, Ansaldo Nucleare*
Reactor type	*PWR, Generation III+*
Coolant/moderator	*Light Water / Light Water*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*1150 / 400*
NSSS operating pressure (primary/secondary), MPa	*15.5 / 7.2*
Core inlet/outlet coolant temperature (°C)	*296 / 324°C*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO2 / standard 17x17 fuel assemblies* *The design also allows for use of reprocessed uranium and up to 30% MOX*
Number of fuel assemblies in the core	*97 fuel assemblies*
Fuel enrichment (%)	*<5*
Core discharge burnup (GWd/ton)	*<75 max (UOX)* *<65 max (MOX)* *Actual burnup values expected to be lower*
Refuelling cycle (months)	*18 to 24 months (40/97 reload for 24M)*
Fuel cycle requirements/approach	*Flexible fuel strategy: standard and reprocess uranium and MOX fuel* *Possible adaptation to multiple fuel supplier*
Reactivity control mechanism	*Control rods, boric acid, solid burnable poisons*
Approach to safety systems	*Active and passive systems*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*Between 120 000 and 150 000 for industrial part (including nuclear island, conventional island, balance of plant)*
RV height/diameter (m)	*9.85 / 4,25*
RV weight (metric ton)	*RPV + RPVH (Reactor Pressure Vessel Head): 260t*

1. Introduction

NUWARD SMR is a generation III+ Pressurised Water Reactor (PWR). It is developed by NUWARD, the EDF Group subsidiary. NUWARD SMR relies on the EDF Group's technical and industrial expertise and Framatome's and Arabelle Solutions' industrial infrastructure and integrates partners to enhance its construction and modularity skills, such as Tractebel and Ansaldo Nucleare.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☒HTSE - electrolysis	☒ <200°C
☒ Load-following	☒PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

NUWARD's mission is to empower industries, utilities and datacenters to deliver comprehensive low-carbon energy, efficiently meeting both electricity and heat demands for end clients. 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's value proposition is to leverage proven technology, enhanced through simplification, modularisation, and prefabrication, to deliver reliable, fast, and predictable projects for our customers.NUWARD SMR is a pressurised water SMR fully relying on proven technologies, an established industrial sector, and a sustainable fuel cycle. From the early design phases onwards, key elements such as safety, security and safeguards, international deployment, cost effectiveness and competitiveness, modularity, manufacturability, constructability, operability and maintainability are integrated in the design.

4. Main Design Features

(a) Nuclear Steam Supply System

The reactor coolant system adopts a two-loop configuration. Each loop, connected to the reactor pressure vessel, consists of a steam generator, reactor coolant pump and piping. A pressuriser is connected to the reactor coolant system to maintain system pressure within operational limits.



Figure 1: Reactor Coolant System

(b) Reactor Core

The reference core is based on proven 17x17 (97) fuel assemblies used in the operating PWR fleet with a shortened height and UO2 rods (enrichment < 5wt% 235U). Even though reference fuel is with U235, the design allows for the use of reprocessed uranium and up to 30% MOX fuel. The design supports fuel cycle lengths of 18 to 24 months, with a half-core reload strategy for the 24-months cycle.

(c) Reactivity Control

The reactor core is controlled by Rod Cluster Control Assemblies inserted in guide tubes. Overall, the core reactivity is controlled by three means: (i) control rods, (ii) boric acid in the coolant, and (iii) gadolinium as neutron absorber integrated in fuel pellets.

(d) Reactor Vessel (RV) and Internals

The RPV features a cylindrical body with a hemispherical bottom head and a removable, flanged hemispherical upper head fitted with a gasketed closure. The vessel provides the required internal volume to accommodate the reactor core, control rod assemblies, heavy reflector, and the associated structural and flow-directing internals. The vessel is fabricated from low-alloy steel, with the entire internal surface clad in stainless steel to ensure corrosion resistance. The RPV has two inlet nozzles, and two outlet nozzles located in a horizontal plane just below the reactor vessel flange and above the top of the core. The vessel design has been optimised to minimise the number of forgings and welds, thereby reducing procurement and fabrication time. The RPV closure head is provided with penetrations for adapters for the Control Rod Drive Mechanism. Additional penetrations are foreseen for instrumentation adapters, with all instrumentation through the closure head and none through the bottom of the RPV vessel.

(e) Reactor Coolant System and Steam Generator

The design of the RCS components is grounded in proven technologies with extensive positive operational feedback. It follows a design-to-manufacturing approach, integrating manufacturing constraints from the earliest design stages. This methodology ensures the technical feasibility, quality, cost efficiency, and schedule reliability of component production. The steam generators are vertical, U-tube, natural circulation heat exchangers. They are completely manufactured in-shop, transported to site and installed in the reactor building as one piece.

(f) Pressuriser

The pressuriser consists in a vertical cylindrical shell, closed at both ends by hemispherical heads. It is made of low-alloy steel, with austenitic stainless-steel cladding on all internal surfaces in contact with the reactor coolant. The pressuriser is fitted with electric heater rods, mounted vertically within the lower head, to maintain and control system pressure under steady-state and transient conditions. A spray system is installed near the top of the upper cylindrical shell to provide fine control of pressure through steam condensation. Safety valves, located on the upper head, ensure overpressure protection of the RCS.

(g) Primary pumps

The primary pumps are vertical, single stage, mixed-flow type. The flow enters the pump vertically from below. The flow leaves the pump casing via the horizontal discharge connection branch. Water suction is axial, and discharge is radial. The primary pump design is based on French fleet design with mechanical sealing.

(h) Distinguishing features

Based on proven technology, simplification, modularisation and prefabrication. Delivering both electricity and heat with high degree of delivery flexibility.

The design is able to burn MOX.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

NUWARD SMR is designed to be compliant with IAEA and WENRA standards. The principle of defense-in-depth as defined by IAEA and WENRA is applied. Regarding radiological risks, the general design nuclear safety approach for NUWARD SMR is based on a conservative deterministic approach taking into account the principle of defense-in-depth, supplemented by probabilistic verification. The deterministic design approach is structured around two distinct fields: Design Basis Conditions (DBC) and Design Extension Conditions (DEC). The design in these two fields is completed by taking into account, at the design stage, internal and external hazards and credible combination of events. In addition, design provisions are taken to demonstrate the practical elimination of certain situations. The NUWARD SMR safety approach integrates an optimised combination of active and passive systems.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

The management of DBC is mainly based on active systems to ensure the three safety functions:

- Reactivity control is performed either using control rods insertion, or boron injection through the Safety Injection System (SIS), or the high-pressure boron injection system.
- Removal of thermal power from radioactive substances and nuclear reactions is performed through different systems: Injection of water into the RCS is performed by the active SIS. Power extraction from the RCS is performed when steam generators are available using the active Emergency Feed Water System (EFWS), and steam is discharged to the atmosphere. In shutdown states, the active Reactor Heat Removal System (RHRS) is used to transfer the residual heat to the heat sink.
- Confinement of radioactive substances is performed ultimately by the containment building.

All systems used to mitigate a DBC event are designed considering the single failure criterion and preventive maintenance. Active systems are electrically backed up by on-site emergency diesel generators in case of a Loss of Off-site Power.

(m) Safety Approach and Configuration to Manage DEC

The management of DEC without core melt (DEC-A) relies upon active and passive SSCs, diversified where appropriate, depending on the considered conditions and the SSC role in DBC conditions.

An additional level of plant robustness is foreseen, by prevention of core-melt when active systems are unavailable due to the absence of electric power. A passive safety condenser system allows for passive decay heat removal from the core and prevents core melt in case of the extended total loss of AC power during power operation.

The management of severe accident scenarios (DEC-B) is based on a passive approach. The Passive Containment Cooling System evacuates the heat from the reactor building and controls the temperature and pressure inside the containment building passively through containment wall condensers. Corium stabilisation and cooling are achieved using an ex vessel retention strategy, in which the molten core is spread over a large surface beneath the reactor pressure vessel and cooled by overflow water from the In-Containment Refueling Water Storage Tank (IRWST). This passive severe accident mitigation strategy—providing both corium cooling and containment heat removal—eliminates the need to recirculate active, contaminated water to areas outside the containment during a severe accident.

(n) Containment System

The confinement of radioactive materials is based on the proven concept of 3 successive barriers: Fuel rod cladding, main primary system and reactor building containment. The reactor building containment is a single structure made of prestressed reinforced concrete with a metal coating (liner) on the internal walls to enhance its leak tightness.

(o) Spent Fuel Cooling Safety Approach / System

The interim storage spent-fuel pool is located in the fuel building, close to the reactor building. The pool is borated to provide reactivity control. The heat removal is ensured by immersed heat exchanges connected to the Component Cooling Water System, removing the need to circulate fuel pool water outside of the pool boundary. Fuel handling and storage follows the same proven principles as the other French reactor series with qualified and proven technological components. The fuel pool sizing ensures sufficient capacity to store spent fuel including 30% MOX up to at least 10 years of operation.

6. Plant Safety and Operational Performances

The alternation of various types of outages (refuelling-only, ten yearly, partial maintenance outage) allows for periodic and systematic maintenance/testing/inspection/surveillance activities required by regulations, while minimising maintenance cost and maximising the capacity factor of the plant over its lifetime, targeting a value of at least 92%. As well as maintenance, plant operation is optimised by design, including consideration of human factors engineering not only in the design of the control room and the Human Machine System but also in the overall design of the plant and its Concept of Operations while taking into account extensive operation experience feedback.

The grid requirements considered are compliant with ENTSO-E network codes (European Network of Transmission System Operators for Electricity). The NUWARD SMR is designed to modulate its power output from 100% down to 30%, twice per day, in under 30 minutes, to follow grid or customer demand. The NUWARD SMR can supply both electricity and heat.

7. Instrumentation and Control Systems

The Instrumentation and Control (I&C) system is designed in compliance with international standards (IAEA, WENRA, IEC). The overall I&C architecture is based on the evaluation of actions to be performed and on the classification criteria for I&C functions important to safety. It complies with I&C safety and cybersecurity requirements, human factor engineering requirements and constraints, general operating rules, and extensive construction/operation/maintenance experience. To minimise as far as reasonably possible the potential for common-cause failure, prevention measures are taken, including physical and electrical separation, as well as technological and functional diversity.

8. Plant Layout Arrangement

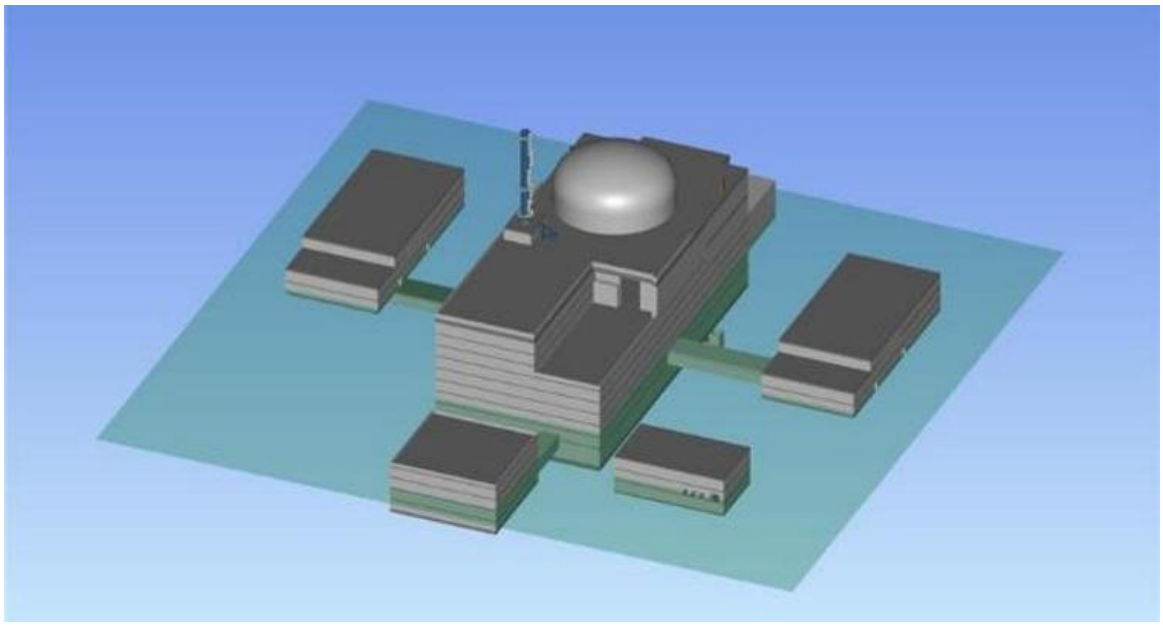


Figure 2:NUWARD simplified 3D plant layout

The NUWARD SMR industrial area fits within any surface area from 12 to 15 ha.The industrial area includes the Nuclear Island, Conventional Island and Balance of Plant. The NUWARD SMR design is adapted to both open and closed heat sink – Standard plant design includes closed heat sink. The plant layout incorporates elements to optimise the construction schedule, such as sufficient building separation to enable parallel construction activities and reduced width of the central island to allow gantry circulation and facilitate open-top construction and modular installation.

9. Testing Conducted for Design Verification and Validation

Most of the NUWARD SMR design relies on proven components already implemented in existing medium and high power PWRs, so the need for testing is inherently limited. The NUWARD SMR development process includes activities to ensure the scientific calculation tools are representative and validated before performing accident analyses for the safety demonstration.

10. Design and Licensing Status

The conceptual design phase for the NUWARD SMR is finalised mid-2026, after which the basic design phase is entered including preparation of the PSAR. Regarding licensing preparation, technical exchanges are in progress on key safety topics with the French nuclear safety authority (ASNR). In order to prepare for a wider international deployment of the NUWARD SMR, a Joint Early Review (JER) initiative of the NUWARD SMR design is ongoing since 2022, gathering the regulators ASNR (France), SUJB (the Czech Republic, with technical support from SURO), PAA (Poland), ANVS (the Netherlands), SSM (Sweden), STUK (Finland) and FANC (Belgium, with technical support from Bel-V). ISIN – the nuclear regulator for Italy – participates in an observer role. These exchanges allow NUWARD to benefit from early feedback from the regulators' perspective, and to anticipate foreign regulators' expectations, paving the way for a successful future deployment with a standard design adapted to international needs.

11. Fuel Cycle Approach

NUWARD SMR uses proven PWR-type fuel assemblies based on EDF's extensive operational experience. The fuel cycle supports 18- to 24-month cycles, with 40 assemblies (out of 97) reload strategy for 24-month cycles and flexibility for stretch-out capabilities. NUWARD SMR is capable of using up to 30% MOX fuel (mixed oxide of recycled plutonium and depleted uranium). This is made possible by a proven and sustainable optimised European uranium cycle, allowing for the recycling of over 96% of fissile material and a significant reduction in nuclear waste.

12. Waste Management and Disposal Plan

The NUWARD SMR is designed to achieve a reduction in waste production compared with current PWRs. The plan for waste management and disposal takes into account industry best practice. The operation of the plant will result in the generation of the same type of waste as for existing PWRs. The disposal plan for this waste will be aligned with the owner's strategy.

13. Development Milestones

Year	Milestone	Status
mid 2024 – mid 2026	Technology Conceptual Design	**Completed**
mid 2026 – 2028	Basic Design and PSAR	**Ongoing**
2029 – 2030	Detailed Design	**Planned**

Last Energy Inc., United States of America

PWR-20



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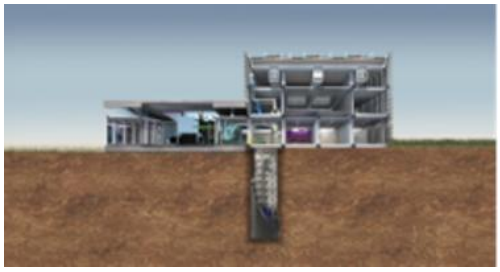


Figure 1: Isolated PWR-20 Nuclear Steam Supply System



Figure 2: 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*
Design status	*Detailed Design*
Thermal/electrical capacity, MW(t)/MW(e)	*80 MWth / 20 MWe (nominal)*
NSSS operating pressure (primary/secondary), MPa	*15.5 / 2.8*
Core inlet/outlet coolant temperature (°C)	*270 / 331*
Primary circulation	*Forced (4 pumps)*
Fuel type/assembly array	*UO2 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*
Fuel cycle requirements/approach	*Once-through*
Reactivity control mechanism	*Control rods and negative moderator coefficient*
Approach to safety systems	*Defense-in-depth, Redundancy, Diversity, and Passive Safety*
Seismic design (SSE)	*0.5g horizontal and 0.4g vertical peak ground accelerations*
Design life (years)	*42*
Plant footprint (m^2)	*968*
RV height/diameter (m)	*8.9/1.8*
RV weight (metric ton)	*39*

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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 Vessel (RV) 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.

(h) Distinguishing features

Modular Construction, Factory-Fabricated, Air-Cooled, Zero Water Consumption.

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

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.

(m) Safety Approach and Configuration to Manage DEC

See (l).

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.

(n) Containment System

The PWR-20's containment system (Figure 3) 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.



Figure 3: PWR-20 containment system

(o) 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

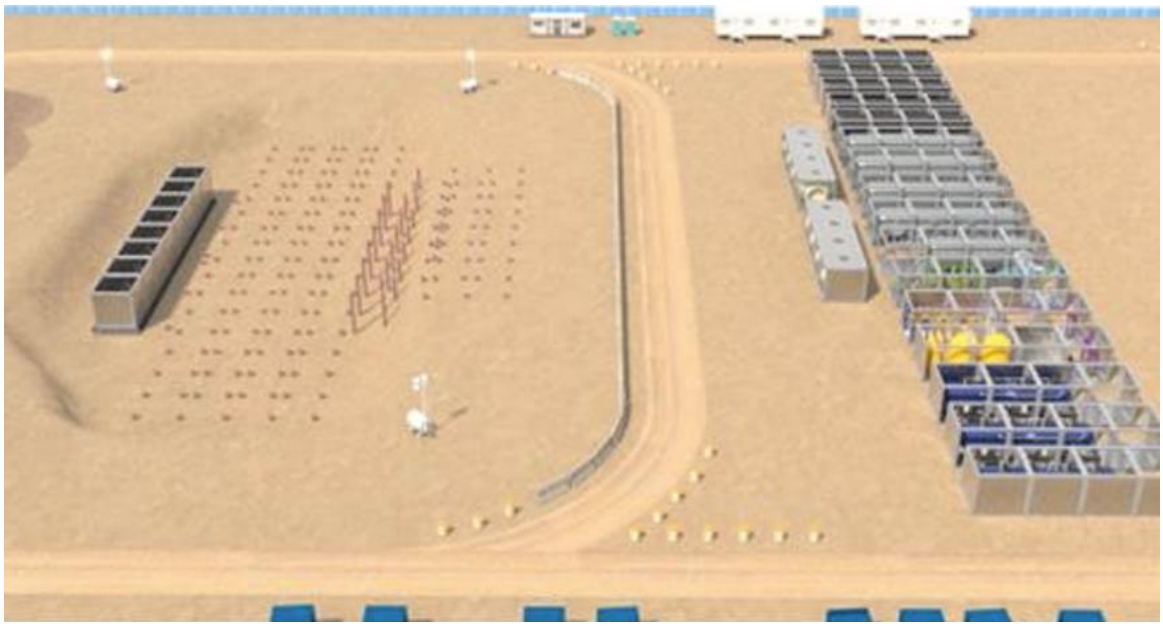


Figure 4: PWR-20 On Site Assembly Layout

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.

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

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

JSC “Afrikantov OKBM”, Russian Federation

RITM-200N

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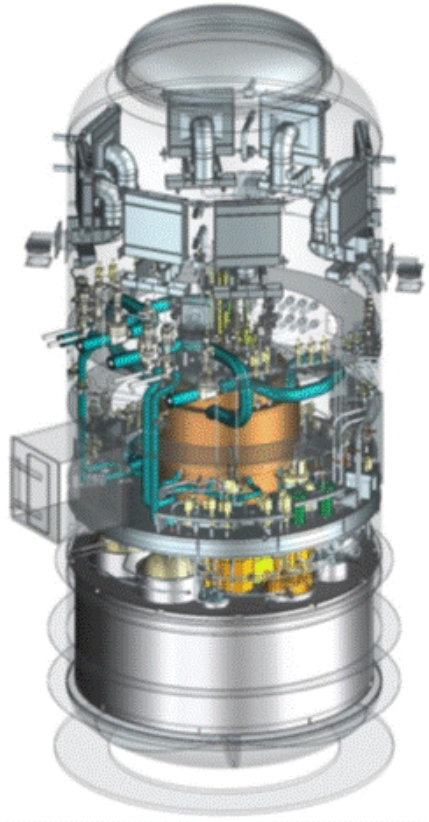


Figure 1: RITM-200N reactor plant

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Afrikantov OKBM, Rosatom, Russian Federation*
Reactor type	*Integral PWR*
Coolant/moderator	*Water / Water*
Design status	*Under construction*
Thermal/electrical capacity, MW(t)/MW(e)	*190 / 55*
NSSS operating pressure (primary/secondary), MPa	*15.7 / 3.83*
Core inlet/outlet coolant temperature (°C)	*283 / 321*
Primary circulation	*Forced (4 pumps)*
Fuel type/assembly array	*UO2 pellet/hexagonal assembly*
Number of fuel assemblies in the core	*199*
Fuel enrichment (%)	*< 20*
Core discharge burnup (GWd/ton)	*N/A*
Refuelling cycle (months)	*60 – 72*
Fuel cycle requirements/approach	*Refueling every 5 – 6 years*
Reactivity control mechanism	*CRD mechanisms of CPS*
Approach to safety systems	*Combined active and passive safety systems*
Seismic design (SSE)	*0.3 g*
Design life (years)	*60*
Plant footprint (m^2)	*120 000*
RV height/diameter (m)	*7.5 / 3.4*
RV weight (metric ton)	*146*

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.

2. Target Application

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

3. Design Philosophy

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 MWe (1 reactor plant + 1 STP in the main building); 110 MWe (2 reactor plants + 2 STPs in the main building); 220 MWe, 330 MWe by a modular principle through constructing additional units with their own main buildings and cooling towers and with common plant-shared systems.

4. Main Design Features

(a) Nuclear Steam Supply System

The integral design of the RITM-200N reactor plant provides that the core and steam generators (SG) be within the reactor vessel. The hydraulic chambers of the reactor coolant pumps (RCP) are welded directly to the reactor pressure vessel (RPV), which eliminates any and all pipelines in the reactor coolant system (RCS). This approach significantly enhances safety through preventing loss-of-coolant accident (LOCA) scenarios. The primary circuit includes an integral reactor, a pressurizer system and a purification and cooling system.

The heat removal from the RITM-200N core is based on forced circulation in the primary circuit divided into four loops. The RCPs are in the cold leg of each loop. Each loop of the circuit consists of three steam generator cassettes (twelve in total) with separate feedwater headers and a common steam header. The configuration of the cassettes makes it possible to compactly place them in the reactor pressure vessel. To monitor the reactor coolant temperature, sensors are provided at the core inlet, at the core outlet and at fuel assembly outlets.

(b) Reactor Core

The RITM-200N reactor core is comprised of 199 fuel assemblies with low-enriched fuel (below 20%). The fuel assemblies are similar to those used in the KLT-40S and meet the international non-proliferation requirements. The cermet fuel consists of uranium dioxide (UO_2) fuel particles embedded in silicon-aluminum alloy (silumin). Compared to KLT-40S, the RITM-200N core has a longer refueling cycle.

The main design features that increase fuel reliability are the following:

- High thermal conductivity reduces the temperature inside the fuel column and lowers the damage probability in accidents. The cermet fuel is also more resistant to swelling throughout its long in-core residence period.
- The fuel cladding is made from a corrosion-resistant alloy. In a long fuel cycle, the alloy is more resistant to corrosion than zirconium. The same cladding alloy is used in the reactor cores on new nuclear-powered icebreakers operated in Russia.

(c) Reactivity Control

Control rods of the electromechanical reactivity control system and heterogeneous burnable absorber are used to compensate for the core reactivity. No soluble absorber is used for core reactivity control and compensation during load-follow operations; as a result, the formation of liquid radioactive waste is limited.

(d) Reactor Vessel (RV) and Internals

See (a).

(e) Reactor Coolant System and Steam Generator

The Steam Generator is vertical, once-through and located within the RPV.

(f) Pressuriser

N/A

(g) Primary pumps

See (a).

(h) Distinguishing features

Modular design; integral reactor; in-vessel corium retention in severe accidents.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The RITM-200M reactor plant safety concept is based upon the defense-in-depth principle combined with plant's inherent safety features and with using passive systems. The inherent safety features make it possible to automatically control energy flux density and an automatic reactor shutdown, primary coolant pressure and temperature limitation, a heatup rate, a primary coolant system leak size and outflow rate, a failed fuel fraction, reactor vessel integrity maintained in severe accidents and a "passive reactor" concept established so that the reactor be resistant to possible malfunctions. The RITM-200N reactor plant optimally combines passive and active safety systems to ensure normal operation and resistance to design-basis accidents.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

In terms of safety, the RITM-200N reactor design is based on achieving higher reliability through the following principles:

- Defense-in-depth:
- Inherent safety
- Physical redundancy and diversity
- Physical separation and independence
- Protection against external events
- Prevention of common cause failures
- Use of proven technologies

The "defense-in-depth" principle is the cornerstone of the safety principle adopted in RITM-200N, with the implemented several protection levels, including sequential barriers preventing the release of radioactive substances into the environment. This principle is applied to three basic safety functions—reactivity control, fuel cooling and radioactive substance containment—in order to protect the population, personnel and the environment. This includes accident prevention and mitigating accident consequences.

The defense-in-depth principle requires that possibilities of multiple failures be more carefully considered and that diverse equipment be used to perform the three basic safety functions mentioned above.

The aim of the reactor inherent safety features is to limit the energy release, limit the pressure and temperature growth, limit the heatup rate and the fuel failure rate. They also prevent initiating events and emergency states, limit their consequences without any operator's actions for a long time.

The internal safety features by design are as follows:

- A negative fuel and coolant temperature coefficient of reactivity that limits a power growth;
- A high thermal conductivity of the fuel composition eliminates hot spots in the core;
- A high heat capacity of the primary circuit, which increases the response time of the safety system;
- The integral reactor configuration practically eliminates large-break loss-of-coolant accidents.

A reasonable combination of redundancy and diversity was used in the design. Physical and spatial separation was applied to safety systems. The auxiliary functions such as power supply, control, cooling are as independent as possible. The special focus was on redundancy and diversity of power supply sources. For developing the protective

structures, external events were taken into account, such as an aircraft crash, seismic events, flooding, tornado and fire.

The combination of passive and active safety systems ensures a higher safety level in a wide spectrum of postulated accident scenarios. Active safety systems are supplemented by passive systems. The combination of passive and active safety systems ensures a higher safety level in a wide spectrum of postulated accident scenarios.

The control system automatically actuates the safety systems when the monitored parameters reach the corresponding values. In case the automated systems fail, the self-actuated devices will be brought into operation directly by the primary coolant system pressure to ensure that the reactor shuts down and the safety systems get actuated. The control rods of the CPS fall into the core by gravity assisted by the spring action when the power supply is disconnected from the holding electromagnets. This shuts down the reactor even in the case of a complete blackout.

Hydrogen safety is ensured by passive hydrogen recombiners. The analysis of the impact that the SNPP has on the radiation situation has shown that the population exposure dose outside the plant is below the levels requiring protective measures.

(m) Safety Approach and Configuration to Manage DEC

The RITM-200N design provides for a protection in a postulated core melt accident. The passive heat removal from the core guarantees that:

- The containment internal pressure do not exceed the design pressure,
- The containment strength be preserved,
- The corium be confined on the bottom of the integral reactor pressure vessel,
- The corium subcriticality be ensured,

A caisson water filling system, specially provided in the integral reactor, guarantee the RPV strength.

In RITM-200N reactor plant, the reactor control is ensured in all modes due to negative reactivity effects (the growth of the nuclear reaction power can never cause its further growth). The instrumentation and control systems have multiple redundancy, a wide range of measuring instruments and have the ability to take all necessary measures without any direct operator's actions, long before approaching the permissible parameter range limits.

(n) Containment System

The RITM-200N integral reactor is placed in a sealed steel containment to confine possible radioactive releases. The RITM-200N steel containment dimensions are dia. 8.7 m × 21 m.

The steel containment is designed to withstand an internal overpressure of 0.9 MPa in severe accidents. The reactor plant containment is a cylindrical protective enclosure with an elliptical bottom and a mouth in the upper part with a removable connection with an elliptical head.

Biological shielding blocks divide the internal volume of the steel containment into two rooms—an equipment space and a reactor space. The equipment space takes the major volume, and the reactor space is right beneath it. The division into two separate sealed rooms enables equipment repairing or replacing.

The metal-water shield tank is a load-bearing structure for the reactor vessel base, the equipment of the primary circuit systems and the dry biological shielding blocks. The metal-water shield tank has caissons for the integral reactor, the pressurizer system equipment and the purification and cooling system equipment.

In the upper part of the equipment space, there is containment heat removal equipment—heat exchangers, fans and air ducts. Access to the equipment space is via an airlock. The equipment space has manholes for visual inspections of the reactor compartment using periscopes and manholes for the personnel.

Outside the reactor plant steel containment, the reinforced concrete sealed enclosure of the reactor unit and the building structures are arranged to:

- Prevent or limit the propagation of released radioactive substances beyond the containment boundaries;
- Protect the personnel and population from ionizing radiation;
- In the cases stipulated by the nuclear plant design, protect—against external natural and human-made impacts—the systems and elements that are inside the containment and that, if failed, may cause a radioactive substance release in quantities exceeding safe operation limits, beyond the boundaries established by the nuclear plant design.

The sealing steel lining of reinforced concrete sealed enclosure is used as formwork.

(o) Spent Fuel Cooling Safety Approach / System

The reactor core refueling is carried out once every 6 years. The refueling equipment is used for loading the core into the reactor, unloading it into the spent fuel pools, and loading spent fuel assemblies into dedicated casks for storage and transport of spent nuclear fuel for reprocessing.

6. Plant Safety and Operational Performances

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 Systems

The reactor plant unit instrumentation and control (I&C) system:

- Controls and monitors all important process parameters and reactor plant equipment;
- Protects and automatically adjusts plant parameters;
- Diagnoses processes and equipment;
- Monitors and displays plant parameters in all operating modes.

The unit instrumentation and control system is implemented as a multilevel, distributing and integrating system that provides structural, functional and interface independence of the unit subsystems, which results in that functions of individual subsystems and their elements be preserved in case the automated control system fails, including at individual levels of the system hierarchy.

In terms of safety in case of failures, the control system architecture is based on the defense-in-depth principle. The multilevel protection using independent automated control subsystems ensures that each basic function be performed during the entire period of various operating modes with account of common cause failures.

The automated I&C system is a single system that includes technical, mathematical, metrological, organizational and information support, operational and design documentation. The system includes engineered features that perform the entire range of monitoring and control operations, in particular, collect, process, transmit and disseminate information, generate and take control actions.

The automated I&C system provides for the maximum use of unified hardware and software based on failsafe schematics and contains built-in monitoring and testing tools. The automated I&C system facilitates further modernization and expansion at all levels of the system hierarchy, since it complies with the protocols of international standards and the modular structure of lower level software and hardware.

In the case of human error or a deliberate attempt to disturb normal operation conditions, automation tools and safety systems will guaranteedly stop the chain reaction and render the reactor plant safe without a possibility of external intervention in this process.

The plant provides for protection against all types of natural impacts—existing, predicted or hypothetically possible impacts in the deployment region. In this case, all impact values are taken with an additional margin in excess of the predicted values.

8. Plant Layout Arrangement

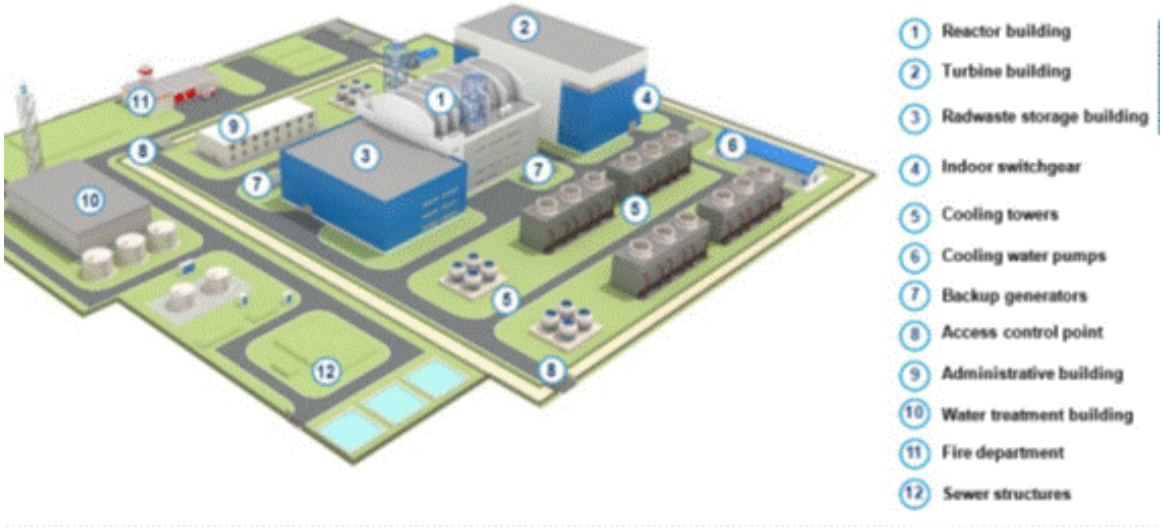


Figure 2: Basic layout of the SNPP with the RITM-200N reactors

The standard NOAK design assumes that two identical reactor plants be placed in the same reactor building. The basic layout for an SNPP incorporating the RITM-200N reactors is shown in Figure 2.

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 km2); for 220 MW, 38 acres (0.15 km2); for 330 MW, 49 acres (0.19 km2).

9. Testing Conducted for Design Verification and Validation

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.

10. Design and Licensing Status

The land-based configurations of SNPP with RITM-200N reactor plants are being developed for both domestic and international markets. The design and supporting documentation have been prepared to secure site and construction licenses for a single-unit SNPP in the area of Ust-Kuyga, the Republic of Sakha (Yakutia). In 2024, Rosatom State Corporation entered into an EPC-contract for the construction of an SNPP with RITM-200N reactor plants in the Jizzakh Region of the Republic of Uzbekistan. The two-unit SNPP with RITM-200N reactor plants will form a part of the Integrated Nuclear Power Plant (INPP), which will include two VVER units and two RITM units. The site-specific design documentation for the SNPP is underway. In 2026, a construction license was issued by the authorized regulatory body of the Republic of Uzbekistan and 'the first concrete pouring' milestone was successfully achieved.

11. Fuel Cycle Approach

Within the life cycle, fuel handling at the RITM-200N SNPP has differences and common points vs. large land-based stations. There are ship-related specific features of the refueling equipment, as well as the SNPP design and operation features that have an effect on the fuel handling process and cyclicity. Among other things, this also has to do with the dispersed fuel in use, which has high thermal conductivity and resistance to cyclic loads in order to ensure high maneuverability.

RITM-200N uses dispersion fuel (UO2 granules in a silumin matrix), which has high thermal conductivity and cyclic load resistance in order to ensure high maneuverability. The fuel-matrix-to-cladding metallurgical bonding eliminates the fuel rod failure mechanism that is caused by the pellet-cladding interaction and that is typical of large PWR fuel. The fuel cladding made of a radiation- and corrosion-resistant alloy eliminates such degradation and failure mechanisms as corrosion and hydriding.

The fuel life cycle includes the following stages: fuel loading into the reactor, fuel operation according to its intended purpose, unloading from the reactor into the spent fuel pool for three years, loading into a specialized storage and transportation container for reprocessing.

In order to ensure the heightened autonomy for the SNPP, the SNPP fuel cycle, unlike the approach used for the large PWRs, does not provide for partial refueling—the core is loaded entirely for a period of 5–6 years and completely replaced after the end of cycle. In addition, the coolant-dissolved boron absorber traditional for the large PWRs is not used to compensate for the excess reactivity in the SNPP. Under these conditions, in order to ensure acceptable power and burnup distributions, it is necessary to use welldeveloped fuel and burnable absorber physical profiling along with coolant flow hydraulic profiling for different fuel assemblies. Thus, the core is designed and supplied as a single product and no alternative fuel/fuel cycle options are provided.

Because no coolant-dissolved boron absorber is used to compensate for excess reactivity, it was made possible to lower the mass-and-size characteristics of the plant, to reduce the number of systems, to simply control, to make the LRW quantities smaller, as well as to prevent boron-dilution accident initiating events. At the same time, in order to ensure long-time operation without refueling, an adequate burnup reactivity margin should be provided, which is compensated for by the gadolinium-based burnable absorber rods and borated rods in fuel assemblies.

12. Waste Management and Disposal Plan

The discharged spent fuel assemblies are stored in a water-filled spent fuel pool, in special racks.

After their storage period is over, spent fuel assemblies are inserted in transportation containers and may be sent to the storage site.

The nuclear fuel handling equipment set ensures nuclear and radiation safety for fuel storage and transportation within the SNPP, as well as for shipping the spent fuel outside the SNPP.

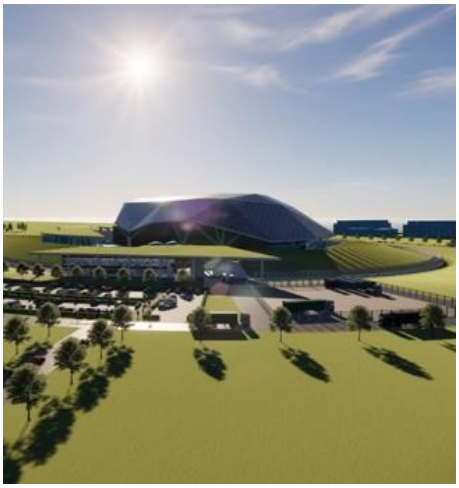
13. Development Milestones

Year	Milestone	Status
2018-2020	The conceptual design of a land-based SNPP was developed, SNPP feasibility studies were carried out for various sites in the Russian Federation	**Completed**
2020-2022	RITM-200N detailed design was developed	**Completed**
2024	The design documentation of the SNPP in the Republic of Sakha (Yakutia) was developed Start of documentation development for construction and licensing of the reactor plant and SNPP in the Republic of Uzbekistan	**Completed**
2026	Start of SNPP construction—first concrete pouring	**Ongoing**

Rolls-Royce SMR Limited, United Kingdom

Rolls-Royce SMR

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Rolls-Royce SMR Limited, United Kingdom*
Reactor type	*3-loop PWR*
Coolant/moderator	*Light Water / Light Water*
Design status	*Detailed Design*
Thermal/electrical capacity, MW(t)/MW(e)	*1,358 / 470*
NSSS operating pressure (primary/secondary), MPa	*15.5 / 7.4*
Core inlet/outlet coolant temperature (°C)	*295 / 322*
Primary circulation	*Forced circulation* *Natural circulation for backup decay heat removal*
Fuel type/assembly array	*UO2 pellet / 17 X 17 square*
Number of fuel assemblies in the core	*121*
Fuel enrichment (%)	*< 4.95*
Core discharge burnup (GWd/ton)	*55*
Refuelling cycle (months)	*18*
Fuel cycle requirements/approach	*Open/once-through*
Reactivity control mechanism	*Control rods*
Approach to safety systems	*Passive and active* *Site autonomy for at least 7 days*
Seismic design (SSE)	*0.3 g*
Design life (years)	*60*
Plant footprint (m^2)	*77,145*
RV height/diameter (m)	*7.82 / 4.20*
RV weight (metric ton)	*169*

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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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 UO2 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 Vessel (RV) 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.

(h) Distinguishing features

RR SMR's modularisation approach is based on maximising off-site manufacture and assembly in controlled factory conditions, followed by transport to site for final installation. The plant is broken down into standardised, repeatable modules, with modular solutions selected according to their scale, complexity, transport constraints and need for bespoke design. This requires module boundaries, interfaces, lifting arrangements, logistics and installation requirements to be considered early in the design process, rather than modularisation being treated solely as a construction-stage activity.

The principal benefits are improved build certainty, quality and repeatability, with less dependence on large-scale site construction. Factory production enables better-controlled working conditions, more consistent inspection and assurance, parallel manufacture of modules while site preparation progresses, and opportunities for learning across successive units. The approach can therefore reduce construction risk and site congestion, improve safety and productivity, support a more predictable programme and cost, and simplify future maintenance, replacement and decommissioning where suitable modular boundaries have been incorporated into the design.

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) 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.

(m) 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 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 heat sinks are provided that are each independently capable of supporting in-vessel retention.

(n) 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.

(o) 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 4E-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 suite of control and instrumentation (C&I) systems.

The duty and preventative control systems manage normal operations and anticipated transients across the power station, utilising a range of industrial programmable logic controllers or distributed control systems. A mix of analogue and non-programmable digital sensors will be utilised. Data will be exchanged via redundant digital data networks,

with redundant controllers providing a fault tolerant design. Opportunities to use smart devices and wireless technologies are being pursued.

Primary protection functions are delivered by a number of protection systems developed using a high-integrity programmable safety platform. The protection systems provide functions such as reactor shutdown, decay heat removal and containment in response to a fault.

Backup protection functions are provided using a diverse, hardwired platform. This hardwired platform uses non-programmable technology and, as such, provides a diverse means to initiate critical safety functions. The diverse protection systems operate entirely independently of the programmable protection systems.

Priority logic systems are provided to manage the interface with plant actuators. These systems will accept a range of demand signals received from various systems with different safety classifications, and determine the priority operation of actuator in accordance with safety prioritisation rules. They incorporate diversity to protect against common cause failure.

Accident monitoring systems are provided to allow the operator to monitor the status of the plant following an accident and take appropriate mitigating actions. In addition, an independent severe accident management system is provided to manage severe accidents. This system is backed by a separate battery and uses accident qualified instrumentation to provide a set of functions to manage the plant in severe accident conditions.

8. Plant Layout Arrangement

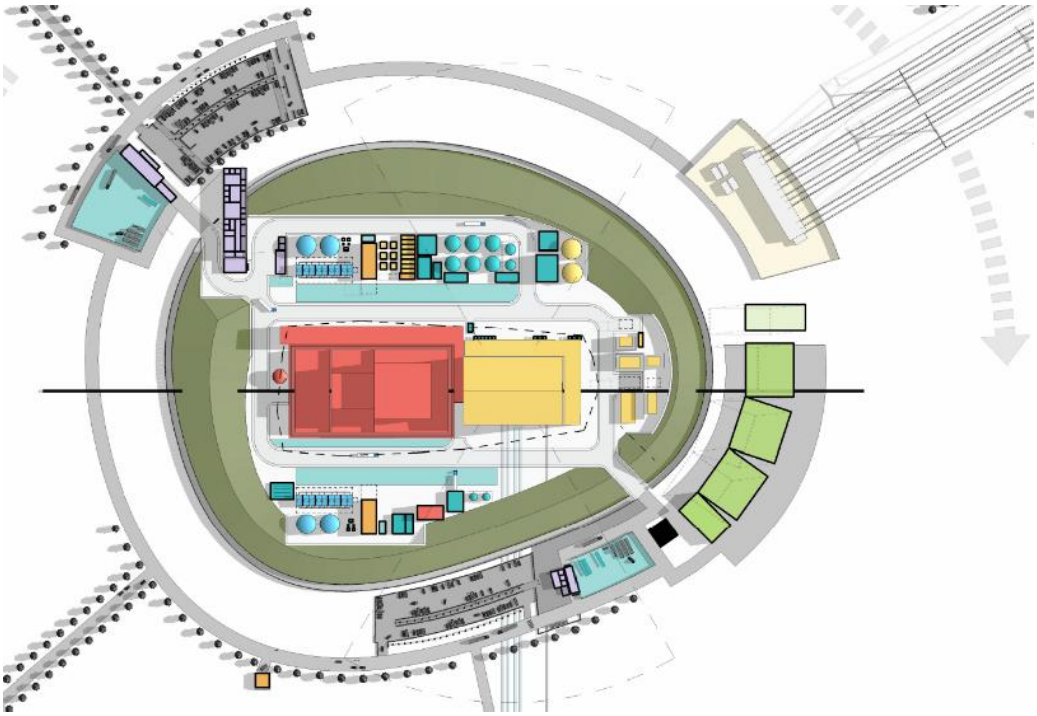


Figure 1. Rolls-Royce SMR Site Layout

The power station is designed for installation on an extensive range of inland and coastal sites, across a wide range of soil and earth conditions, whilst maintaining a compact site footprint. 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.

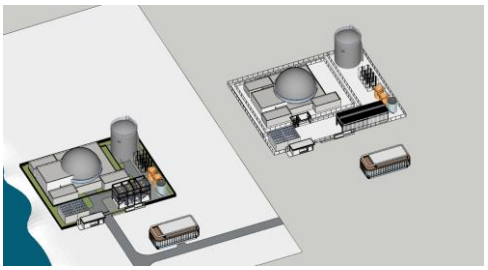
13. Development Milestones

Year	Milestone	Status
2015	Development of initial reference design	**Completed**
2017	Development of whole power station conceptual design	**Completed**
2022	Formal regulation entered in the UK	**Completed**
2026	Contracts signed for delivery of first units	**Completed**
Late 20s	Construction of first plant begins	**Planned**
Mid 30s	Commercial operation of first plant	**Planned**

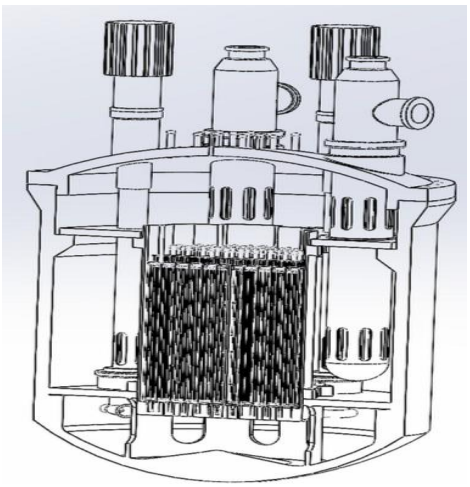
SAIHEAT, Singapore
SAIR



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*SAIHEAT, Singapore*
Reactor type	*PWR*
Coolant/moderator	*Light Water / Light Water*
Design status	*Pre-Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*200 / 45*
NSSS operating pressure (primary/secondary), MPa	*15.5 / 3.8*
Core inlet/outlet coolant temperature (^{o}C)	*285 / 310*
Primary circulation	*Forced*
Fuel type/assembly array	*UO2*
Number of fuel assemblies in the core	*89*
Fuel enrichment (%)	*5%*
Core discharge burnup (GWd/ton)	*40*
Refuelling cycle (months)	*36*
Fuel cycle requirements/approach	*once-through*
Reactivity control mechanism	*Control rods (B_4C), burnable absorbers, backup liquid absorber system*
Approach to safety systems	*Passive*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*3000*
RV height/diameter (m)	*4.0 / 2.5*
RV weight (metric ton)	*20*

1. Introduction

The reactor plant has been developed based on the experience of designing, manufacturing and operating transport nuclear power plants with maximum use of proven technical solutions for the main elements and systems and complies with regulations, standards and modern safety requirements.

The reactor plant uses an integral reactor. This ensures the compactness of the equipment and environments with natural or induced radioactivity and increases the reliability of the plant by reducing the communications under the pressure of the coolant of the primary circuit.

The reactor plant uses materials, parameters and characteristics of the environments that are widely used in world reactor construction. Together with the proven design elements of the main equipment (core, steam generator, pump, etc.), this allows using the accumulated scientific reserve in thermal hydraulics, properties of structural materials, corrosion, water chemistry, etc., eliminating the need for any research and limiting ourselves to performing a minimum amount of experimental design work when creating the lead model of the plant. The reactor plant is designed to operate as part of a ground-based small nuclear power plant, which, in addition to the reactor plant, includes a

turbogenerator, equipment for the automated control, monitoring and protection system for technical means, including equipment for regulating, managing and monitoring the generated electric power, auxiliary systems equipment and other equipment.

The concept provides for the delivery of all small nuclear power plant equipment in the form of single functionally tested units (modules) with high manufacturing quality at specialized machine-building enterprises.

The concept provides for the use of small nuclear power plants as a power source for data centers.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

A rational location for SNPPs is compact, remote, and hard-to-reach industrial areas or sites (facilities) and populated areas with the prospect of stable and growing demand for electric power. SNPPs can be used to provide targeted electricity and heat to both industrial production and the domestic sphere.

3. Design Philosophy

The design relies heavily on proven technologies, modular construction, and passive safety. The integral configuration reduces system complexity and improves safety by limiting potential leak paths. Factory fabrication and modular assembly enable high-quality manufacturing and efficient deployment.

4. Main Design Features

At the operating site, the modules are placed in accordance with the general plan of the station and are connected to each other by communications. To reduce the radiation background at the station site and protect the reactor plant from external influences, the reactor plant container is insta led inside a concrete building that serves as external biological protection. The concept of the small nuclear power plant project provides for the removal of the reactor plant module for refuelling, replacement of exhausted equipment and disposal at the end of its service life.

The procedure for transporting the reactor module with the loaded core adopted in the project does not meet the current nuclear and radiation safety standards and requires the development of a special regulatory framework for this type of small nuclear reactor. When designing the core, the main structural element was a fuel assembly with fuel rods with a metal ceramic composition of uranium dioxide in a zirconium alloy cladding. The size of the core, its composition and loading were selected to ensure the specified campaign and operating conditions. The coolant and moderator in the core is high-purity water with the addition of ammonia to generate hydrogen in it in order to prevent the formation of corrosive oxidative radiolysis products in it.

(a) Nuclear Steam Supply System

The reactor plant uses a built-in steam generator based on Field tubes. The design of the reactor plant uses an autonomous emergency cooldown system (ECS), designed to remove residual heat from the core both during scheduled shutdowns and in emergencies situations. This system consists of four independent sections, each of which includes an intermediate and an autonomous circuit designed to transfer heat from the coolant of the primary circuit to the coolant of the intermediate circuit, then to the coolant of the autonomous circuit and further into the atmosphere. All equipment of the primary circuit, with the exception of some equipment of the make-up system and emergency input of the absorber, is placed in a strong and sealed safety casing, within which, in case of design basis accidents associated with depressurization, the release of radioactive substances from the primary circuit system is localized. To prevent the leakage of the coolant of the primary circuit beyond the safety casing in the event of po ssible ruptures of the pipelines of the make-up system and emergency input of the absorber, shut-off valves are installed inside the safety casing, and check valves are insta led in the branch pipes connecting the pipelines of the system to the reactor. To retain radioactive products in case of beyond design basis accidents, the reactor plant is placed in a strong and tight protective shell, which is the main element of the reactor plant container body.

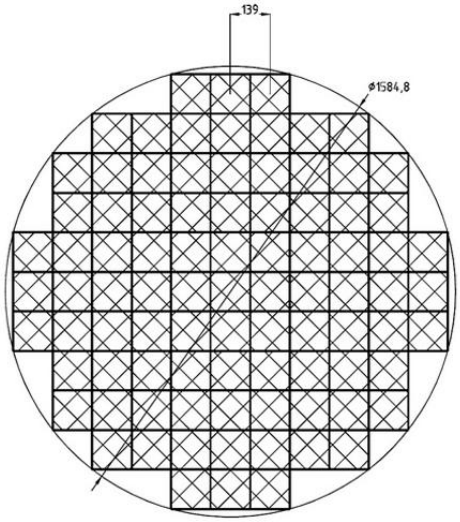
The protective shell together with the external biological protection ensures protection of the reactor plant from severe external impacts: hurricanes, tsunamis, aircraft crashes, etc. Steam pipelines, feedwater, electrical communications, etc. pass through the protective she l. The reactor plant ensures long-term and stable operation in the power range

from 20 to 100% of the nominal without limiting the number of power maneuvering cycles. All main equipment and systems are designed so that in the event of a single failure of any of them (except for the reactivity compensation organs), the reactor plant can continue to operate at a power level of at least 50% of the nominal. To prevent emergency situations from escalating into accidents and to reduce their possible consequences, the reactor plant is equipped with a number of technical safety features: emergency cooldown systems (ECS), emergency reactor cooling (ERC), make-up and emergency input of absorber, prevention of overpressurization of the reactor, steam generator and intermediate heat exchanger of the ECS, safety casing and protective shell, equipment cooling. All of them are passive (except for the makeup system and emergency input of the absorber), i.e. they are put into operation without operator intervention and do not require such intervention at least during the first 72 hours after the accident.

When developing the design concept of the reactor plant, primary attention was paid to issues of ensuring the reliability and safety of the SNPP operation at al stages of its life cycle. It was assumed that the SNPP meets safety requirements if its radiation impact on personnel, the population and the environment during normal operation and design basis accidents is significantly lower or, at least, does not lead to exceeding the established radiation doses of personnel and the population, as well as standards for emissions and the content of radioactive substances in the environment. In the event of beyond design basis accidents, these impacts should be limited to the maximum possible extent. The main technical solutions adopted in the reactor plant design are aimed at eliminating damage to the core beyond the established safe operating limits in al design basis accidents without personnel intervention or external assistance for at least 72 hours. The same task, if possible, should be solved for beyond design basis accidents caused by any considered initiating events and accompanied by postulated failures of electrical control systems and active systems requiring power supply for their operation.

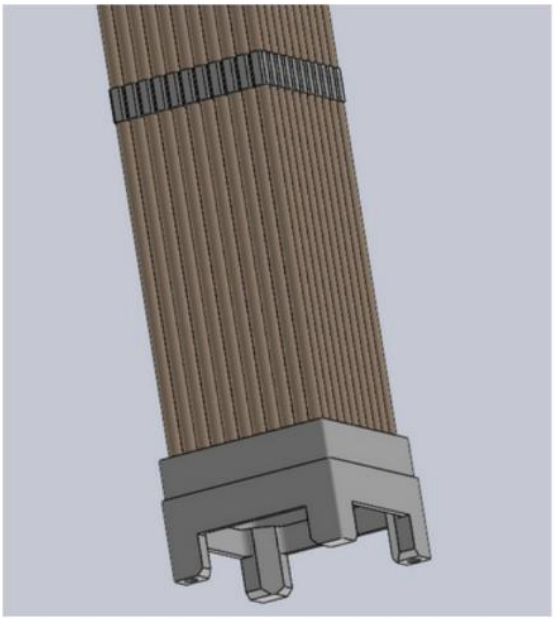
(b) Reactor Core

The fuel assembly of the basic design variant contains 121 fuel elements installed in nodes of a regular tetragonal grid. The fuel elements are self-distanced by mutual contact of the rib tops and are fixedly secured by their upper ends in the holders consisting of plates welded in pairs, and are spaced by the lower ends in the holders tail.



The holders are secured by locking half rings in the head and tail of the cassette. The fuel elements, during thermal and radiation expansions under operating conditions, have the ability to freely extend downwards in the axial direction. The head and tail of the fuel assembly are connected by a square cover. The cover material is zirconium alloy E110. A throttle device is installed in the tail of the fuel assembly, with the help of which its hydraulic characteristic is adjusted. The fuel assemblies are fixed by their tails in the lower plate of the removable reactor screen and have the ability to freely move relative to it in the axial direction during thermal and radiation expansions.





The design is similar to AFA3G Fuel Assembly. The fuel element is a cylindrical rod with four spacer ribs on its outer surface. To ensure self-distance in the bundle, the ribs are spirally twisted relative to the central axis with a pitch of 400 mm. Inside, in the center of the fuel element, there is a cross-shaped compensator with a wall thickness of 0.15 mm, made of zirconium alloy.

(c) Reactivity Control

Reactivity control is achieved through a combination of:

- Control rod assemblies using B_4C absorbers
- Burnable absorbers incorporated into the core
- Liquid absorber injection system (backup shutdown mechanism)

Control rods are driven by electromechanical systems designed to ensure rapid insertion by gravity and spring force, even under loss-of-power conditions.

(d) Reactor Vessel (RV) and Internals

The reactor vessel is a compact integral structure designed to contain all primary components. It uses proven materials (e.g., stainless steel) and incorporates internal structures such as:

- Core support structures
- In-vessel instrumentation

- Integrated flow paths for coolant circulation

This configuration enhances reliability and eliminates large external primary piping.

(e) Reactor Coolant System and Steam Generator

The primary circuit system is designed to remove heat from the reactor core and transfer it to the secondary circuit environment in the steam generator and to the intermediate circuit environment in the intermediate heat exchanger of the ACS. It consists of the main circulation path and the coolant volume change compensation system.

The primary circuit coolant circulation path is located inside the integral reactor vessel and includes:

- core;
- pump;
- steam generator;
- intermediate heat exchanger of the ACS;
- control and protection system drives;
- in-vessel devices;
- instrumentation sensors.

The coolant volume change compensation system includes a pressure compensator connected to the reactor by a pipeline. Nitrogen is used as the working medium in the volume compensation system. The primary circuit is connected (hydraulic connected) to the fo lowing fsafety, auxiliary and plant-wide systems:

- reactor emergency cooling system (system pipelines are connected to the pipelines of the makeup system and emergency absorber input);
- reactor overpressurization protection system (by one branch pipe on the reactor annular cover);
- makeup and emergency absorber input system (by three branches on the reactor annular cover);
- nitrogen supply system (through the gas main of the volume change compensation system);
- drainage and air removal system.

The primary circuit coolant circulation in the reactor is performed as fo llows. From the core distribution manifold, the main part of the coolant enters the fuel assemblies (FA) input, and the rest enters the interchannel space of the reactor core. The main part of the coolant heated in the active zone enters the traction section of the fuel assembly (suspension) and then, exiting the windows made in the fuel assembly suspensions, mixes with the coolant moving in the interchannel space. Then the entire coolant enters the collecting manifold of the active zone. From the collecting manifold of the active zone, the coolant enters the annular lifting section around the upper screen and, passing through the windows of the removable screen, enters the suction manifold of the pump and then to the inlet of the pump impeller. Having passed the impeller, the coolant turns 180 degrees and exits into the pressure manifold of the pump. From the pressure manifold of the pump, the coolant enters the pipe system of the steam generator and the intermediate heat exchanger SAR. Descending in the intertube space of the pipe system, the coolant gives off heat to the medium I or intermediate circuit and exits into the space of the side protective screens. Having passed it, the coolant enters the distribution manifold to the inlet of the active zone. Nominal characteristics of the primary circuit system: Reactor thermal power, MW 200, Parameters of the primary circuit: nominal pressure, MPa 15.5.

The secondary circuit system is designed to produce superheated steam from feedwater supplied by the ASMM turbogenerator unit by transferring heat from the primary circuit coolant to the steam generator. The secondary circuit system of the reactor plant includes a steam generator located inside the reactor vessel and pipelines with fittings outside its vessel. The system consists of four sections, each of which, in the event of a breach of the heat exchange surface tightness, can be disconnected by valves for steam and feedwater from the ASMM turbogenerator unit, which ensures localization of the primary circuit leak in the secondary circuit within the secondary circuit pipelines from the steam generator to the localizing fittings. In case of failure of any of the valves of the SG sections, common valves for each pair of sections are additionally installed on the steam and feedwater pipelines. The sections of the steam pipeline and the feedwater pipeline, including the localizing fittings, are designed for th e primary circuit pressure that may occur during inter-circuit depressurization. The search for the current section is carried out using a special system for monitoring the density of the heat exchanger tube surface, for which a corresponding steam extract ion into this system is provided behind the steam valves of each section. As a backup method, it is possible to conduct a search by sequentially disconnecting sections with valves while monitoring the activity at the ejector exhaust, the pressure in the disconnected section and the level in the reactor. Drain valves are connected to the system of the second circuit behind the section valves. Before the section valves on the feedwater pipelines, a connection of a safety automatic device (SAD) of the system for preventing overpressurization of the steam generator is provided.

(f) Pressuriser

The pressurizer is a separate vessel connected to the reactor, used for maintaining primary system pressure. It operates using a gas (nitrogen) cushion to accommodate coolant volume changes and ensure stable pressure control under varying operating conditions.

(g) Primary pumps

The reactor employs multiple electrically driven primary circulation pumps (sealed design), which have demonstrated reliability through extensive operational experience.

The system is also designed such that, in the event of pump failure:

- Natural circulation can sustain decay heat removal
- The reactor can operate at reduced power (~20%)

(h) Distinguishing features

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The SAIR reactor adopts a defence-in-depth safety approach, combining inherent safety characteristics with engineered systems. The design relies primarily on passive safety systems that operate based on natural physical processes (gravity, natural circulation, heat conduction), thereby reducing dependence on active components and operator actions.

Key safety systems include:

- Passive emergency reactor cooling system (ERCS)
- Passive residual heat removal/autonomous cooldown system
- Systems for overpressure protection and coolant inventory control

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

For design basis conditions such as loss of coolant accidents (LOCA), station blackout (SBO), and equipment failures, the reactor relies on a combination of inherent safety features and passive systems.

Key characteristics include:

- Integral reactor configuration, minimizing large primary pipe breaks
- Natural circulation capability, ensuring core cooling even with pump failure
- Passive heat removal systems, capable of removing decay heat for at least 72 hours without operator intervention
- Automated shutdown using gravity-driven control rod insertion

(m) Safety Approach and Configuration to Manage DEC

For beyond design basis and severe accident conditions, the design incorporates features to limit consequences and maintain containment integrity.

These include:

- Safety casing (high-pressure vessel) enclosing the primary circuit to localize coolant release
- Protective shell (containment) to prevent radionuclide release to the environment
- Passive systems for long-term heat removal and core cooling
- Multiple physical barriers preventing radioactive material release

(n) Containment System

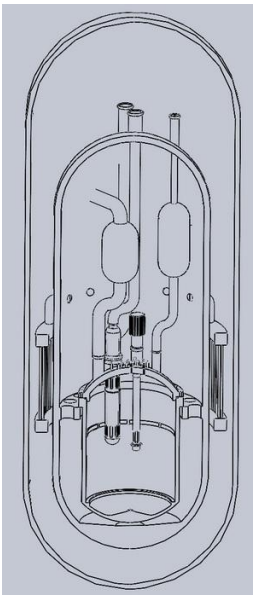
When developing the reactor design, primary attention was paid to ensuring the reliability and safety of the SNPP operation at all stages of its life cycle. It is assumed that the SNPP meets safety requirements if its radiation impact on personnel, the population and the environment during normal operation and design basis accidents is significantly

lower, or, at least, does not lead to exceeding the established radiation doses of personnel and the population, as well as the standards for emissions and content of radioactive substances in the environment. In the event of beyond design basis accidents, these impacts must be limited to the maximum possible extent. For this purpose, a system of safety barriers is installed on the path of radionuclide propagation, including:

- fuel matrix;
- sealed fuel rod cladding;
- sealed primary circuit - reactor vessel;
- safety casing;
- localizing valves;
- protective shell.

The first two barriers retain the most dangerous part of the radioactive products - fission fragments. But even if they perform their role completely, a significant amount of products with induced radioactivity accumulates in the primary circuit. There are fission fragments in the primary circuit water caused by the presence of some initial surface contamination of the fuel elements, as well as radioactive noble gases penetrating through microcracks in the fuel element cladding.

The situation is complicated if the third barrier - the primary circuit boundary - loses its tightness. This is the main accident leading to deterioration of safety conditions. Therefore, when designing a small nuclear reactor, maximum attention is paid to preventing the possibility of such accidents. The most fundamental measure in this direction is the use of an integral design of a nuclear reactor with a minimum of external boundaries of the primary circuit. Along with external boundaries, the primary circuit also has internal boundaries separating it from the coolants of the secondary circuit, intermediate circuits of the SAR, the cooling circuit, etc. Loss of tightness of these boundaries is fraught with the spread of radioactive media beyond the reactor facility.



To prevent such situations, localizing valves are installed on these circuits, which are closed at the time of an accident. In addition, the reactor plant has systems hydraulically connected to the primary circuit: makeup, pressure compensation, emergency reactor cooling, prevention of primary circuit overpressurization, etc. Loss of tightness by them is often equivalent to LOCA accidents. To localize the consequences of such accidents, as well as to maximally contain radioactive media from spreading during repair work and reloading the core, modern NPP designs provide for protective shells, inside which the main equipment of the reactor plant is located. They have the strength to withstand the internal pressure that increases during an accident, and the tightness to prevent the spread of radioactivity.

A protective shell is used to localize the spread of radionuclides during repair work, reloading of the core and in beyond-design-basis accidents, as well as to protect the reactor plant equipment from severe external impacts: hurricanes, tsunamis, aircraft crashes, etc. The safety casing is a tightly sealed high-pressure vessel made of ANSI304 steel, manufactured and tested for strength and density in the conditions of the machine-building plant-manufacturer, which guarantees its high quality. It can be manufactured with any predetermined strength and ensures a high level of tightness. The safety casing limits and stops the coolant leakage from the primary circuit during LOCA accidents by equalizing the pressure between the reactor and the safety casing and, together with the ACS and ECCS, keeps the core below the coolant level for at least 72 hours without corrective actions by the operating personnel.

(o) Spent Fuel Cooling Safety Approach / System

6. Plant Safety and Operational Performances

The SAIR reactor is designed to achieve a high level of safety, reliability, and operational flexibility, based on inherent safety characteristics, passive systems, and proven engineering solutions.

The plant incorporates an integral reactor configuration, which significantly reduces the number of primary circuit penetrations and eliminates large external piping, thereby minimizing the risk of loss-of-coolant accidents (LOCAs) and improving overall operational reliability.

A key safety feature is the use of passive safety systems that ensure decay heat removal and core cooling without operator intervention or external power supply for at least 72 hours following an accident. These systems operate based on natural physical principles such as gravity, natural circulation, and heat transfer.

The reactor core is designed with negative reactivity coefficients (temperature, density, and power) across the entire operating range, providing inherent self-regulation and stability during normal operation and transient conditions.

7. Instrumentation and Control Systems

The SAIR reactor uses an Advanced Process Control System (APCS) to monitor and control all plant operations under normal and accident conditions. The system includes a reactor control and protection system with redundant shutdown capability, along with real-time monitoring of key parameters such as temperature, pressure, and neutron flux. A centralized human–machine interface enables operator control and supervision, while the design incorporates redundancy, independence, and fail-safe principles to ensure high reliability and safety.

8. Plant Layout Arrangement



Figure 1. SAIR Plant Layout Arrangement

A rational location for SNPPs is compact, remote, and hard-to-reach industrial areas or sites (facilities) and populated areas with the prospect of stable and growing demand for electric power. SNPPs can be used to provide targeted electricity and heat to both industrial production and the domestic sphere. SNPPs must comply with all safety requirements established by government agencies and international standards. This includes protection from external influences, reliability of safety systems, ncontrol over background radiation, and accident prevention.

The choice of location for SNPPs should take into account geological, hydrological, and climatic conditions in order to minimize the risks associated with natural disasters (earthquakes, floods, landslides, etc.). Distance to populated areas: There are standards that determine the minimum permissible distance between SNPPs and the nearest populated areas. These distances depend on the power of the plant and the mtype of reactor. Power and water supply: SNPPs need a reliable source of electricity for their own supply and sufficient water for reactor cooling. Therefore, proximity to power lines and water bodies is an important factor. Environmental aspects: When designing and constructing SNPPs, it is necessary to take into account the impact on the environment, including radiation background, emissions into the atmosphere and discharges into water bodies. It is important to conduct environmental monitoring and take measures to minimize the negative impact on nature.

9. Testing Conducted for Design Verification and Validation

A protective shell is used to localize the spread of radionuclides during repair work, reloading of the core and in beyond-design-basis accidents, as well as to protect the reactor plant equipment from severe external impacts: hurricanes, tsunamis, aircraft crashes, etc. The safety casing is a tightly sealed high-pressure vessel made of ANSI304 steel, manufactured and tested for strength and density in the conditions of the machine-building plant-manufacturer, which guarantees its high quality. It can be manufactured with any predetermined strength and ensures a high level of tightness.

The safety casing limits and stops the coolant leakage from the primary circuit during LOCA accidents by equalizing the pressure between the reactor and the safety casing and, together with the ACS and ECCS, keeps the core below

the coolant level for at least 72 hours without corrective actions by the operating personnel. The pressure in the safety casing during LOCA accidents is reduced by condensing the steam-gas mixture on cooled surfaces inside the safety casing.

10. Design and Licensing Status

The SAIR reactor is currently in the pre-conceptual design stage, with licensing under review.

11. Fuel Cycle Approach

The reactor uses UO_2 fuel with ~5% enrichment in a once-through fuel cycle, with a refuelling interval of approximately 36 months. No reprocessing is currently foreseen.

12. Waste Management and Disposal Plan

Not provided

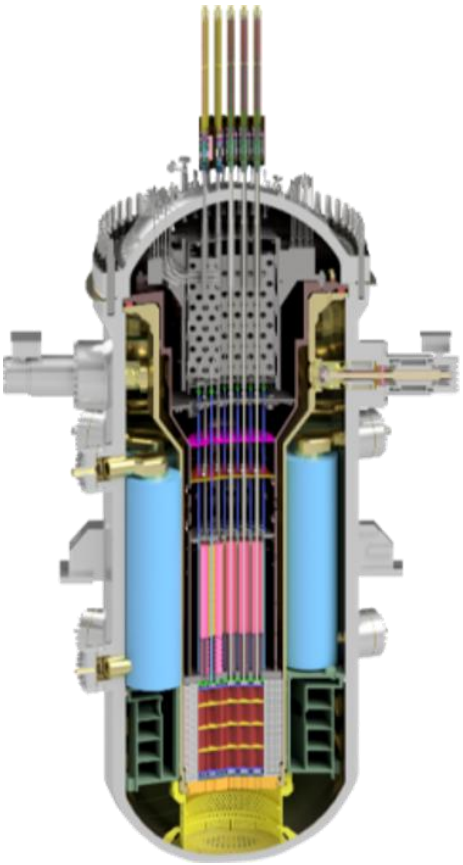
13. Development Milestones

Year	Milestone	Status
2025	Concept design	Ongoing
2027	Technical project	Planned

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



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*KAERI, Republic of Korea K.A.CARE, Kingdom of Saudi Arabia*
Reactor type	*Integral PWR*
Coolant/moderator	*Light Water / Light Water*
Design status	*Detailed Design*
Thermal/electrical capacity, MW(t)/MW(e)	*365 / 107*
NSSS operating pressure (primary/secondary), MPa	*15 / 5.8*
Core inlet/outlet coolant temperature (°C)	*296 / 322*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO2 pellet / 17 X 17 square*
Number of fuel assemblies in the core	*57*
Fuel enrichment (%)	*< 5.0*
Core discharge burnup (GWd/ton)	*< 54.0*
Refuelling cycle (months)	*30*
Fuel cycle requirements/approach	*Conventional LWR requirements applied (spent fuel storage capacity: 30 years)*
Reactivity control mechanism	*Control rod drive mechanisms, burnable poison and soluble boron*
Approach to safety systems	*Passive ≥72h*
Seismic design (SSE)	*> 0.3 g with 0.18 g of automatic shutdown*
Design life (years)	*60*
Plant footprint (m^2)	*90,000*
RV height/diameter (m)	*16.8 / 6.0*
RV weight (metric ton)	*1070 (including coolant)*

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

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☒ Load-following	☒PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

The SMART is a multi-purpose reactor for electric power generation, hydrogen production using PEM, 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 UO2 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 UO2 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 Vessel (RV) 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 and 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.

(f) Pressuriser

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

(g) 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.

(h) Distinguishing features

The SMART reactor has achieved a high level of design under two SDA licensing. Its design flexibility enables adaptation to a various customer requirement such as a deployment near data center with reduced EPZ, floating power system for offshore application. Specially, single to multiple units can be arranged in a block configuration to accommodate the various power demands of AI DC or high-tech complex. SMART also provides load-following capability in response to abrupt power demand by renewable energy, enabling a harmonization with the renewable energy as a balanced energy mix.

5. Safety Features

(i) 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.

(j) 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.

(k) 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.

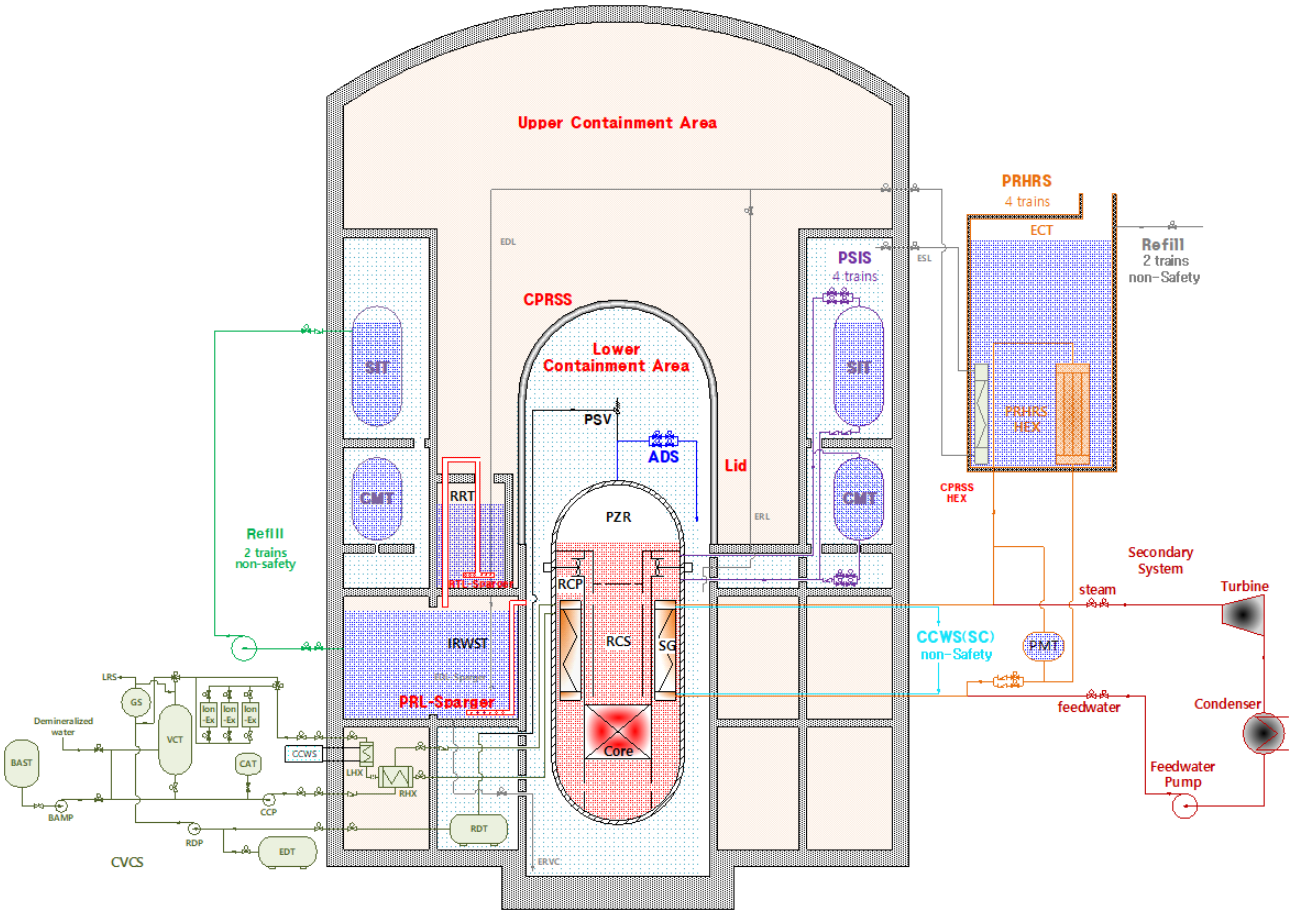


Figure 1. Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

SMART addresses DBCs through a defense-in-depth design concept and engineered safety features (ESFs). SMART considers a broad spectrum of anticipated operational occurrences (AOOs) and postulated accidents (PAs) that may occur in the reactor systems as DBCs. In the event of a DBC, the Passive Residual Heat Removal System (PRHRS), consisting of four independent trains, removes decay heat generated in the core and stored energy from the system to the ultimate heat sink. The reactor coolant inventory is also replenished by the Passive Safety Injection System (PSIS), which likewise consists of four independent trains. The operation of these passive safety systems is supported by the Automatic Depressurization System (ADS). Through the operation of these highly reliable engineered safety features, the integrity of the reactor core and Reactor Coolant System (RCS) is maintained, thereby protecting the public and the environment from potential radiological consequences.

(m) Safety Approach and Configuration to Manage DEC

The passive safety systems of SMART are designed to achieve the objectives of preventing and mitigating severe accidents under Design Extension Conditions (DECs) by employing highly reliable passive safety features. The passive safety systems provided for this purpose include the Passive Residual Heat Removal System (PRHRS), Automatic Depressurization System (ADS), Passive Safety Injection System (PSIS), and Containment Pressure and Radioactivity Suppression System (CPRSS).

The PRHRS removes heat from the RCS by natural circulation under emergency conditions in which normal steam extraction or feedwater supply through the secondary system is unavailable. The ADS depressurizes the RCS, enabling the early injection of coolant through the PSIS. The PSIS is designed to provide adequate core cooling even during low-probability accident conditions. The CPRSS limits increases in pressure and temperature within the containment and removes fission products released into the containment atmosphere.

If core damage occurs despite these measures, cooling water is injected into the reactor cavity to cool the external surface of the reactor vessel, thereby preventing reactor vessel failure. Maintaining reactor vessel integrity in this manner can arrest or significantly limit the progression of a severe accident. For severe accident mitigation under DEC conditions, systems and features such as the ADS, Cavity Flooding System (CFS), and Passive Autocatalytic Recombiners (PARs) are employed.

(n) 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.

(o) Spent Fuel Cooling Safety Approach / System

The safety-related function of cooling and shielding the fuel in the spent fuel pool (SFP) is performed by the water in the pool. The SFP is designed that a water level maintained above the spent fuel assemblies for at least 7 days following a loss of the SFP cooling and cleanup system (SFPCCS), using only onsite makeup water. Even if the SFPCCS is inoperable under abnormal operation, the SFP water level will be maintained above the top of the spent fuel assemblies for at least 72 hours without the aid of safety-related makeup water.

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 Systems

High reliability and performance of I&C systems is achieved using advanced features such as digital signal processing, remote multiplexing, signal validation and fault diagnostics, and sensing signal sharing for protection and control system. The ex-core neutron flux monitoring system consists of safety and start-up channel detectors which are located within the RPV, and digital signal processing electronics. The in–core instrumentation system consists of 29 detector assemblies which are developed as mini type for SMART with four stacked rhodium self-powered neutron detectors(SPNDs).

8. Plant Layout Arrangement

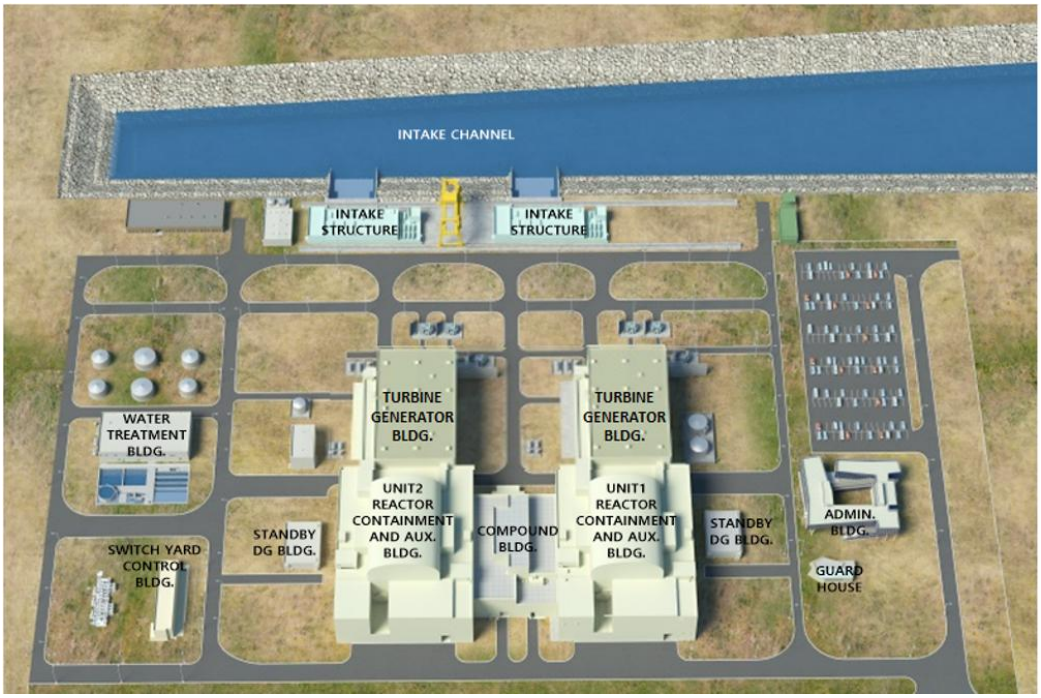


Figure 2. Plant Layout Arrangement

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.

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 with 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. In September 2024, the NSSC granted Standard Design Approval for SMART100.

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

Year	Milestone	Status
1997	Conceptual Design Development	**Completed**
2002	Basic Design Development	**Completed**
2009	SMART-PPS (Pre-Project Service)	**Completed**
2012	Technology Verification, Standard Design Approval (SDA)	**Completed**
2012	SMART Safety Enhancement Project	**Completed**
2015	SMART Pre-Project Engineering Agreement Signed Between KAERI and K.A.CARE for the Global Commercialization of SMART	**Completed**
2019	SMART Pre-Project Engineering Completed	**Completed**
2019	SMART100 Standard Design Approval Applied	**Completed**
2024	SMART100 Standard Design Approval Issued	**Completed**

Holtec International, United States of America SMR-300



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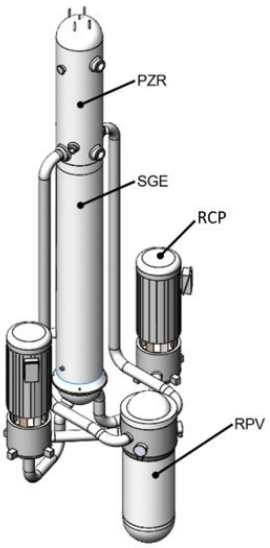


Figure 1: SMR-300 Primary Circuit



Figure 2: SMR-300 Fuel Assembly

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Holtec International, United States of America*
Reactor type	*PWR*
Coolant/moderator	*Light Water / Light Water*
Design status	*Detailed design*
Thermal/electrical capacity, MW(t)/MW(e)	*1050 MWth (nominal) / 385 MWe gross / 340 MWe net (typical)*
NSSS operating pressure (primary/secondary), MPa	*15.5 / 6.2*
Core inlet/outlet coolant temperature (°C)	*286 / 321*
Primary circulation	*Forced (2 RCPs in normal operation); Natural (for decay heat removal)*
Fuel type/assembly array	*UO2 pellet / square array (17x17)*
Number of fuel assemblies in the core	*69*
Fuel enrichment (%)	*4.9% (average) / 5.0% (max)*
Core discharge burnup (GWd/ton)	*51 (average)*
Refuelling cycle (months)	*18 (capability to longer cycles)*
Fuel cycle requirements/approach	*Once-through*
Reactivity control mechanism	*Soluble Boron and Control Rods*
Approach to safety systems	*Redundant robust passive safety systems, requiring no operator action for accident mitigation*
Seismic design (SSE)	*0.4g (reference plant) to 0.6g*
Design life (years)	*80 (expected service life: 100 years)*
Plant footprint (m^2)	*> 50,000*
RV height/diameter (m)	-
RV weight (metric ton)	-

1. Introduction

Holtec International SMR-300 is an advanced pressurized water reactor (PWR) with a design informed by decades of operating reactor experience and industry lessons learned, able to provide clean and affordable power with passive safety systems and improved safety compared to presently operating nuclear plants. The plant design specification is risk and value engineering-informed to facilitate a readily licensable and competitive plant product embodiment, planned for deployment both in the United States and international markets.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☒ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

The primary objective of the SMR-300 design is to produce electricity safely with optional cogeneration equipment (i.e., hydrogen generation, thermal energy storage, district heating, seawater desalination, or process heat applications). The turbine generator is intended for base load operation and has the ability to include load following capability into the design if required by the electric grid regulations, depending on plant location.

3. Design Philosophy

SMR-300's design philosophy prioritizes safety and simplicity by incorporating advanced passive safety systems that require no operator intervention or rely on off-site or on-site AC power sources for accident mitigation. This approach ensures that the reactor operates efficiently and securely with minimal operational challenges.

SMR-300's modularity, with associated construction components, methods, and means, offers promising opportunities to speed nuclear power plant construction time with associated reduced field time and costs, while further enhancing standardization and quality of the final power plant product. The essence of SMR-300 modularity is the application of Concrete Strengthened Steel Modules (CSSMs). CSSMs are factory-fabricated civil structural embodiments evolved from decades of similar Holtec-designed, licensed, fabricated, and constructed HI-STORM overpacks.

Furthermore, SMR-300 Smart Modular Advanced Reactor Technology (SMART) incorporates a compact reactor coolant system and minimal loop piping for fast, efficient construction, ease of operation, and maintenance. With an in-containment spent fuel pool and integrated used fuel management and storage solution and componentry, SMR-300 addresses the entirety of the plant's fuel lifecycle with Holtec's proven and globally deployed multi-purpose canister system and equipment.

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 (SGE) with an integral pressurizer (PZR) stacked on top. 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 SMR-300 core employs a standard 17x17 PWR fuel assembly into a roughly cylindrical shape, with 69 total assemblies. The SMR-300 design implements a fuel management program that uses fuel assemblies with varying levels of burnup to optimize power shape; strategic placement of each assembly within the core will reduce peaking.

The core is designed for a nominal 18-month cycle length with flexibility for longer or shorter cycles depending upon utility energy requirements. The design always ensures a sufficient margin to the Specified Acceptable Fuel Design Limits throughout the cycle.

(c) Reactivity Control

The SMR-300 design uses Rod Cluster Control Assemblies (RCCAs) and soluble boron for reactivity control. Use of gadolinia as an integral burnable absorber within the fuel assemblies and strategic arrangement of fuel assemblies with varying levels of burnup will further support reactivity control and optimize the power shape. Combining RCCAs with other reactivity control methods aligns with a standard, well-proven approach to PWR reactivity control and limits the need for rod movement during the operating cycle. Boron concentration is controlled by the Chemical and Volume Control System (CVC) and is generally used for slower, longer-term adjustments. RCCAs are used for rapid reactivity control and shutdown, and for temperature control during power manoeuvres. This approach ensures effective management of reactor reactivity throughout its operating cycle.

(d) Reactor Vessel (RV) and Internals

The reactor pressure vessel (RPV) is a thick-walled cylindrical pressure vessel with an integrally welded bottom head and a removable top head. The RPV is designed to be compact and robust, housing the core, primary coolant system, and the internals (upper and lower). 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

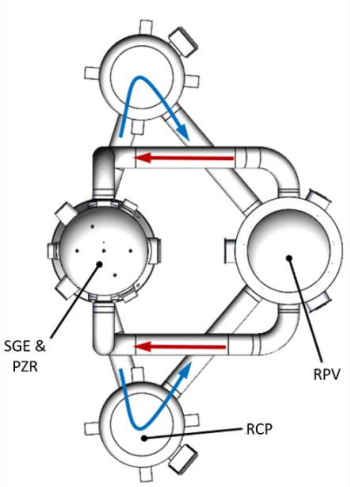


Figure 3: SMR-300 Primary Loop Flow Paths

The SMR-300 consists of one RPV and one Steam Generator (SGE) connected by two hot legs and two cold legs. Each cold legs each have a Reactor Coolant Pump (RCP) to provide forced RCS flow during normal operations to maximize power output. The major components of the RCS include the RPV, SGE with integral PZR, and the RCPs. All the RCS equipment, piping, and components are located inside the Containment Structure (CS).

The SMR-300 features a single, vertical, straight tube, once-through, counter-current flow SGE with reactor coolant flowing down the inside of thermally treated Inconel 690 tubes and the secondary liquid/steam flowing through the shell side of the SGE. The two RCS hot legs route reactor coolant from the RPV to the top tube sheet of the SGE.

The system ensures efficient heat removal and transfer to the SGE. The integrated design of the SGE and PZR 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 (PZR) is integral to the steam generator; an internal divider plate separates the circulating RCS volume from the saturated PZR volume, maintaining the pressure of the reactor coolant system. This design reduces system complexity, eliminating the need for power-operated relief valves and enhancing 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, increasing operational margins and resulting in a more reliable plant.

(g) Primary pumps

The SMR-300 uses two vertically mounted reactor coolant pumps (RCPSs) 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, with a flywheel that extends flow coast down after a loss of pump electrical power (to maintain flow through the core), and a controlled pump shaft sealing system that minimizes leakage from the pump into the containment, including during a station black-out event.

(h) Distinguishing features

Passive safety systems and active non-safety systems; smart modular design; flexible configurations; UHS and spent fuel pool within containment (no transfer channel); integrated dry spent fuel storage; etc

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The SMR-300 incorporates multiple levels of defense-in-depth to remove heat from the reactor and assure safety. It employs advanced passive safety systems designed to function without operator intervention or external power sources and maintain the reactor in a safe shutdown condition. These systems are in tegrated into the reactor's robust containment structure, enhancing overall safety. The design minimizes the likelihood of accidents and ensures an effective response in emergency scenarios.

The systems are simpler than most current operating reactors. By eliminating active pumps and other large equipment from safety service, the electrical systems are also made more reliable, using only low-voltage power supplied from batteries.

Passive safety systems provide heat removal for the reactor core and spent fuel pool (SFP) for 72 hours without the need for operator action, before transitioning to long-term cooling to safely remove decay heat for at least 30 days. All

makeup water needed for a postulated LOCA is inside containment, making the containment fully isolable. This minimizes the dose to the public and the effects on the environment from this event.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

For Design Basis Events (DBCs), the SMR-300 incorporates redundant safety features and diverse pathways for heat removal 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.

(m) 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.

In addition, several active systems can be used to mitigate DECs, connections for diverse and flexible coping strategies (FLEX), and passive autocatalytic recombiners for hydrogen management are incorporated into the design based on the lessons learned from Fukushima.

(n) 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 annular water reservoir aids in cooling and managing decay heat during accidents.

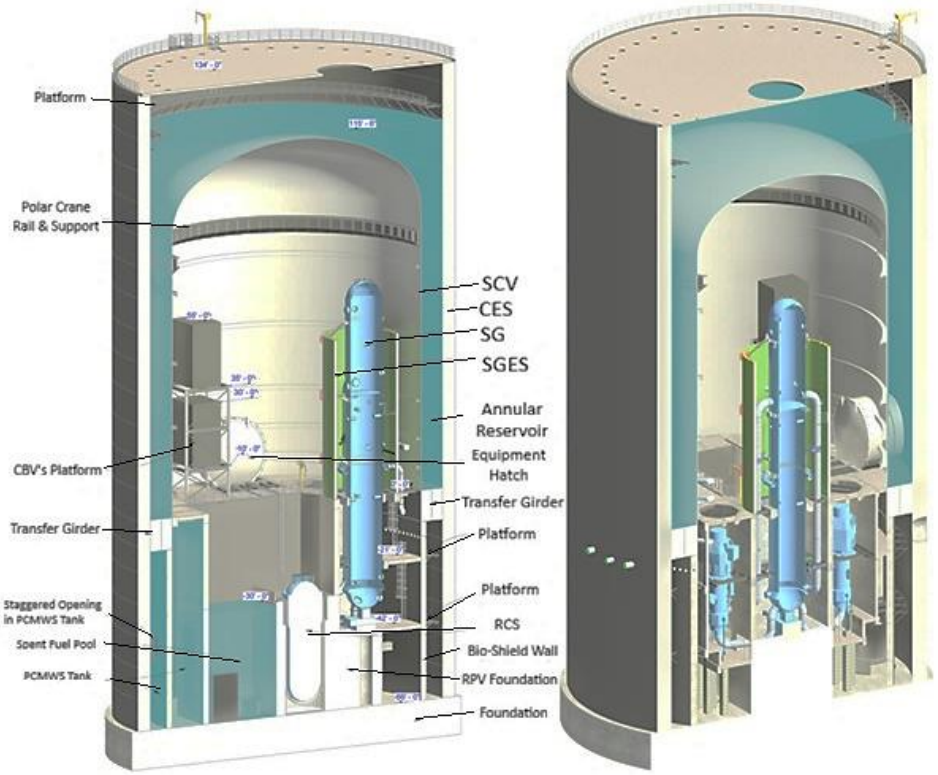


Figure 4: Arrangement of the SMR-300 Containment Enclosure Structure, Annular Reservoir, and Containment Structure

The containment structure (CS) is a cylindrical steel containment vessel (above grade); below grade, the CS is formed from concrete and steel prefabricated modules, including its foundation. The CS enhances overall plant safety and environmental protection.

(o) Spent Fuel Cooling Safety Approach / System

Spent fuel cooling is managed through a small Spent Fuel Pool (SFP) located within the reactor containment, providing initial cooling and radiation shielding. SFP criticality is controlled using Holtec's patented neutron absorber, Metamic™, in a conventional stainless steel spent fuel rack geometry. Long-term cooling is achieved using dry cask storage (Holtec's HI-STORM UMAX system), 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 along with an I&C system with multiple levels of anticipatory reactor trip signals and highly reliable active non-safety features to prevent and, if necessary, mitigate the consequences of DBCs and DECs. After a postulated accident, the plant is designed to automatically achieve and maintain a safe shutdown state without operator actions, without external water, without external power, and without active systems.

The reactor's design ensures minimal operator intervention and simplified maintenance procedures while maintaining high reliability and efficiency, with a calculated lifetime capacity factor of more than 95%. 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 I&C architecture is based on a USNRC-accepted approach that implements overarching design principles of redundancy, independence, diversity, determinism, and simplicity. The SMR-300 utilizes the mature (proven) Mitsubishi Electric 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



Figure 5: Rendering of the first two Holtec SMR-300 units at the Palisades Energy site in Covert, Michigan.

The SMR-300 features a compact and efficient plant layout designed to maximize space utilization and simplify construction, supported by modular construction. Major components, including the reactor core, steam generator, and safety systems, are arranged to facilitate easy maintenance and operational efficiency. The site layout ensures that nuclear assets, including the reactor core, are deeply embedded for protection against hazards, and unauthorized access is prevented by multiple barriers. The layout supports streamlined operations and reduces the need for extensive infrastructure.

The design ensures that site boundaries such as the Exclusion Area Boundary (EAB), Low-Population Zone (LPZ), and Emergency Planning Zone (EPZ) can be as close to the site boundary as possible.

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 and is undergoing licensing in the USA under the 10 CFR Part 50 two-step process, consisting of a Construction Permit Application (CPA) followed by an Operating License Application (OLA). As part of this process, the Preliminary Safety Analysis Report (PSAR), Limited Work Authorization (LWA), and Environmental Report (ER) were submitted to the U.S. Nuclear Regulatory Commission (NRC) at the end of 2025, completing the first portion of the Construction Permit Application (CPA-1).

In parallel, Holtec is advancing the SMR-300 through the UK Generic Design Assessment (GDA) with the Office for Nuclear Regulation (ONR), providing regulator-facing validation under IAEA-aligned safety principles.

The FOAK dual-unit SMR-300 deployment at Palisades received support from DOE's ARDP and the 2025 $400 million Tier 1 "First Mover" award to accelerate deployment under a standardized fleet model. The reactor is expected to enter commercial operation in 2031. 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, with the capability to accommodate longer cycles. 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 fuel management strategies incorporate the Holtec International designed and manufactured new fuel storage vault (dry), spent fuel pool racks (wet), and dry storage cask system (HI-STORM FW/UMAX) to safely handle and store spent fuel assemblies with a high margin of safety. 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

Year	Milestone	Status
February 2024	SMR-300 FOAK site selection (Palisades site in MI, USA)	**Completed**
Dec. 2025	Submission of Construction Permit Application Part I for the Palisades site. CPA-1: Limited Work Authorization (LWA) and Preliminary Safety Analysis Report (PSAR)	**Completed**
Jan. 2026-June 2027	Preconstruction Activities (tree clearing, grading…)	**Ongoing**
Sept. 2026	Submittal of Full Construction Permit Application Part II for the Palisades site. CPA-2	**Ongoing**
Dec. 2028	Submittal of Operating License Authorization (OLA)	**Planned**
June 2030	Completion of Construction	**Planned**
2031	Commercial Operation Date (COD)	**Planned**

STAR ENERGY SA, Switzerland
STAR



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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*
Design status	*Basic Design*
Thermal/electrical capacity, MW(t)/MW(e)	*30 / 10*
NSSS operating pressure (primary/secondary), MPa	*12.5 / 4.4*
Core inlet/outlet coolant temperature (^{o}C)	*270 / 300*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO2 pellet assemblies in individual pressure channels*
Number of fuel assemblies in the core	*138*
Fuel enrichment (%)	*19*
Core discharge burnup (GWd/ton)	*60*
Refuelling cycle (months)	*120*
Fuel cycle requirements/approach	*Once-through HALEU; no on-site storage*
Reactivity control mechanism	*Control rods and Er / Gd2O3 burnable absorber*
Approach to safety systems	*Passive*
Seismic design (SSE)	*Degree VIII (MSK-64)*
Design life (years)	*60*
Plant footprint (m^2)	*36 000*
RV height/diameter (m)	*7.6 / 2.3*
RV weight (metric ton)	*20*

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 UO2 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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☒PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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) Nuclear Steam Supply System

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 MWt. 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.

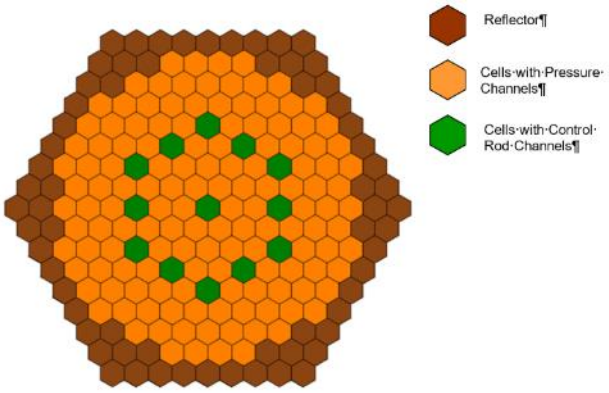


Figure 1: STAR Reactor Core

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 Gd2O3 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

(c) Reactivity Control

Reactivity control is performed with burnable erbium and gadolinium oxide absorbers in a number of the fuel assemblies and 10B 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.

(d) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

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.

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.

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.

(f) Pressuriser

N/A

(g) Primary pumps

N/A

(h) Distinguishing features

Compact, low-weight design without a pressure vessel; flexibility of operation; intrinsic passive safety.

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

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.

(m) Safety Approach and Configuration to Manage DEC

N/A

(n) 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.

(o) 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.

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 Systems

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.

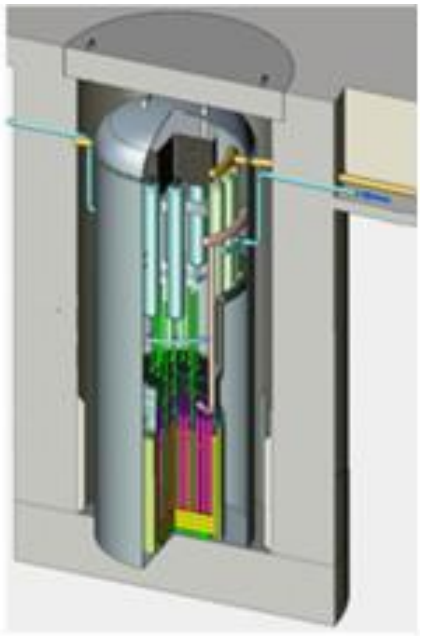


Figure 2:“STAR” module placement option inside an underground reactor shaft.

8. Plant Layout Arrangement

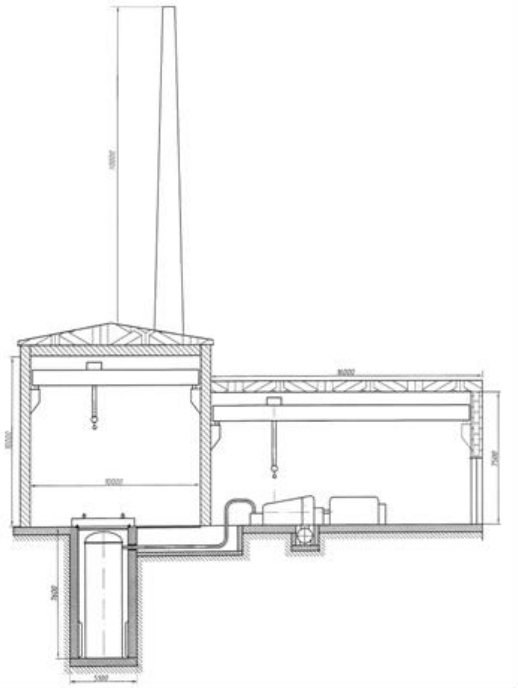


Figure 3: Example layout of a reactor building with an adjacent secondary island

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.

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.

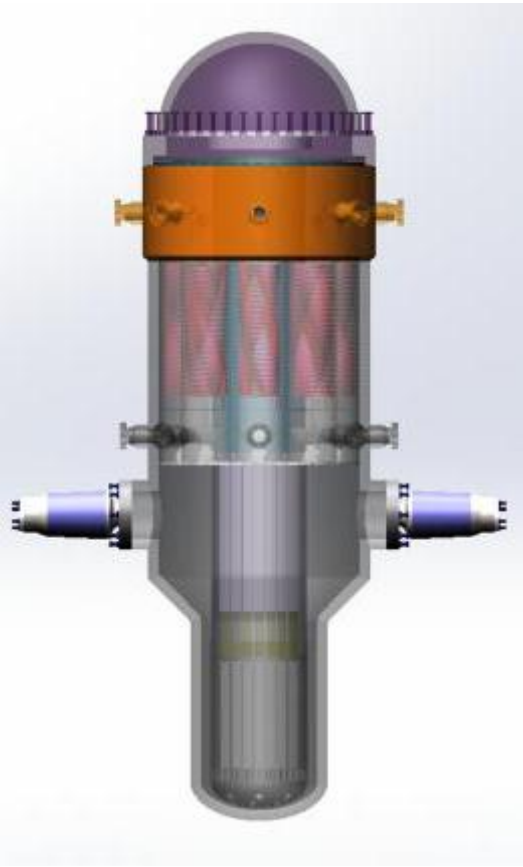
13. Development Milestones

Year	Milestone	Status
2015	Initial concept developed; patent priority date secured	**Completed**
2019	Concept submitted for peer review	**Completed**
2021	Conceptual design developed; preliminary feasibility study for a FOAK unit completed	**Completed**
2022	Conduction of feasibility study for FOAK unit and initiation of licensing procedures with local regulator and basic design	**Completed**
2024	Development of basic design and PSAR, conduction of site survey	**Ongoing**
2026	Launch of FOAK plant project, development of detailed design and extended SAR	**Planned**
2027-2028	Pilot unit installation and launch	**Planned**

Shanghai Weilan NuEnergy Technology, CHINA
WL-100



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Weilan NuEnergy, CHINA*
Reactor type	*Integral PWR*
Coolant/moderator	*Water*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*300/100*
NSSS operating pressure (primary/secondary), MPa	*15.5/4.5*
Core inlet/outlet coolant temperature (°C)	*291/320*
Primary circulation	*Forced with two RCPs*
Fuel type/assembly array	*UO2 fuel / 17x17 square lattice*
Number of fuel assemblies in the core	*37*
Fuel enrichment (%)	*Less than 5 wt. % U-235*
Core discharge burnup (GWd/ton)	*Less than 52*
Refuelling cycle (months)	*18~24*
Fuel cycle requirements/approach	*Open*
Reactivity control mechanism	*Rods, soluble boron*
Approach to safety systems	*Full passive*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*3,400*
RV height/diameter (m)	*14/4.4*
RV weight (metric ton)	*280 T(net)*

1. Introduction

The WL-100 is a 300 MWth small modular pressurized water reactor (SMR) developed by Shanghai Weilan NuEnergy Technology Co., Ltd., China. As a domestic private science and technology innovation enterprise, the company's mission is to develop advanced nuclear energy systems to provide green, low-carbon, and clean energy worldwide.

The developers of the WL-100 have fully leveraged proven technologies and mature domestic supply chains while incorporating selected advanced features. Key design characteristics include:

1. Integrated Reactor Coolant System: All major components are integrated within the reactor pressure vessel (RPV).
2. Passive Safety Systems: System actuation does not rely on external power, using gravity and thermal-hydraulic density differences to establish natural circulation, thereby maintaining long-term core cooling during accident conditions.
3. Modular Design: Pre-fabrication and assembly of major equipment components are performed at the factory, thereby shortening construction schedules and reducing capital costs.
4. Broad Site Suitability: Designed with a Safe Shutdown Earthquake (SSE) of 0.3g to cover most candidate site conditions. The core damage frequency is < 1E-7 per reactor-year, and the large release frequency is < 1E-8 per

reactor-year, practically eliminating the risk of severe accidents, significantly reducing the Emergency Planning Zone (EPZ), and enabling close proximity to end-users.

5. Multi-Application Scenarios: Applicable for district heating, centralized power generation, seawater desalination, and energy storage, as well as cogeneration to provide diverse solutions for users.

2. Target Application

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

3. Design Philosophy

The core design principles of the WL-100 are as follows:

1. High Safety Standards: Fully compliant with the latest Chinese nuclear safety regulations, IAEA safety standards, and international utility requirements—including the Utility Requirements Document (URD) and European Utility Requirements (EUR)—to achieve the highest global safety performance.
2. Proven & Advanced Technologies: Fully leverages proven technologies while incorporating global advanced SMR concepts, operating and construction feedback from existing pressurized water reactors (PWRs), and recent advancements in nuclear R&D.
3. Enhanced Economic Competitiveness: Maximizes economic performance through system simplification, modular construction, and extended refueling cycles without compromising safety, ensuring the commercial competitiveness of the WL-100.
4. High Deployment Flexibility: Offers high adaptability with configurable single-unit or multi-unit station layouts tailored to different domestic and international customer needs. The unit can be configured with tailored Conventional Island (CI) systems to support centralized power generation, district heating, combined heat and power (CHP), and other multi-purpose applications.

4. Main Design Features

(a) Nuclear Steam Supply System

The WL-100 Nuclear Steam Supply System (NSSS) adopts an integrated system configuration, in which eight helical coil steam generators, one pressurizer, and two canned-motor reactor coolant pumps are built into the reactor pressure vessel. This design eliminates the main piping and external loops. The reactor coolant circulates through internal flow paths formed by the reactor internals, steam generators, reactor coolant pumps, and pressure vessel to carry away the heat generated by the core.

(b) Reactor Core

The WL-100 reactor core utilizes 37 square-array 17×17 fuel assemblies with an active fuel length of 220 cm. Each fuel assembly contains 264 fuel rods, one central instrumentation tube for in-core neutron flux measurement, and 24 guide thimbles in the remaining positions.

A total of 20 control rod assemblies are uniformly arranged across the core, consisting of 12 control rod groups (for power and temperature control) and 8 shutdown rod groups. For the first cycle, the core fuel assemblies are arranged in a three-region pattern based on three different enrichment levels, the enrichment of U 235 in fresh fuel assemblies for equilibrium cycle refueling is less than 5%.

Fixed Self-Powered Neutron Detectors (SPNDs) are deployed within the core to continuously measure neutron flux, displaying key parameters such as 3D core power distribution, linear power density, and DNBR in real time.

(c) Reactivity Control

The core reactivity is controlled using soluble boron dissolved in the coolant, 20 rod cluster control assemblies (RCCAs), and solid burnable poison rods.

(d) Reactor Vessel (RV) and Internals

The reactor pressure vessel is a variable-diameter, long cylindrical pressurized vessel. To minimize the number of welds, the core section shell is manufactured using an integral forging process. The reactor pressure vessel is composed of the top cover assembly, the vessel body assembly and the fastening sealing components, etc.

Reactor internals are arranged inside the reactor pressure vessel to support and hold down the fuel assemblies; together with the reactor vessel and the housed-in steam generators , these internals form the coolant flow paths.

(e) Reactor Coolant System and Steam Generator

The WL-100 is equipped with eight helical coil once-through steam generators (OTSGs). The heat transfer tubes are arranged in a helical pattern. The OTSG is mainly composed of the upper steam head, main feed water pipe, steam connection pipe, tube bundle assembly, heat transfer tube support, central support column, tube plate, and cylinder structure, etc.The eight OTSGs are evenly arranged in the annular space between the reactor pressure vessel (RPV) and the core barrel inside the reactor. Their bottom ends are positioned and constrained by the hole locations on the support ring of the pressure vessel shell section, while their top ends are welded to the steam chamber of the pressure vessel via the steam outlet nozzles.

(f) Pressuriser

The WL-100 pressurizer is built into the upper part of the reactor pressure vessel (RPV) top head, equipped with a spray pipe at the top and electric heaters at the bottom. It maintains the pressure of the reactor pressure vessel within a controllable range through electric heating and spraying.

A safety valve discharge line is installed on the pressurizer section of the RPV top head to prevent over pressurization of the reactor coolant system (RCS) via the safety valve. Additionally, relief valves are provided on the pressurizer section of the top head to depressurize the reactor coolant system under accident conditions as required by the safety systems.

(g) Primary pumps

The WL-100 features two canned motor pumps symmetrically mounted on both sides of the pressure vessel .

Each canned motor pump consists of hydraulic components, a thermal insulation flange, and a canned motor. This structural configuration eliminates large breaks in the main coolant piping. Furthermore, the shaft-seal-less design eliminates the possibility of shaft seal failure.

(h) Distinguishing features

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The engineered safety features of the WL-100 mainly comprise the Emergency Cooling System (ECS), the Automatic Depressurization System (ADS), and the Passive Containment Cooling System (PCCS). The ECS is used for core heat removal under accident conditions (including both LOCA and non-LOCA events). The ADS is designed for reactor coolant system depressurization under LOCA conditions to enable water injection into the core from the in-containment refueling water storage tank. The PCCS is utilized for containment temperature and pressure reduction during SLB accidents and the early stages of LOCA events. Designed with fully passive safety systems, WL-100 allows these safety features to perform their designated safety functions under various accident conditions without relying on external power sources, such as AC power.

The safety system is significantly simpler than those of conventional pressurized water reactors (PWRs). Due to its integrated design and the substantial coolant inventory above the core within the pressure vessel, the high-pressure safety injection system (HPIS) and accumulators (ACCs) are not necessary. Furthermore, because the ECS and PCCS employ air-cooling methods using the ultimate heat sink of the atmosphere, they achieve an indefinite cooling capability under accident conditions.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

The WL-100 utilizes a fully passive system design and incorporates no active safety systems; thus, under Station Blackout (SBO) accident conditions, the loss of power exerts no impact on the operation of the safety systems. The ECS and ADS can achieve reactor pressure vessel coolant inventory maintenance and core heat removal under LOCA conditions.

(m) Safety Approach and Configuration to Manage DEC

The WL-100 is equipped with an Emergency Boration Injection System to cope with Anticipated Transient Without Scram accidents. In addition, the engineered safety features of the WL-100 can address various other Design Extension Conditions, such as a Steam Generator Tube Rupture combined with a Steam Line Break, or a single-line Direct Vessel Injection pipe break coupled with the failure of one train of the Emergency Cooling System.

(n) Containment System

The WL-100 containment system consists of the containment vessel (steel), Passive Containment Cooling System(PCCS), and Containment Isolation System, among others. The PCCS utilizes natural air convection for cooling; to enhance heat transfer, an outer shell is added outside the containment vessel to form an annular flow path, which acts as a stack effect.

(o) Spent Fuel Cooling Safety Approach / System

Spent fuels unloaded from the reactor core are stored in the spent fuel pool, where an independent cooling system cools the fuel, removes decay heat and maintains it in a sub-critical state.

6. Plant Safety and Operational Performances

The design core damage frequency of the WL-100 is < 1E-7 per reactor-year, and the large release frequency is < 1E-8 per reactor-year, meeting SMR safety goals. The unit operates in load-following mode within the 30% - 100% rated power range. The availability factor is larger than 90% in the design service life. The integrated reactor design dampens transient fluctuations, while the negative moderator temperature coefficient feedback enhances the operational stability of the core.

7. Instrumentation and Control Systems

The WL-100 instrumentation and control (I&C) system is designed in accordance with the single failure criterion and the principle of defense-in-depth, meeting regulatory requirements for independence, redundancy, defense-in-depth, and diversity. The I&C system employs domestically developed digital control system (DCS) and reactor protection system (use FPGA) based on a 2-out-of-4 logic, virtual diverse actuation system(DAS) is used to mitigate risks of protection function failure due to common-mode faults in the digital reactor protection system. Advanced AI technologies are integrated to support nuclear power plant operations and maintenance diagnostics. The core instrumentation system utilizes Rhodium-based Self-Powered Neutron Detector and thermocouples(Class 1E) to ensure reliable monitoring during normal operation and post-accident. The main control room (MCR), designed based on human factors engineering, minimizes the risk of human error. The WL-100 I&C system comprises process measurement and control systems, reactor control systems, core instrumentation systems, rod control and position monitoring systems, loose parts and vibration monitoring system, seismic monitoring systems, radiation monitoring systems, etc.

8. Plant Layout Arrangement

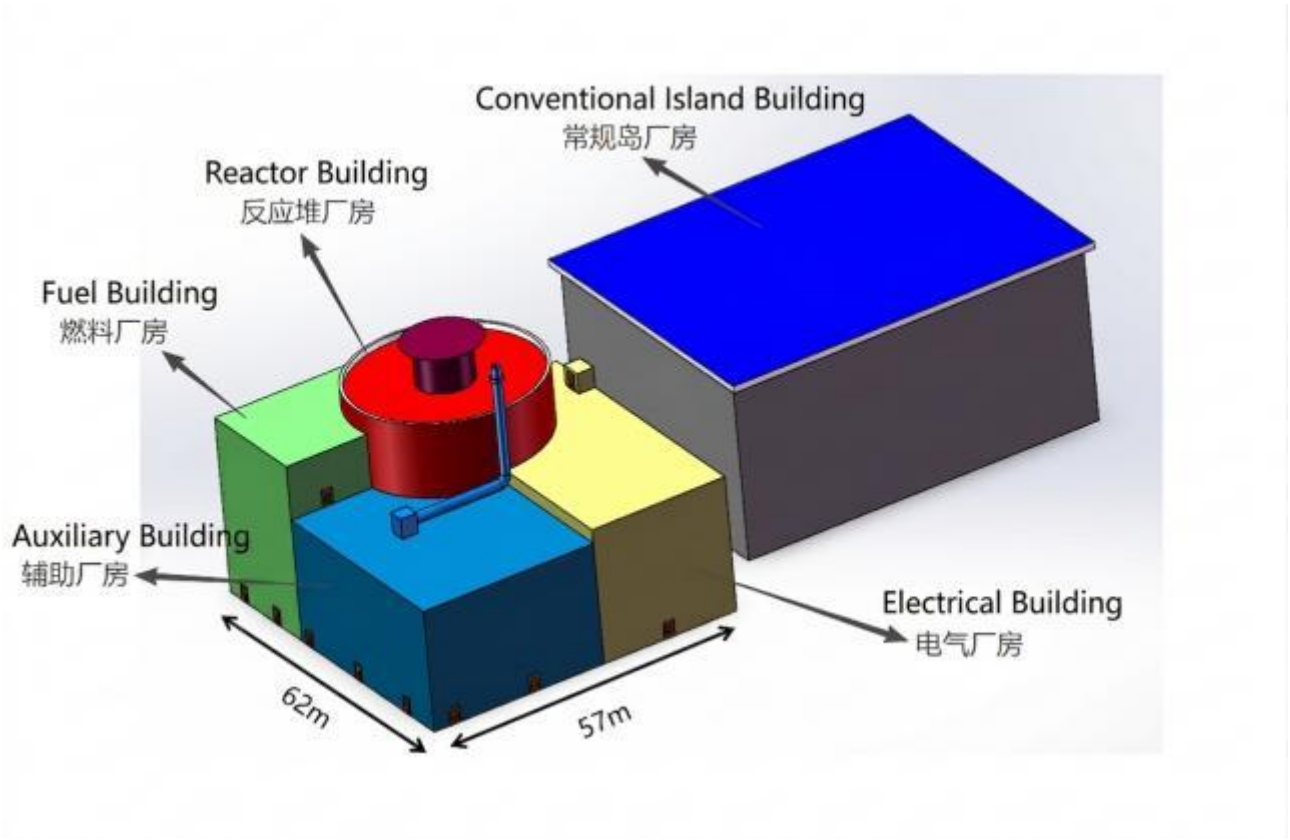


Figure 1. WL-100 Plant Layout Arrangement

The WL-100 currently adopts a single-unit layout, with simultaneous consideration of a dual-unit layout, comprising the Nuclear Island (NI) building and the Conventional Island (CI) building.

The Nuclear Island features a compact arrangement, the Reactor Building is surrounded by the Fuel Building, Electrical Building and Nuclear Auxiliary Building. Occupying a surface footprint of approximately 3,400 m², it incorporates a semi-embedded layout and a double-containment structure, designed for comprehensive modular design, manufacturing, construction, and installation.

9. Testing Conducted for Design Verification and Validation

During the development of the WL-100, corresponding verification test programs have been scheduled for the integrated reactor, key primary components, and safety systems.

10. Design and Licensing Status

The WL-100 is currently in the conceptual design phase, with completion of the basic design scheduled for the end of 2029.

Engagement with regulatory authorities and potential customers has been initiated, and project licensing applications will be pursued in an orderly manner according to schedule.

11. Fuel Cycle Approach

The U-235 enrichment for the WL-100 fuel is under 5%, with a refueling cycle length of 18 to 24 months. Spent fuel management aligns with standard pressurized water reactors (PWRs), utilizing on-site wet storage followed by reprocessing and recycling.

12. Waste Management and Disposal Plan

The technologies for the treatment of liquid, gaseous and solid radioactive wastes generated by the WL-100 are similar to those employed in conventional pressurized water reactor (PWR) plants. Proven and effective treatment processes will be adopted, together with established waste management and disposal technologies. Spent fuel will initially be stored in the spent fuel pool and subsequently transferred to a dry storage facility.

13. Development Milestones

Year	Milestone	Status
2025	Technology concept finalized	**Completed**
2025-26	Pre-conceptual design phase and technology validation	**Completed**
2026	Completion of conceptual design	**Ongoing**
2029	Basic design package issued	**Planned**
2031	Detailed design completed	**Panned**
2031	First concrete poured	**Planned**

Shanghai Universal Nuclear Energy Technology Co., Ltd., China
YL-100



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*UNE, CHINA*
Reactor type	*Integral PWR*
Coolant/moderator	*Light water*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*240/80*
NSSS operating pressure (primary/secondary), MPa	*14.0/3.4*
Core inlet/outlet coolant temperature (°C)	*250/315*
Primary circulation	*Natural circulation*
Fuel type/assembly array	*UO2 fuel / 17x17 square*
Number of fuel assemblies in the core	*37*
Fuel enrichment (%)	*<4.95.%*
Core discharge burnup (GWd/ton)	*45*
Refuelling cycle (months)	*24*
Fuel cycle requirements/approach	-
Reactivity control mechanism	*Control rods*
Approach to safety systems	*Passive*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*20,000*
RV height/diameter (m)	*18.18/<2.80*
RV weight (metric ton)	*220*

1. Introduction

The YL-100 Small Modular Reactor (SMR) is being developed as a market-oriented, economics informed, low-carbon nuclear power generation solution. Building upon proven and mature technologies, the design incorporates optimization and simplification to achieve industry-leading safety performance and strong market competitiveness. Through optimized plant layout and extensive application of modular construction, standardization, and commercial off-the-shelf approaches, the YL-100 enables rapid, reliable, and replicable project delivery.

2. Target Application

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

The YL-100 SMR is designed to provide baseload electricity generation, district and industrial heat supply, or combined heat and power (CHP) services. The plant is suitable for deployment at either coastal or inland sites.

3. Design Philosophy

The YL-100 SMR adopts proven advanced small pressurized water reactor (PWR) technology with a modular manufacturing approach. By reducing upfront capital investment, the design aims to lower the levelized cost of electricity/processing heat and enhance overall project economics. The unit size has been specifically selected to support full lifecycle modularization and standardization.

4. Main Design Features

(a) Nuclear Steam Supply System

The YL-100 SMR employs an integral, natural-circulation reactor design. The reactor core, steam generators, and pressurizer are all housed within a single reactor pressure vessel (RPV). No reactor coolant pumps are required, and the primary coolant system operates without soluble boron, resulting in a simplified system configuration.

A compact steel containment vessel surrounds the reactor pressure vessel and, together with the fuel cladding and reactor pressure vessel, forms three successive barriers against radioactive release

(b) Reactor Core

Nuclear fuel pellets are made of standard uranium dioxide (UO_2) with a maximum enrichment of 4.95% and zirconium alloy cladding. Fuel assemblies adopt a 17×17 lattice arrangement. The core contains 37 fuel assemblies in total, with an active fuel length of 2.15 m. Gadolinium oxide solid burnable poisons and control rod assemblies are used for core reactivity compensation. The flattening of the core power distribution during operation is achieved through adjustment of control rod positions in accordance with the burnable poison and fuel enrichment distributions.

(c) Reactivity Control

During normal operation, reactor reactivity is controlled through the movement of 21 control rod assemblies. The reactor exhibits a negative moderator temperature coefficient, providing a favorable inherent safety characteristic. Soluble boron is not utilized for routine reactivity control in the reactor coolant system. This approach simplifies plant systems, eliminates operational risks associated with boron chemistry management, and minimizes environmental impacts related to boron discharge. An emergency boron injection system is provided as an independent and diverse second shutdown system.

(d) Reactor Vessel (RV) and Internals

The reactor pressure vessel (RPV) consists of an upper shell, a lower shell, and an ellipsoidal bottom head. The upper shell incorporates four steam generator feedwater inlet nozzles and four steam outlet nozzles. The vessel section corresponding to the active core region is free of circumferential welds. Twenty-one control rod drive mechanisms (CRDMs) and two Reactor Safety Valves (RSVs) are installed on the reactor vessel head.

As part of the Emergency Core Cooling System (ECCS), two Reactor Vent Valves (RVVs) are located on the vessel head and two Reactor Recirculation Valves (RRVs) are installed on the vessel shell. These valves establish a passive flow path that allows containment condensate to return to the reactor core under accident conditions.

Lower reactor internals include the core barrel (to fix fuel assemblies and guide coolant upward from the core inlet through fuel assemblies) and a radial metallic neutron reflector.

The upper reactor internals collect coolant exiting the core and distribute it to the steam generator inlet region. These structures include upper control rod guide assemblies, in-core instrumentation guide tubes, and associated support components.

(e) Reactor Coolant System and Steam Generator

A helical coil steam generator is selected and installed outside the upper part of the core barrel. It features mature technology and good engineering feasibility. The primary coolant flows outside the helical coils, while secondary-side steam flows inside the tubes.

(f) Pressuriser

The pressurizer is integrated into the upper head region of the reactor pressure vessel. Primary system pressure is controlled by electric heaters located at the bottom of the pressurizer and spray nozzles located at the top. A steam-water equilibrium is maintained within the pressurizer to provide the required system pressure.

(g) Primary pumps

N/A

(h) Distinguishing features

Integral RPV, natural circulation, and simple systems, without soluble boron in primary loop

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The YL-100 SMR is designed using a safety-goal-driven philosophy and incorporates strong inherent and passive safety characteristics. The plant implements the defense-in-depth principle through multiple levels of accident prevention and mitigation. Independent, diverse passive safety systems are provided with redundant trains for key safety functions. Passive safety systems are capable of maintaining required safety functions autonomously for at least 72 hours without operator intervention or external power supplies.

(j) Decay Heat Removal System

The YL-100 SMR is equipped with two independent passive Decay Heat Removal Systems (DHRs), each capable of removing 100% of the reactor core decay heat load. The DHR is actuated following a failure of the secondary-side heat removal function. During operation, the system receives superheated steam directly from the steam generators and transfers its heat to a water pool that serves as the ultimate heat sink. The steam is condensed and returned through a dedicated feedwater flow path to the steam generator inlet. In this manner, the DHR establishes a closed secondary-side heat removal loop with the steam generators, providing long-term passive decay heat removal without reliance on external power or operator action.

(k) Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) provides core cooling during and after anticipated operational occurrences and design-basis accidents. It is a critical safety system for the plant to perform safety functions related to loss-of-coolant accidents, and also constitutes part of the reactor coolant system and containment pressure boundary. Cooperating with containment heat removal functions, the emergency core cooling system removes core decay heat in the event of loss-of-coolant accidents.

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

The RPV and other key equipment are enclosed within a compact steel containment designed to contain radioactive releases under fault and accident conditions. The RPV incorporates molten core material retention capability, allowing it to retain or even eliminate hypothetical molten core material under severe accident conditions. The containment free volume is deliberately limited so that it can accommodate only a portion of the RPV coolant inventory; the remaining inventory is sufficient to keep the entire reactor core covered. This design precludes exposure of the fuel rod cladding to air.

(o) Spent Fuel Cooling Safety Approach / System

Soluble boron is not adopted in the safety system for spent fuel cooling. Reactivity control is achieved by geometric spacing of storage racks and solid neutron absorber grid plates inside the fuel assembly pool. Under accident conditions, passive cooling systems form multi-layer defense-in-depth to reliably remove decay heat.

6. Plant Safety and Operational Performances

Plant transients and accident scenarios have been evaluated using industry-validated analytical methods and computer codes, demonstrating adequate safety margins across all levels of defense-in-depth.

Probabilistic Safety Assessment (PSA) results indicate:

- A Core Damage Frequency (CDF) of less than 1×10^{-7} per reactor-year.
- No single initiating event or accident category makes a disproportionately large contribution to overall plant risk, resulting in a balanced risk profile.

Advanced automation technologies, including applications of artificial intelligence, are being evaluated to support reduced-staff and potentially autonomous control room operation.

The YL-100 SMR is designed to comply with Chinese grid requirements and the needs of utility customers in most international markets. The plant supports both stable baseload generation and load-following operation and is particularly well suited for supplying process heat and steam to industrial parks and chemical complexes.

The planned duration of each refueling and maintenance outage is 15 days, supporting a projected plant capacity factor of at least 90%.

7. Instrumentation and Control Systems

The YL-100 SMR nuclear power plant incorporates multiple instrumentation and control (I&C) systems to provide plant monitoring, control, and protection functions. The Reactor Protection System (RPS) automatically initiates a reactor trip upon detection of fault conditions. Built-in priority logic evaluates multiple process and safety signals to determine the need for trip actuation. An independent and diverse second shutdown system is provided to ensure reliable reactor shutdown and adequate subcriticality in the unlikely event that control rod insertion is unavailable.

The reactor is equipped with fixed self-powered neutron detectors and temperature sensors for continuous core monitoring.

8. Plant Layout Arrangement

Benefiting from high safety and reliability, the YL-100 SMR can be constructed at coastal or inland sites. The plant features a compact footprint and offers substantial flexibility in site selection.

9. Testing Conducted for Design Verification and Validation

The YL-100 SMR adopts optimized, proven technologies, and its design has been verified by relevant tests.

10. Design and Licensing Status

Not in a licensing state yet.

11. Fuel Cycle Approach

The YL-100 SMR has a designed refueling cycle of 24 months with batch refueling. An equilibrium cycle core can be achieved after approximately 3 or 4 fuel cycles. The projected discharge burnup of spent fuel assemblies reaches 45 GWd/tU.

12. Waste Management and Disposal Plan

The waste treatment system of the YL-100 SMR employs mature technologies and best processes available. Drawing on industry experience and good practices, especially through the design and operation of a boron-free core, the generation of radioactive and non-radioactive waste as well as emissions will be greatly reduced, environmental discharges minimized, and the waste treatment system simplified.

13. Development Milestones

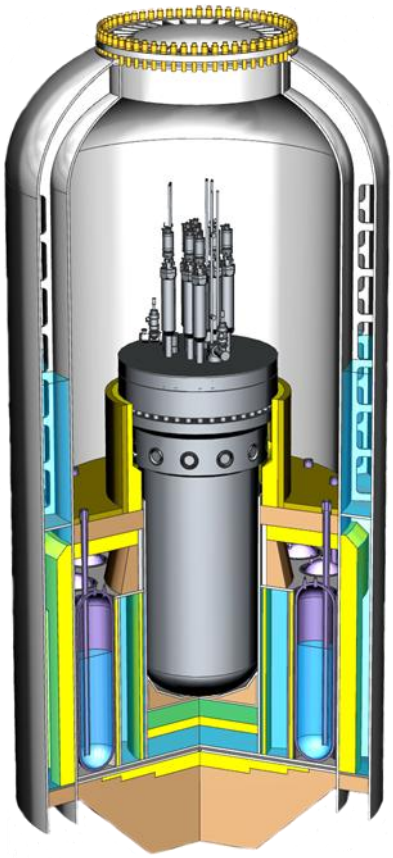
Year	Milestone	Status
2025	Conceptual design	Completed
2026-27	Basic design. Submission of safety analysis report to IAEA for TSR (Technical Safety Review)	Ongoing
2028	Completion of IAEA TSR. Completion of detailed design	Planned
2029-2030	Licensing of construction. Start of construction of the first demonstration reactor	Planned
2031-32	Commercial operation of the first demonstration reactor	Panned

MARINE-BASED SMALL MODULAR REACTORS

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



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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*
Design status	*Detailed design*
Thermal/electrical capacity, MW(t)/MW(e)	*38 / 6 – 9*
NSSS operating pressure (primary/secondary), MPa	*16.2*
Core inlet/outlet coolant temperature (°C)	*250 / 325*
Primary circulation	*Natural circulation*
Fuel type/assembly array	*UO2 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*
Fuel cycle requirements/approach	*N/A*
Reactivity control mechanism	*Control rod driving mechanism*
Approach to safety systems	*Passive*
Seismic design (SSE)	*7 per Richter scale (basic design)*
Design life (years)	*40*
Plant footprint (m^2)	*20 000 (basic design)*
RV height/diameter (m)	*6 / 2.4*
RV weight (metric ton)	*N/A*

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

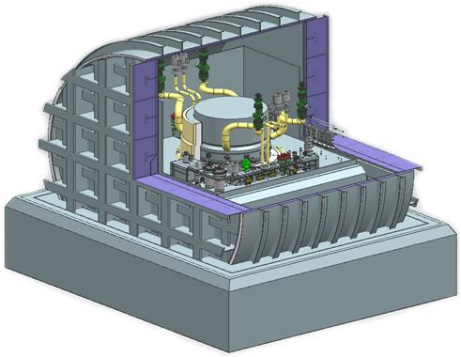
☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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.

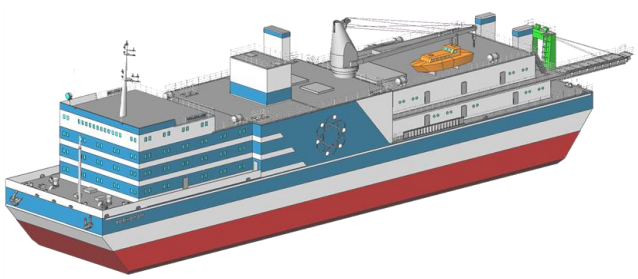
4. Main Design Features

(a) Nuclear Steam Supply System

N/A

(b) Reactor Core

The core comprises 121 hexagonal fuel assemblies (FA) of cassette type with active part height of 900 mm, similar to the FAs in KLT-40S. Cermet fuel is used with less than 20% enriched 235U. Special stainless steel is used as fuel cladding.



(c) Reactivity Control

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

(d) Reactor Vessel (RV) and Internals

The RPV is a welded cylindrical 'container' with an elliptical bottom. At the top of the vessel there are pipes for feedwater supply and superheated steam removal, as well as those for the connection of the primary circuit systems and the auxiliary process systems. The RPV head consists of a load-bearing slab, a shell attached to this slab and sealed by a weld, and a top slab welded to the shell. The cavity between the top slab and the load-bearing slab is filled with serpentine which acts as a biological shielding, and the heat insulation is located at the top. The posts of the CPS drives and thermal converters, etc. are welded to the load-bearing slab and penetrate through the cover. Points of penetration through the top slab are sealed. Fuel assemblies are located in the in-vessel shaft. The protective tubes and devices provide the necessary coolant flow rate distribution between the fuel assemblies and an arrangement of connectors for joining the absorber elements of fuel assemblies into CPS control rods and connecting the CPS control rods to CPS drives.

(e) Reactor Coolant System and Steam Generator

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.

(f) Pressuriser

N/A

(g) Primary pumps

N/A

(h) Distinguishing features

Natural circulation in the primary circuit.

5. Safety Features

(i) 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

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

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.

(m) Safety Approach and Configuration to Manage DEC

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

(n) 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.

(o) Spent Fuel Cooling Safety Approach / System

N/A

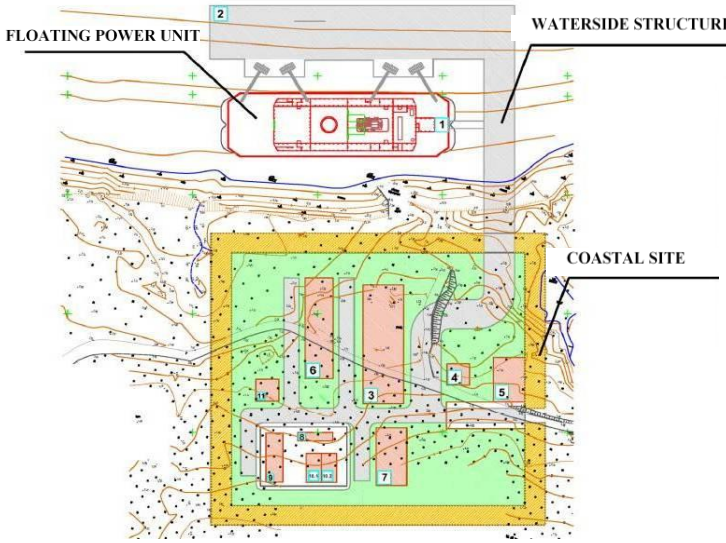
6. Plant Safety and Operational Performances

N/A

7. Instrumentation and Control Systems

N/A

8. 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

9. Testing Conducted for Design Verification and Validation

N/A

10. Design and Licensing Status

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

11. Fuel Cycle Approach

N/A

12. Waste Management and Disposal Plan

N/A

13. Development Milestones

Year	Milestone	Status
2006	Feasibility study developed for construction of the floating NPP with ABV-6M for the Far North (settlement Tiksi, settlement Ust-Kamchatsk)	Completed
2007	Feasibility study developed for construction of the floating NPP with ABV-6M for Kazakhstan (City of Kurchatov)	Completed
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)	Ongoing

CNNC, China
ACP100S



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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*
Design status	*Basic Design*
Thermal/electrical capacity, MW(t)/MW(e)	*385 / 125*
NSSS operating pressure (primary/secondary), MPa	*15 / 4.6*
Core inlet/outlet coolant temperature (°C)	*286.5 / 319.5*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO2 / 17×17 square pitch arrangement*
Number of fuel assemblies in the core	*57*
Fuel enrichment (%)	*< 4.95*
Core discharge burnup (GWd/ton)	*< 52*
Refuelling cycle (months)	*24*
Fuel cycle requirements/approach	*Shore refuelling, temporally stored in the spent fuel pool*
Reactivity control mechanism	*Control rod drive mechanism (CRDM), Gd2O3 solid burnable poison and soluble boron acid*
Approach to safety systems	*Passive + Active*
Seismic design (SSE)	*Not applicable*
Design life (years)	*40*
Plant footprint (m^2)	*8 000 (Floating NPP)*
RV height/diameter (m)	*10 / 3.35*
RV weight (metric ton)	*300*

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

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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 235U 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 Vessel (RV) 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 and 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.

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) Pressuriser

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

(g) 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.

(h) Distinguishing features

5. Safety Features

(i) 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.

(j) 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.

(k) 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.

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) 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.

(o) 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.

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 Systems

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.

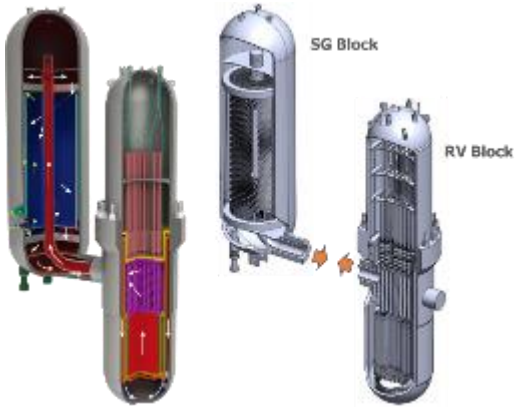
13.Development Milestones

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

KEPCO E&C, Republic of Korea
BANDI



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*KEPCO E&C, Republic of Korea*
Reactor type	*Semi-integral PWR*
Coolant/moderator	*Light water / light water*
Design status	*Basic Design*
Thermal/electrical capacity, MW(t)/MW(e)	*200 / 60*
NSSS operating pressure (primary/secondary), MPa	*15 / 5.5*
Core inlet/outlet coolant temperature (°C)	*293 / 322*
Primary circulation	*Forced circulation (2 canned-motor pumps)*
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*
Fuel cycle requirements/approach	*Optional (single batch or multi-batch refuelling)*
Reactivity control mechanism	*Control rods via CRDM, MTC, and BAs*
Approach to safety systems	*Passive (PECCS, PRHRS, and PCCS)*
Seismic design (SSE)	*N/A*
Design life (years)	*60*
Plant footprint (m^2)	*TBD*
RV height/diameter (m)	*13.6 / 2.8 – 3.4*
RV weight (metric ton)	*110*

1. Introduction

BANDI is a Small Modular Reactor (SMR) developed by KEPCO E&C. Named after a tiny and cute Korean firefly, symbolizing its image of emitting bright yellowish green light in a deep forest on a summer night, BANDI is designed to provide a clean and sustainable energy solution to remote areas. The reactor features a semi-integral Pressurized Water Reactor (PWR) layout, emphasizing enhanced safety and simplified design. KEPCO E&C is now working closely with a Korean global leading shipbuilding company on engineering and business planning.

2. Target Application

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

BANDI's primary business model focuses on floating nuclear power plants designed to replace aging fossil fuel power plants. Its secondary business model involves nuclear-powered merchant ships. Both models are anticipated to play a crucial role in achieving the International Maritime Organization's (IMO) net-zero greenhouse gas emissions target by 2050.

3. Design Philosophy

The BANDI design is carefully designed, based on KEPCO E&C's broad technology and over four decades of proven experience in conventional nuclear power engineering. Key nuclear safety functions, such as reactor shutdown, decay heat removal and radiation protection, are greatly improved through the use of inherent and passive safety features. In addition, the design includes advanced technologies including advanced nuclear fuel, simplified and efficient control system, automatic load following, and autonomous functions.

4. Main Design Features

(a) Nuclear Steam Supply System

The reactor system features a single-loop, block-type (or semi-integral) configuration, directly connecting the Reactor Vessel (RV) and Steam Generator (SG) blocks without large pipes. This design, by eliminating large-diameter piping, significantly enhances structural integrity and safety. Its nozzle-to-nozzle connection adheres to ASME BPVC Sec. III, Class 1 NB-3300 standards. Compared to fully integral systems, this block-type design offers easier maintenance and improved scalability.

The soluble boron-free design eliminates concerns associated with corrosive boron, a common issue in conventional PWR plants. This significantly simplifies the Chemical and Volume Control System (CVCS), reduces the plant's footprint, mitigates corrosive environments, and minimizes liquid radioactive waste.

(b) Reactor Core

The reactor core utilizes conventional UO_2 ceramic fuel with an enrichment of up to 4.95 % (weight percent), arranged in a 17×17 array. The core consists of 52 fuel assemblies with an active core height of 2 meters, designed to generate 200 MW of thermal power. To manage reactivity throughout the fuel cycle, the core incorporates burnable absorbers, specifically Pyrex (B_2O_3-based) rods, which replace 24 fuel rods per assembly. The core's configuration allows for a 4.79-year operation cycle at full power and the discharge burnup is designed to be 29.4 GWd/ton.

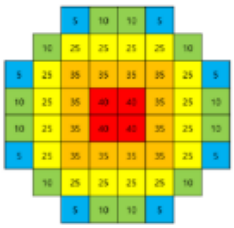


Figure 1. Reactor Core with Burnable Poison

(c) Reactivity Control

BANDI operates without soluble boron for its core reactivity control. This design increases the magnitude of the negative moderator temperature coefficient (MTC), thereby enhancing the inherent safety and stability of reactivity control. The reactor core incorporates 40 control rod assemblies, each containing 24 B4C control rods. These are categorized into two types: 18 for regulation and 22 for shutdown. The regulation rods are organized into four banks, and the shutdown rods into two banks.

The integration of burnable absorbers, specifically Pyrex (B2O3) rods, aids in managing excess reactivity throughout the operational life of the core. This overall configuration ensures sufficient negative reactivity for shutdown capability without relying on soluble boron and also supports the operational efficiency and safety of the reactor.

(d) Reactor Vessel (RV) and Internals

The BANDI reactor pressure vessel (RPV) and its internals constitute the reactor vessel (RV) block. The RPV design features a cylindrical structure with a hemi-spherical lower head and a removable top cap. The top-mounted in-core

instrumentation (TM-ICI) eliminates the need for penetration in the lower head, thereby minimizing the risk of vessel failure even during severe core meltdown accidents.

The control rod drive mechanism (CRDM) is installed inside the RPV, a configuration referred to as In-Vessel CRDM. This design eliminates the rod ejection reactivity accident from the design basis. The consequences due to the rod ejection accident are considered more severe for small, boron-free reactors than conventional large reactors, as the individual control rod's reactivity worth is significantly higher in the small reactor core.

The RPV is designed to be fabricated from SA-508, Grade 3, Class 1.

(e) Reactor Coolant System and Steam Generator

The reactor coolant system (RCS) comprises one RV block and one SG block, thereby making a single-loop configuration. The SG block houses one steam generator, one pressurizer, and two canned-motor Reactor Coolant Pumps. The once-through steam generator features a vertical arrangement with helically coiled heat transfer tubes; reactor coolant flows through the tube side, while secondary steam and feedwater flow through the shell side.

(f) Pressuriser

The pressurizer is integrated into the upper section of the SG block, managing system pressure via heaters and sprays. It is designed with sufficient capacity to control reactor coolant system volume across the full range of power levels. Safety Depressurization Valves (SDVs) are included on the top of the pressurizer to facilitate rapid depressurization during small break loss-of-coolant accidents (SBLOCAs).

(g) Primary pumps

The BANDI reactor employs two leak-tight canned-motor reactor coolant pumps (RCPs) located in the SG block's bottom plenum. Their sealed design minimizes maintenance requirements and enhances operational safety.

(h) Distinguishing features

The BANDI has several unique design features as presented above: block-type NSSS, soluble boron-free design, TM-ICI, IV-CRDM, etc. These designs have a common goal to achieve both inherent safety and compact design for a maritime application.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The BANDI reactor employs a Passive Emergency Core Cooling System (PECCS) comprising an Emergency Core Cooling Tank (ECCT), Safety Depressurization Valves (SDVs), and Safety Recirculation Valves (SRVs) to facilitate emergency coolant injection without reliance on active safety systems. By integrating passive safety mechanisms such as gravity-driven injection and natural circulation cooling, the design ensures reactor safety under all accident conditions.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

For Design Basis Conditions (DBC), the BANDI reactor employs a block-type design with a single heat transfer loop to minimize Loss of Coolant Accident (LOCA) risks. Key components include a robust Reactor Pressure Vessel (RPV) and a helical tube Steam Generator (SG), integrated with Safety Depressurization Valves (SDVs) and Safety Recirculation Valves (SRVs) for emergency core cooling. The design further incorporates redundant cooling and pressure control systems, alongside advanced instrumentation, to ensure early warning and effective response during extreme conditions.



Figure 2. Major Safety Features of BANDI

(m) Safety Approach and Configuration to Manage DEC

For Design Extension Conditions (DEC), the BANDI reactor utilizes passive safety systems, including the Passive Emergency Core Cooling System (PECCS) and the Containment Vessel Cooling Tank (CVCT), to manage severe accidents without reliance on external power.

(n) Containment System

The containment system comprises two main structures: the Containment Vessel (CV) and the Reactor Compartment. The BANDI CV is a steel cylindrical vessel designed to confine radioactive materials that could be released during both normal and accident conditions, including design basis accidents and severe accidents. The Reactor Compartment encloses the entire primary coolant system, providing protection against external hazards such as aircraft impact.

(o) Spent Fuel Cooling Safety Approach / System

The BANDI spent fuel cooling system features a pool designed to store spent fuel assemblies for up to 10 years, equipped with a dedicated heat removal system and chemistry control. This system ensures effective decay heat removal and accommodates both new and spent fuel during refuelling outages. Fuel assemblies are transferred to and from the pool via a fuel transfer system, with the refueling process completed in approximately 10 days.

6. Plant Safety and Operational Performances

The BANDI reactor prioritizes safety via passive systems for emergency cooling and decay heat removal. Operational efficiency is enhanced by a boron-free reactivity control design fully utilizing the enhanced feedback effect of the negative Moderator Temperature Coefficient (MTC), alongside a Reactor Trip Prevention System (RTPS) designed to prevent unintended shutdowns.

7. Instrumentation and Control Systems

The Instrumentation and Control (I&C) system is a fully digital architecture featuring a data communication network, designed with a simple and compact structure optimized for the BANDI reactor. To mitigate software Common Cause Failure (CCF), the I&C design incorporates a proprietary diversity strategy within the safety system.

8. Plant Layout Arrangement

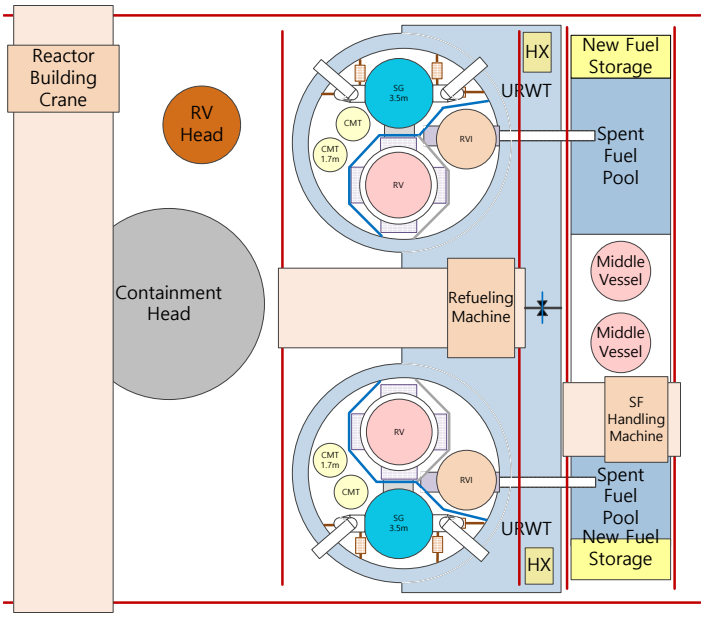


Figure 3. Plant View of Plant Layout

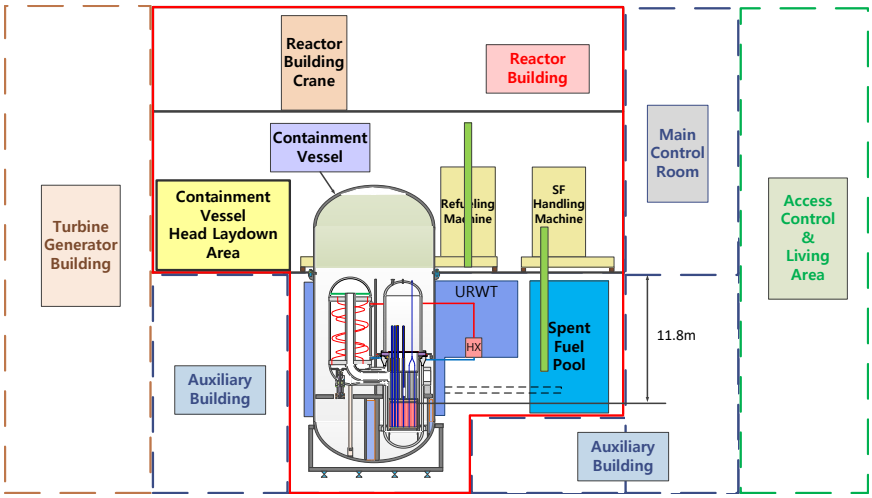


Figure 4. Section View of Plant Layout

The plant layout is designed for optimal safety and efficiency, featuring clear zoning for reactor operations, support systems, and emergency facilities. In the dual-unit configuration of the floating nuclear power plant, the design incorporates two separate Containment Vessels (CVs), each with individual reactor systems, to mitigate the impact of potential accidents. The auxiliary compartment safeguards safety-related systems against natural hazards through robust protective measures, while the turbine compartment houses the turbine generators, condensers, and heaters.

9. Testing Conducted for Design Verification and Validation

The reliability of the In-Vessel Control Rod Drive Mechanism (IV-CRDM) has been verified through full-scale testing. These tests were conducted under high-pressure and high-temperature conditions to simulate normal operating environments, ensuring the mechanism's operational performance and integrity.



Figure 5. In-Vessel CRDM Demonstration Test Facility

10. Design and Licensing Status

The project is currently in the basic design phase, integrating verified design features based on over 40 years of PWR design experience and recent innovations. Major design updates were implemented in 2023. Further refinements are currently underway in collaboration with Korea's leading shipbuilding company. The schedule for licensing reviews is not yet determined.

11. Fuel Cycle Approach

The BANDI reactor design supports a single batch refuelling every 4 to 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 to 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 the temporary on-site storage of radioactive waste prior to its transfer to long-term disposal facilities. Low- and Intermediate-Level Wastes (LILW) are safely contained, while High-Level Waste (HLW) is managed with enhanced safety measures. Spent fuel is stored on-site for up to 5 to 10 years before being shipped to a dedicated storage facility, with fuel cycle options varying based on owner preferences.

13. Development Milestones

Year	Milestone	Status
2012-2015	Preliminary and basis studies for SMRs	Completed
2013-2018	Government-supported R&D projects for key innovative element technologies of SMR	Completed
2016-2022	Nuclear Steam Supply System (NSSS) conceptual design for maritime SMRs	Completed
2023	Overall design change	Completed
2023-2026	NSSS, Balance of Plant (BOP), and floating system conceptual design for FNPPs	Completed
2026-2028	Basic design in collaboration with a shipbuilding company	Ongoing
2029-2034	Standard (detailed) design and licensing review	Planned
2035	Design certification, manufacturing, and construction	Planned
2039	Commissioning	Planned

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



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*JSC “Afrikantov OKBM”, Rosatom, Russian Federation*
Reactor type	*Block-type PWR*
Coolant/moderator	*Light water / light water*
Design status	*In Operation*
Thermal/electrical capacity, MW(t)/MW(e)	*150 / 35*
NSSS operating pressure (primary/secondary), MPa	*12.7 / 3.82*
Core inlet/outlet coolant temperature (°C)	*280 / 316*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO2 pellet in silumin matrix / hexagonal array*
Number of fuel assemblies in the core	*121*
Fuel enrichment (%)	*18.6*
Core discharge burnup (GWd/ton)	*45.4*
Refuelling cycle (months)	*30-36*
Fuel cycle requirements/approach	*SNF handling within the vessel*
Reactivity control mechanism	*Control rod driving mechanism*
Approach to safety systems	*Active (partially passive)*
Seismic design (SSE)	*9 point on the MSK scale*
Design life (years)	*40*
Plant footprint (m^2)	*4320 (Floating NPP)*
RV height/diameter (m)	*4.8 / 2.0*
RV weight (metric ton)	*N/A*

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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 Vessel (RV) 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 operating nuclear icebreakers «Yamal» and «50 Years of Victory».

(e) Reactor Coolant System and Steam Generator

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 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).

(f) Pressuriser

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. It is not an integral part of the reactor systems. Gas-type compensator is formed by one group and three groups of gas tanks.

(g) Primary pumps

The reactor is equipped with four sealed pumps of the primary circuit.

(h) Distinguishing features

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.

(i) Engineered Safety System Approach and Configuration

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

(j) 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.

(k) 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.

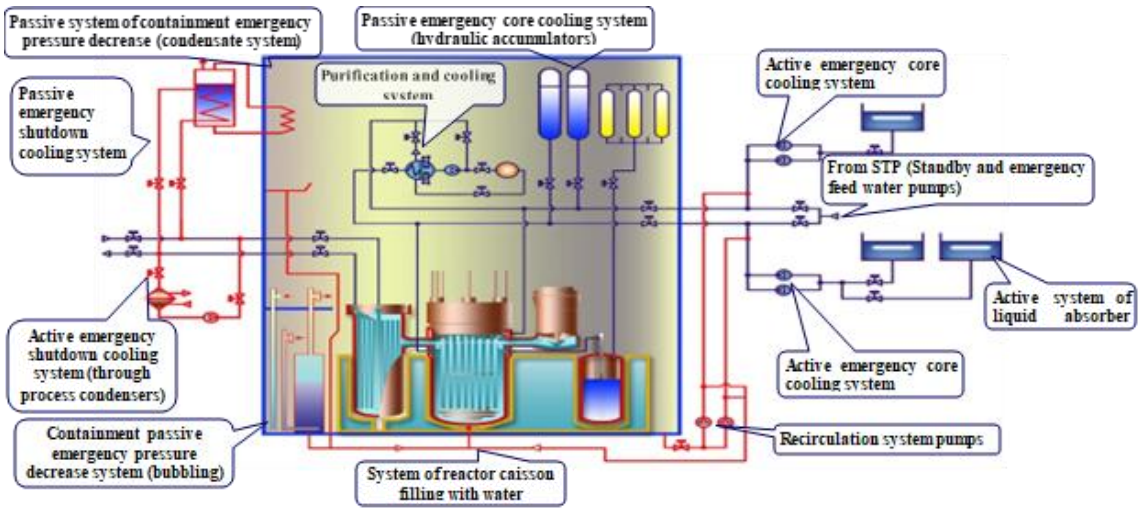


Figure 1. KLT-40S reactor safety systems

(l) Safety Approach and Configuration to Manage DBC

Active safety systems are used to manage design basis accidents. The automatic control system ensures the transfer of the reactor plant to a safe state in automatic mode without the participation of personnel. Safety systems are designed for all types of design basis and beyond design basis accidents. Each safety system has at least two channels, each channel is designed to perform the safety function in full. Efficiency and sufficiency of safety systems is confirmed by calculations and justified in the safety analysis report.

(m) Safety Approach and Configuration to Manage DEC

The following is used to control beyond design basis accidents:

- passive safety systems in case of failure of active systems or power supply;
- means of beyond design basis accident management in case of failure of safety systems;
- self-actuating devices in case of control system failure.

Switching the reactor plant to a safe state is carried out by any available means automatically or remotely. To ensure each safety function, there are active and passive safety systems, as well as special technical means. The design provides systems for retaining molten fuel in the reactor vessel in the event of a severe accident. Efficiency and sufficiency of safety systems and technical means is confirmed by calculations and justified in the safety analysis report. At the disposal of the personnel there is a guide for managing beyond design basis accidents, containing additional instructions for restoring safety functions in beyond design basis conditions. At the disposal of personnel there is an emergency communication channel with the RP designer for obtaining additional instructions and consultations on eliminating beyond design basis conditions.

(n) 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.

(o) Spent Fuel Cooling Safety Approach / System

The spent fuel storage facility is equipped with a dual-circuit cooling system that provides heat removal from spent fuel during normal operation and in case of violations of normal operation, including design basis and beyond design basis accidents.

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. Instrumentation and Control Systems

An automated control system is provided in the KLT-40S 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. Plant Layout Arrangement

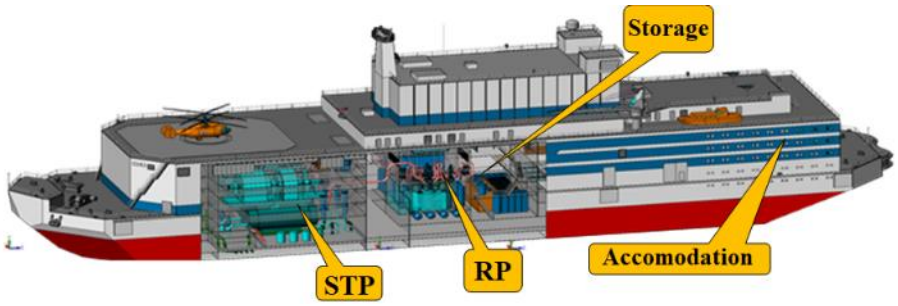


Figure 2. General cross-section view of FPU

The coastal 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.

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.

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.

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. Testing Conducted for Design Verification and Validation

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. KLT-type reactors are used in a series of nuclear-powered icebreakers "Taymyr" и "Vaygach" и nuclear lighter carrier "Sevmorput"

10. Design and Licensing Status

Project development is completed. The project is licensed by Rostechnadzor for operation in the town of Pevek in Chukotka region. The construction of Akademik Lomonosov was completed in 2017. The Akademik Lomonosov has started test operation in December 2019 and commercial in May 2020. Currently, the FPU is operated for its intended purpose.

11. Fuel Cycle Approach

The FPU is delivered to the site with fresh fuel in its reactors. During operation, spent fuel handling operations are performed within the vessel. Spot nuclear fuel post-irradiation handling and processing are performed in the Russian Federation.

12. Waste Management and Disposal Plan

The waste is stored within the FPU, 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.

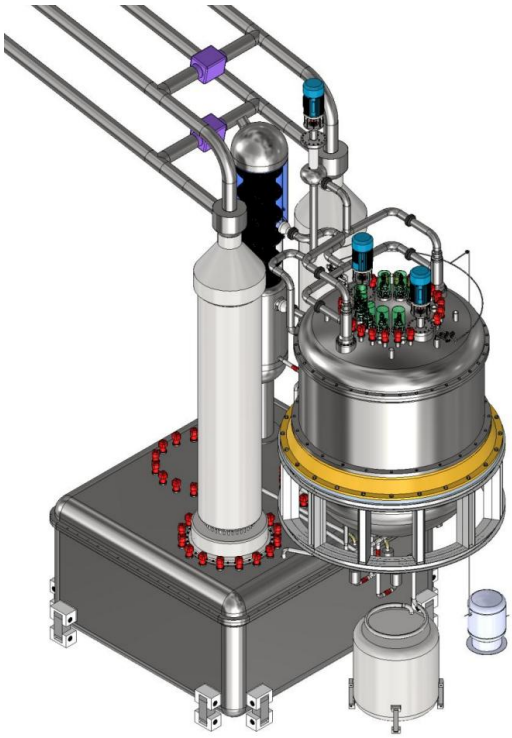
13. Development Milestones

Year	Milestone	Status
1988	The first floating nuclear power plant (FNPP) design has been developed	Completed
2002	The environmental impact assessment was approved by the Ministry of Natural Resources of the Russian Federation	Completed
2006	After several delays the project was revived by Minatom (Ministry for Atmoic Energy of the Russian Federation)	Completed
2012	Pevek was selected as the site for the NPP installation Baltic Shipyard JSC assumed responsibility for the construction, installation, testing and commissioning the first FPU	Completed
2017	Completed of construction and testing of the floating power unit at the Baltic Shipyard	Completed
2018	Dock-side trials, fuel loading, completion of the reactor core final tests, the reactor fisrt criticality	Completed
2019	Transportation of the FPU to the town of Pevek and the connection to the grid on December 19	Completed
2020	Fully commissioned in Pevek on May 22	Completed
2023	The first core refueling of RP-1 was performed	Completed
2024	The first core refueling of RP-2 was performed	Completed

KAERI Consortium, Republic of Korea MARINA



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*KAERI Consortium, Republic of Korea*
Reactor type	*MSR(fast neutron spectrum)*
Coolant/moderator	*Molten chloride salt / N/A*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*100 / ≥35*
NSSS operating pressure (primary/secondary), MPa	*Atmospheric*
Core inlet/outlet coolant temperature (°C)	*540 / 620*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*Dissolved fuel salt*
Number of fuel assemblies in the core	*N/A*
Fuel enrichment (%)	*≤ 19.75*
Core discharge burnup (GWd/ton)	*Undisclosed*
Refuelling cycle (months)	*No refuelling during the ship's service life (25 years)*
Fuel cycle requirements/approach	*N/A*
Reactivity control mechanism	*Control drums*
Approach to safety systems	*Hybrid*
Seismic design (SSE)	*Undisclosed*
Design life (years)	*25*
Plant footprint (m^2)	*Undisclosed*
RV height/diameter (m)	*Undisclosed*
RV weight (metric ton)	*Undisclosed*

1. Introduction

MARINA (Molten-salt Advanced Reactor for Integrated Nautical Applications) is an advanced, small modular reactor (SMR) designed for maritime depolyment. It operates on a fast neutron spectrum utilizing a chloride salt-based liquid fuel. The reactor delivers a thermal power capacity of 100 MW(t), which is converted into an electrical output of 35 MW(e) or higher. Developed by Korean Consortium – led by Korea Atomic Energy Research Institute (KAERI) in collaboration with leading construction engineering firms, major shipbuilders – MARINA is engineered specifically for marine applications, such as ship propulsion and floating nuclear power plants. It operates on HALEU (High-Assay Low-Enriched Uranium) fuel and is designed to run completely without refueling for the entire operational lifetime of the vessel, drastically reducing maintenance overhead and maximizing operational availability. Upon the decommissiong of the vessel, the spent fuel salt can be recycled, offering a highly sustainable and environmentally responsible closed-loop lifecycle.

To optimize the reactor for a fast neutron spectrum, MARINA utilizes chloride-based molten salts. Operating at near-atmospheric pressure, the reactor eliminates the risk of high-pressure catastrophic failures. Its operational safety is governed by inherent and passive safety systems; it features a passive decay heat removal system alongside a emergency drain tank mechanism. In the event of an abnormal transient or emergency, the entire inventory of the

liquid fuel salt is automatically and passively discharged into drain tanks, securing robust, foolproof safety under marine environments.

MARINA maximize versatility through its highly modular design. Depending on the vessel type and required power output, multiple reactor modules can be flexibly combined to meet diverse energy demands. While MARINA is currently in the conceptual design stage, having already received multiple AiPs (Approvals in Principle) from prominent international classification societies alongside major global shipbuilding companies.

2. Target Application

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

MARINA is primarily designed for marine applications, serving as a versatile and high-efficiency marine energy source. It can be deployed for ship propulsion, generating electriity to derive electric motors for large vessel operations. Additionally, MARINA can be configured as a Floating Nuclear Power Plant (FNPP) to supply electricity directly to coastal grids. The FNPP barge can further integrate specialized industrial facilities on-board, utilizing its power and heat for applications such as hydrogen production.

3. Design Philosophy

The primary objective of MARINA is to develop a cost-competitive and inherently safe marine molten salt reactor based on the following guiding principles:

- Integrated & Modular Configuration
- Compactness & Lightweight Design
- Maritime Adaptability
- Localized Solidification Prevention
- Optimized Radiation Shielding
- Defense-in-Depth Confinement
- Corrosion & High-Temperature Integrity
- Ultimate Passive Safety

4. Main Design Features

(a) Nuclear Steam Supply System

The Nuclear Steam Supply System (NSSS) of MARINA transfers nuclear fission energy from the reactor system module to a coolant salt loop and ultimately produces high-temperature steam for electricity generation or industrial applications. In the reactor system module, the molten fuel salt functions as both the fuel and the primary coolant, absorbing heat from the reactor core and transferring it to the secondary coolant salt through the fuel salt heat exchanger.

The coolant salt loop acts as an intermediate heat transport circuit and provides a safety barrier between the radioactive fuel salt and the water/steam system. The heated coolant salt flows to the steam generator, where it transfers heat to produce high-temperature steam. The cooled coolant salt then recirculates to the fuel salt heat exchanger, while the generated steam exits the NSSS boundary and drives a conventional turbine-generator system.

(b) Reactor Core

MARINA reactor features a cylindrical reactor vessel and uses NaCl-KCl-UCl_3 molten salt as its fuel salt. The fissile material is dissolved directly in the molten salt, forming a homogeneous liquid-fuel core. The reactor is configured to maintain a fast neutron spectrum without a moderator, while a surrounding reflector minimizes neutron leakage and improves neutron economy.

(c) Reactivity Control

Reactivity control during normal operation is managed through a combination of control drums located in the reflector and burnable absorbers around the core to compensate for long-term fuel burnup.

(d) Reactor Vessel (RV) and Internals

In the MARINA design, the reactor vessel and internals are defined to encompass the reactor core region, the surrounding reflector, control drums located within the reflector region, and the internal structures required to support the liquid-fuel core configuration. The reactor vessel provides the primary structural boundary of the reactor module and maintains the geometric integrity of the fuel salt, reflector, and control drum assembly.

Since MARINA operates at near-atmospheric pressure, the reactor vessel does not require a thick pressure-retaining structure. Material selection is therefore driven primarily by the high-temperature molten salt environment, radiation exposure, corrosion resistance against NaCl-KCl-UCl_3 fuel salt, irradiation tolerance of internal components, and long-term structural integrity under relevant service conditions.

(e) Reactor Coolant System and Steam Generator

In the MARINA design, the fuel salt loop does not directly transfer heat to the steam generator. Instead, an intermediate heat transport system is employed, where the high-temperature fuel salt transfers its heat to a coolant salt through a primary heat exchanger. The circulating coolant salt then generates steam within the steam generator.

(f) Pressuriser

A pressuriser is not required for the MARINA design, as the system operates at a low pressure close to atmospheric pressure.

(g) Primary pumps

MARINA adopts a forced circulation design to circulate both the fuel salt and coolant salt, utilizing an independent pump for each individual loop.

(h) Distinguishing features

MARINA is being designed to operate throughout the 25-year sevice life of the ship without the need for refueling. This is expected to reduce the burden associated with maintenance and nuclear fuel handling.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The reactor adopts a defence-in-depth safety approach that combines inherent reactor characteristics, engineered safety functions, and operational safety provisions. The design is based on a fast-spectrum molten chloride fuel salt concept, in which the fuel is maintained in liquid form during normal operation.

The safety approach makes use of inherent negative temperature-reactivity feedback. An increase in fuel salt temperature results in a passive reduction of reactivity, supporting reactor power stabilization under abnormal thermal conditions. This inherent response is complemented by engineered functions for reactor shutdown, heat removal, radioactive material confinement, monitoring, and fuel salt inventory management.

The liquid fuel form provides flexibility for establishing a safe configuration under abnormal or accident conditions. Where required by the safety design, fuel salt can be placed in a configuration that supports subcriticality, confinement, and decay heat removal, thereby contributing to long-term stabilization without reliance on complex early operator actions.

The design also includes provisions for managing gaseous and volatile radioactive species generated in the fuel salt. The off-gas management function supports controlled separation, retention, monitoring, and decay of such radionuclides. This function contributes to source term reduction while maintaining radioactive material within controlled confinement boundaries.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

The design basis safety analysis is structured around representative postulated conditions associated with reactivity control, heat removal, fuel salt circulation, confinement, support functions, and marine operating environments. Event grouping, bounding assumptions, acceptance criteria, and safety margins are developed through the plant safety analysis process.

Under abnormal thermal or hydraulic conditions, negative temperature-reactivity feedback provides a stabilizing response by reducing reactor power. Engineered protection and heat removal functions are used to maintain fuel salt, structural, and confinement conditions within established safety limits. Depending on the applicable operating condition and safety limits, the reactor may transition to a reduced-power state or to a safe shutdown state. For selected design basis conditions requiring fuel salt relocation, the drain tank system, where credited in the safety analysis, provides a favorable configuration to support subcriticality, decay heat removal, and confinement of radioactive material.

(m) Safety Approach and Configuration to Manage DEC

The design extension safety analysis considers representative severe conditions involving degradation of normal support functions, impairment of heat removal or circulation capability, multiple equipment unavailability, and marine-related challenges. The corresponding safety approach is developed to maintain core safety functions and confinement under severe postulated conditions.

Under such conditions, the reactor is intended to transition toward a stable shutdown or otherwise safe state through inherent negative reactivity feedback, engineered protection functions, and accident management measures. Where required by the safety design, the fuel salt inventory can be placed in a passively coolable and subcritical configuration to support long-term stabilization.

Monitoring and accident management provisions are provided for key safety parameters, including thermal state, fuel salt inventory, radiological condition, and confinement status. The overall objective is to maintain an inspectable and recoverable plant condition following severe postulated events.

(n) Containment System

The primary confinement function is provided by the fuel salt boundary and associated components. Secondary confinement provisions are used to detect, collect, confine, and monitor radioactive material that could be released from primary boundaries. The off-gas treatment and confinement function is integrated into this overall radioactive material confinement strategy.

For marine application, containment and confinement functions are integrated with the ship safety concept while preserving the independence and reliability of nuclear safety functions. Relevant shipboard conditions, including motion, vibration, flooding, fire, collision-related effects, and isolation from shore-based support, are considered in the containment and confinement approach.

(o) Spent Fuel Cooling Safety Approach / System

The reactor is intended to operate without routine replacement of discrete fuel assemblies during ship operation. Therefore, a conventional shipboard spent fuel assembly cooling system is not part of the normal operating configuration.

6. Plant Safety and Operational Performances

MARINA is designed to enhance plant safety and operational availability by supporting long-duration operation over the expected service life of a marine vessel. The reactor module is intended to operate for approximately 25 years without refueling, thereby eliminating routine fuel replacement during normal vessel operation and reducing the need for fuel-handling activities at ports. The reactor plant design also considers key marine interface requirements, including integration with propulsion and electrical loads, limited onboard space, port operation constraints, and abnormal marine conditions that may affect heat removal and plant availability.

Operational resilience is further enhanced by the use of multiple primary salt pumps within each reactor module. This configuration provides functional redundancy in the primary circulation system. In the event of a single pump failure, the reactor module is expected to continue operating at reduced power, while maintaining adequate heat removal capability within applicable operating limits. When integrated with the vessel propulsion and electrical systems, this capability may support limited power availability for safe-return-to-port functions, where credited in the overall plant and ship design. This contributes to reactor plant availability, fault tolerance, and the safe integration of nuclear-powered marine systems under abnormal operating conditions.

7. Instrumentation and Control Systems

The instrumentation and control (I&C) system of MARINA is designed to satisfy the reliability and safety expectations generally required for nuclear power systems. The system incorporates sufficient redundancy in monitoring, control, and protection functions to support safe operation under normal and abnormal conditions. A high level of automation is applied to reduce operator workload and enable operation with a minimized number of onboard personnel.

8. Plant Layout Arrangement

The plant layout arrangement of MARINA is developed in close coordination with the overall vessel design to achieve an optimized spatial configuration for marine application. Major reactor module components, heat transport systems, auxiliary systems, shielding, access routes, and maintenance spaces are arranged to support safe operation, efficient space utilization, and integration with shipboard systems.

9. Testing Conducted for Design Verification and Validation

A staged testing and verification programme is being conducted to support the design development and validation of the marine molten chloride salt reactor concept. Experimental activities include the preparation and handling of chloride-based molten salts under controlled chemical conditions. These tests support the development of salt purification, salt chemistry control, impurity management, and quality control methods relevant to reactor fuel salt and associated salt systems.

Testing is also being performed on the behaviour and management of gaseous and volatile fission product species. The experimental scope includes collection, separation, retention, monitoring, and treatment concepts for radionuclides that may be generated or transported within the fuel salt system. These activities support the development of off-gas management functions and source term reduction strategies.

Material compatibility testing is being conducted to evaluate corrosion behaviour in high-temperature molten chloride salt environments. The programme considers corrosion mechanisms, salt chemistry effects, material degradation characteristics, and corrosion mitigation methods. The results support material selection, chemistry control strategy, component design, inspection planning, safety analysis assumptions, and regulatory discussions.

10. Design and Licensing Status

Since 2026, MSR developers and the regulatory authority have initiated formal engagement to support the development of a robust licensing basis for MARINA. Feedback from this regulatory communication will be progressively incorporated into the design to ensure that key safety and licensing considerations are addressed from the early stages of development. Through this process, MARINA aims to establish a reactor design suitable for future licensing.

11. Fuel Cycle Approach

At the end of service life, the irradiated fuel salt is expected to be removed and managed through a dedicated back-end fuel cycle process. Recovery and recycling of usable nuclear material may be considered as a potential option where technically feasible and authorized by the competent authorities. The detailed back-end process will be determined in accordance with national policy, safeguards, security, transport, waste management, and regulatory requirements.

12. Waste Management and Disposal Plan

The waste management and disposal plan for MARINA will be established as the reactor design and fuel cycle strategy are further developed. The plan will cover radioactive waste generated during fabrication, operation, maintenance, decommissioning, and post-operation fuel salt handling. It will address waste characterization, minimization, treatment and conditioning, interim storage, transportation, and final disposal in accordance with applicable regulatory requirements. Particular consideration will be given to the chemical and radiological characteristics of chloride fuel salt and activated materials to support safe and traceable waste management throughout the reactor life cycle.

13.Development Milestones

Year	Milestone	Status
2026	Completion of MARINA conceptual design	Ongoing
2027-2032	MARINA basic design and integrated design of MSR-powered ship, Critical experiment and integral effect test	Planned
2033	MARINA and MSR-powered ship construction permit application	Planned
2035	Initiation of MSR-powered ship construction	Planned
2039	Completion of MARINA conceptual design	Planned

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



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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*
Design status	*Under Construction*
Thermal/electrical capacity, MW(t)/MW(e)	*198 / 50*
NSSS operating pressure (primary/secondary), MPa	*15.7 / 3.83*
Core inlet/outlet coolant temperature (°C)	*284 / 321*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO2 (cermet fuel) pellets / hexagonal*
Number of fuel assemblies in the core	*241*
Fuel enrichment (%)	*<20*
Core discharge burnup (GWd/ton)	*Undisclosed*
Refuelling cycle (months)	*Up to 120*
Fuel cycle requirements/approach	*Without on-site refueling*
Reactivity control mechanism	*Control rod drive mechanism; within the control and protection system (CPS)*
Approach to safety systems	*Combined active and passive system*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*3360*
RV height/diameter (m)	*8.6 / 3.45*
RV weight (metric ton)	*265*

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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 are 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 Vessel (RV) and Internals

The RITM-200M reactor has a four-loop forced and natural circulation coolant loop; the latter is used only in the emergency heat removal mode. Reactors of this type is utilized at universal nuclear icebreakers.

(e) Reactor Coolant System and 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.

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.

(f) Pressuriser

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.

(g) Primary pumps

The reactor is equipped with four sealed pumps of the primary circuit.

(h) Distinguishing features

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.

(i) Engineered Safety System Approach and Configuration

The high safety level of RITM series reactors is achieved both by inherent safety features and a combination of passive and active safety systems. Moreover, redundancy of safety system equipment and channels and their functional and/or physical separation are provided to ensure high reliability. Safety systems are driven automatically by the control system, when controlled parameters achieve appropriate set points. In case of automated systems failure, self-actuating devices will actuate directly under the primary circuit pressure to ensure reactor trip and initiate the safety systems.

(j) Decay Heat Removal System

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

- Active safety loop with forced circulation through steam generator.
- Active safety loop with forced circulation through primary-third circuit heat exchanger of 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.

(k) 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.

(l) Safety Approach and Configuration to Manage DBC

Active safety systems are used to manage design basis accidents. The automatic control system ensures the transfer of the reactor plant to a safe state in automatic mode without the participation of personnel. Safety systems are designed for all types of design basis and beyond design basis accidents. Each safety system has at least two channels, each channel is designed to perform the safety function in full. Efficiency and sufficiency of safety systems is confirmed by calculations and justified in the safety analysis report.

(m) Safety Approach and Configuration to Manage DEC

The following is used to control beyond design basis accidents:

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- means of beyond design basis accident management in case of failure of safety systems;
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Switching the reactor plant to a safe state is carried out by any available means automatically or remotely. To ensure each safety function, there are active and passive safety systems, as well as special technical means. The design provides systems for retaining molten fuel in the reactor vessel in the event of a severe accident. Efficiency and sufficiency of safety systems and technical means is confirmed by calculations and justified in the safety analysis report. At the disposal of the personnel there is a guide for managing beyond design basis accidents, containing additional instructions for restoring safety functions in beyond design basis conditions. At the disposal of personnel there is an emergency communication channel with the RP designer for obtaining additional instructions and consultations on eliminating beyond design basis conditions.

(n) 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.

(o) Spent Fuel Cooling Safety Approach / System

Storage of spent fuel on the vessel is not carried out. Fuel is unloaded into a separate storage at a specialized enterprise.

6. Plant Safety and Operational Performances

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.

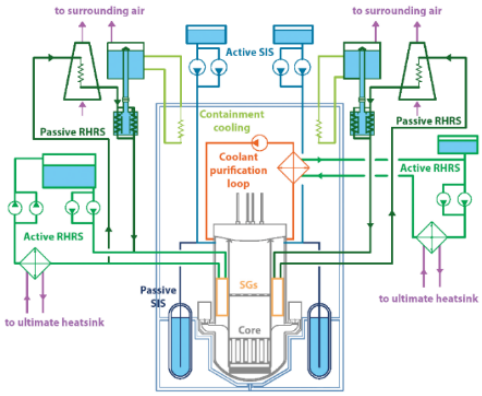


Figure 1. RITM-200M reactor safety system

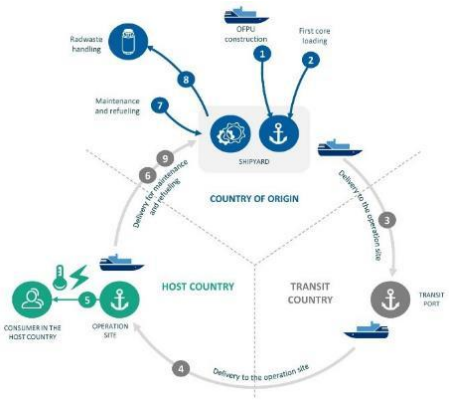


Figure 2. OFPU life cycle

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement



Figure 3. OFPU fitted with the RITM-200M reactor 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

9. Testing Conducted for Design Verification and Validation

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. Fuel Cycle Approach

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 and 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

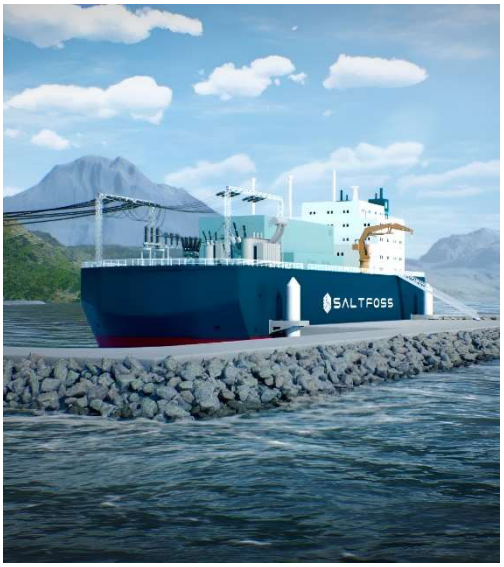
Year	Milestone	Status
2016	The first RITM-200 installed on board the icebreaker Arktika	**Completed**
2020	The icebreaker Arktika under testing	**Completed**
2020	A preliminary design of the FPU with RITM-200M reactor plant	**Completed**

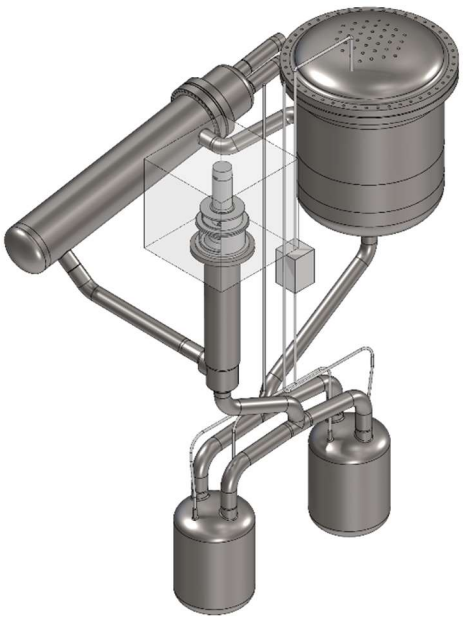
Year	Milestone	Status
2021	The icebreaker Sibir was commissioned in December	**Completed**
2022	The operation of the icebreaker Ural started in November	**Completed**
2025	The operation of the icebreaker Yakutia started in March	**Completed**
2026	The detailed designs of FPU and RITM-200M reactor plant were completed	**Completed**

Saltfoss Energy, Denmark
seaMSR-100



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Saltfoss Energy, Denmark*
Reactor type	*MSR (Thermal Spectrum)*
Coolant/moderator	*Fluoride fuel salt / graphite*
Design status	*Basic Design*
Thermal/electrical capacity, MW(t)/MW(e)	*250 / 110*
NSSS operating pressure (primary/secondary), MPa	*0.8 / 1.0*
Core inlet/outlet coolant temperature (°C)	*600 / 650*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*LEU / Molten salt fuel*
Number of fuel assemblies in the core	*N/A*
Fuel enrichment (%)	*≤ 2*
Core discharge burnup (GWd/ton)	*15*
Refuelling cycle (months)	*Online fuel addition*
Fuel cycle requirements/approach	*LEU with online fuel addition*
Reactivity control mechanism	*Negative temperature coefficients, regulating and safety rods, fuel salt draining*
Approach to safety systems	*Passive*
Seismic design (SSE)	*N/A*
Design life (years)	*24*
Plant footprint (m^2)	*2500-15000m2 (100-600Mwe, Depending on power barge configuration)*
RV height/diameter (m)	*5.5 / 5*
RV weight (metric ton)	*240*

1. Introduction

The Saltfoss seaMSR-100 is an advanced, small modular molten salt reactor characterized by a liquid fluoride molten fuel salt and a graphite moderator. The reactor has a thermal power capacity of 250 MWth which is converted into 100 MWe for grid transmission in addition to 10MWe to cover house loads. It operates on standard LEU fuel with enrichment below 5% U-235, making use of high-frequency online refuelling to achieve its operating lifetime of 24 years while maintaining low excess reactivity and minimizing control rod movement. To facilitate inspection and maintenance, the seaMSR-100 is a loop-type reactor implementing a single fuel salt circulation loop. Due to its very high boiling point, the fuel salt circuit operates at low pressure, 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 seaMSR-100 is deployed on modular and standardized non-self-propelled Power Barges, scalable from 100 MWe to 600 MWe

(containing one to six so-called Power Modules, each equipped with a single seaMSR-100), each power module has an operating life of 24 years.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☒ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☐ Desalination	☒ >550°C

The seaMSR Power Barge (MSPB) is designed to generate power for the electrical grid and for industrial applications. The first of a kind MSPB will deliver electricity, however additional applications are of interest for subsequent and Nth of a kind projects. For subsequent projects, applications under investigation include high temperature industrial heat or steam supply for industrial processes.

3. Design Philosophy

The overarching design goal with the seaMSR Power Barge is to centrally produce highly standardized, scalable and cost-competitive nuclear power plants that can be reliably deployed within three years from the date of order. The design philosophy is guided by the following:

- Proliferation resistant design enabling worldwide deployment.
- Safety and integrity achieved by passive means.
- Reducing the number and complexity of nuclear-grade components.
- Modularity of design to facilitate efficient construction, transportation, installation, commissioning, maintenance, and decommissioning.

4. Main Design Features

(a) Nuclear Steam Supply System

The fuel salt 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, leveraging technology from concentrated solar power plants. The Power Conversion System receives heat in the form of superheated steam at temperatures around 550ºC. The steam produced drives a conventional condensing steam turbine. Electrical output from the connected generators allows delivery of 100 MWe net from each Power Module through a common electrical interface to the high-voltage onshore electrical transmission grid. Through this arrangement, a seaMSR Power Barge can provide between 100 and 600 MWe net to the grid depending on the number of modules installed.

(b) Reactor Core

The reactor core consists of graphite blocks arranged into columns in a central active core region and a reflector region in the periphery. In total, the graphite core measures approximately 4.6m in diameter and 4 m in height. The fuel salt flows through channels in the graphite core configuration and transports heat deposited into the fuel salt, graphite, and structural materials to the primary heat exchanger. 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 seaMSR-100 credits two main methods for reactivity control: control rods and fuel salt draining, which are fast and slow acting, respectively. Inherent reactivity control means include negative temperature coefficients and the geometrical configuration of the fuel salt drain tanks which ensure sub-criticality.

Regular fuel addition ensures that minimal excess reactivity is maintained across the entire fuel cycle.

(d) Reactor Vessel (RV) and Internals

The reactor vessel measures approximately 5m in diameter, accommodating the graphite active core and reflector regions. An upcomer 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. The upper vessel head contains penetrations for the control rod guide tubes and inspection channels. The reactor vessel internals include the control rod guide tubes, structural supports maintaining alignment of graphite structure, and equipment for material surveillance.

(e) Reactor Coolant System and Steam Generator

(f) Pressuriser

(g) Primary pumps

The fuel salt circuit contains a single primary molten salt pump. The pump is a centrifugal cantilever design, with the pump motor located in a shielded coffer dam for maintenance purposes.

(h) Distinguishing features

The fuel consists of NaF, KF and UF4. 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 enter the cover gas. 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

(i) Engineered Safety System Approach and Configuration

The seaMSR-100 follows a risk-informed safety approach inspired by the U.S. NRC 10 CFR Part 53 regulatory framework. The aim is to implement an effective safety approach that properly considers the inherent safety characteristics of the seaMSR-100, simplifying the design while improving safety.

The safety approach is based on a systematic and iterative safety implementation process comprising selection of representative licensing-basis events (LBEs), identification of safety functions and associated design features (SSCs and inherent characteristics), safety classification of SSCs, and evaluation of DiD; in order to verify compliance with safety acceptance criteria.

Safety acceptance criteria have been defined to ensure the overarching seaMSR-100 safety goal of eliminating the need for off-site emergency response under all identified LBEs, with an explicit consideration of DiD levels.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

The safety architecture comprises the full set of design features and programmatic controls required to fulfill safety functions and effectively limit the release of radioactive materials from the plant.

The seaMSR-100 safety architecture consists of a combination of active, passive and inherent safety means to verify compliance with safety acceptance criteria.

DiD level 3 relies exclusively on passive and inherent safety means to cope with unlikely event sequences and DBAs, thus reducing reliance on human intervention and ensuring sufficient safety margins. Fuel salt decay heat removal is ensured by a fully passive and redundant heat removal system, which provides indefinite cooling capability, robustness, and reliability considering its continuous operation.

DiD levels 2 and 4 rely on a combination of active, passive and inherent means used to cope with anticipated event sequences and very unlikely event sequencies, respectively.

The seaMSR-100 utilizes a functional containment to meet the safety acceptance criteria for DiD level 4, with a set of barriers effectively limiting the release of radioactive material to the environment: The combination of inherent, passive and active means that are credited to effectively limit radioactive releases under each LBE are identified and classified according to their safety significance.

The overarching safety goal of the design and safety architecture is to eliminate the need for off-site emergency response under all LBEs.

An overview of the main seaMSR-100 design features available to fulfil the fundamental safety functions under LBEs is provided below.

Fundamental Safety Function	Design features
Control of reactivity	Control of reactivity across different LBE's and levels of DiD is achieved by a combination of negative reactivity coefficients, regulating rods, safety rods and emergency fuel salt draining.
Heat removal	A combination of active (heat transport train), passive and diverse means are implemented to robustly remove decay heat from the fuel salt and cover gas across all LBE's and levels of DiD, thereby ensuring barrier acceptance criteria are met.
Confinement	The fuel salt and cover gas is confined by multiple barriers to limit radioactive releases including leak tight metallic barriers and additional metallic enclosures forming part of a functional containment. Furthermore, inherent features such as fission product retention of the fuel salt, a large margin to boiling, and absence of energetic phenomena complement the performance of the containment barriers in eliminating the need for off-site emergency response under all LBEs.

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

The fuel salt and cover gas is confined by multiple barriers to limit radioactive releases including leak tight metallic barriers and additional metallic enclosures forming part of a functional containment. Furthermore, inherent features such as fission product retention of the fuel salt, a large margin to boiling, and absence of energetic phenomena complement the performance of the containment barriers in eliminating the need for off-site emergency response under all LBEs.

(o) Spent Fuel Cooling Safety Approach / System

Once a power module reaches its design lifetime, fuel salt is drained to the drain tanks for intermediate storage. Cooling is achieved via the same means as accident scenarios, whereby long-term cooling is managed via passive decay heat removal through a drain tank cavity cooling system. Fuel salt will be stored in a solidified state. This configuration provides a long term, passive, redundant solution to spent fuel cooling without the need for additional systems or active management.

6. Plant Safety and Operational Performances

The inherenent and passive safety characteristics of the seaMSR-100 translate to improved performance across the safety attributes and associated indicators in TECDOC-1141. With respect to the attribute "plant operates smoothly," the low-pressure primary circuit, high boiling point of the fuel salt, and negative temperature feedback coefficients reduce the frequency and magnitude of operational transients that would otherwise result in shutdown events, forced power reductions or unplanned outages.

7. Instrumentation and Control Systems

The I&C systems provide control, monitoring, and protection functions. The I&C architecture reflects the modular nature of the Power Barge and is designed to ensure separation of safety and non-safety systems. Each seaMSR-100 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 systems in a safe state. Taking advantage of inherent and passive safety features, the reactor protection system is simplified with minimal automated actions required across licensing basis events.

8. Plant Layout Arrangement

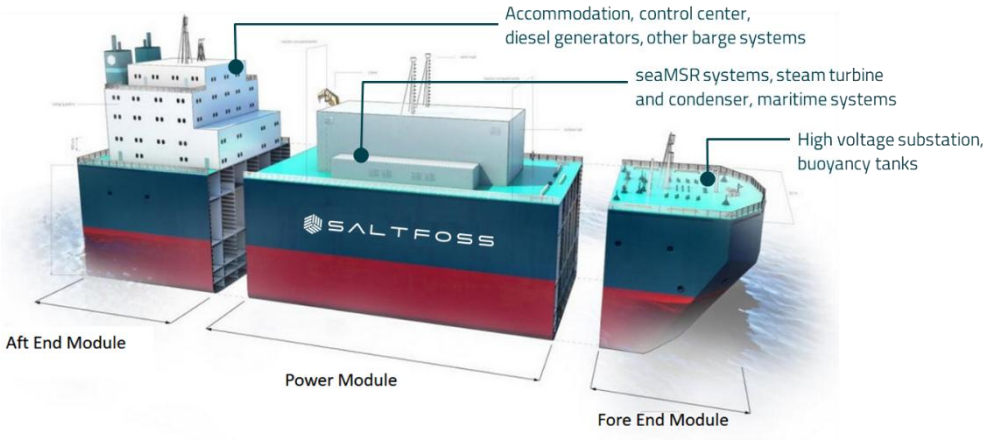


Figure 1. seaMSR-100 Plant Layout

The seaMSR 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, auxiliary 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 reactor compartment containment system and auxiliary systems. The Power Barge is not self-propelled and shall be equipped with suitable towing and mooring facilities.

9. Testing Conducted for Design Verification and Validation

Saltfoss' design verification and validation (V&V) program for the first-of-a-kind seaMSR-100 has been structured to support both regulatory licensing and systematic technology maturation. From a licensing perspective, testing activities are aligned with the need to demonstrate that safety functions are reliably achieved across the complete licensing basis. This includes generating evidence to support safety analyses, confirming the performance of structures, systems, and components credited in the safety case, and reducing key uncertainties associated with materials, thermal-hydraulic behavior, and salt chemistry. The V&V approach follows a risk-informed, performance-based methodology, prioritizing testing based on safety significance and incorporating regulatory feedback to ensure that the resulting data provides an adequate basis for design substantiation and licensing.

10. Design and Licensing Status

The seaMSR-100 is in the basic design phase. Saltfoss has been in informal pre-licensing discussions with the Korean regulatory body and has communicated the intent to use the current two-step licensing process. Saltfoss is also taking part in initiatives to inform and prepare for licensing of advanced reactors in South Korea. Saltfoss is ready for formal licensing activities to commence by the end of 2026.

11. Fuel Cycle Approach

The initial fuel salt inventory for the seaMSR-100 will be produced at a centralized fuel facility, packed and transported under a controlled atmosphere as fluoride salt pellets, and loaded into the fuel salt drain tanks. 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 additional fuel salt volume, together with volumetric changes due to thermal expansion, are accommodated by the fuel salt drain tanks. At the end of the 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

Year	Milestone	Status
2020	Feasibility Statement received from ABS	Completed
2022	Power Barge Approval in Principle from ABS	Completed
2025	End of concept verification	Completed
2026	Start of pre-licensing	Planned
2029	Construction Permit Application	Planned
2030	Start of Construction	Planned
2033	Construction Complete	Planned

JSC “Afrikantov OKBM”, Russian Federation VBER-300



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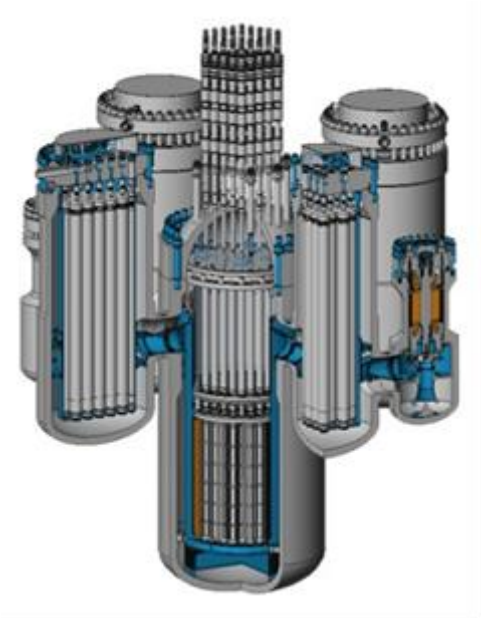


Figure 1: VBER-300 design concept

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*
Design status	*Licensing stage*
Thermal/electrical capacity, MW(t)/MW(e)	*917 / 325*
NSSS operating pressure (primary/secondary), MPa	*16.3*
Core inlet/outlet coolant temperature (°C)	*292 / 325*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UO2 pellet/hexagonal*
Number of fuel assemblies in the core	*85*
Fuel enrichment (%)	*4.95*
Core discharge burnup (GWd/ton)	*50*
Refuelling cycle (months)	*72*
Fuel cycle requirements/approach	*N/A*
Reactivity control mechanism	*Control rod rods and soluble boron*
Approach to safety systems	*Hybrid (active and passive) system*
Seismic design (SSE)	*0.25g*
Design life (years)	*60*
Plant footprint (m^2)	*N/A*
RV height/diameter (m)	*9.3 / 3.9*
RV weight (metric ton)	*N/A*

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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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 UO2 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 UO2 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 Vessel (RV) 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 and 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.

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) Pressuriser

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.

(g) Primary pumps

N/A

(h) Distinguishing features

Power source for transportable Floating NPPs, cogeneration options, compact design.

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

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.

(m) Safety Approach and Configuration to Manage DEC

The Emergency Core Cooling System (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.

(n) 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.

(o) Spent Fuel Cooling Safety Approach / System

N/A

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. Instrumentation and Control Systems

N/A

8. Plant Layout Arrangement

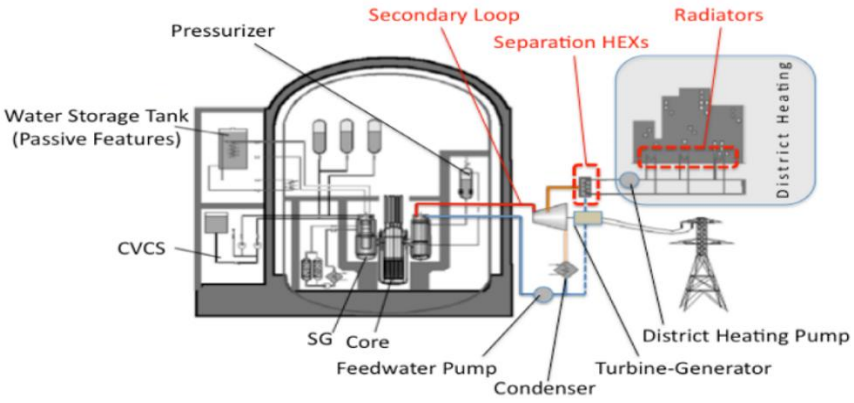


Figure 2. Plant layout

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.

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.

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.

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.

9. Testing Conducted for Design Verification and Validation

N/A

10. 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.

11. 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.

12. Waste Management and Disposal Plan

N/A

13. Development Milestones

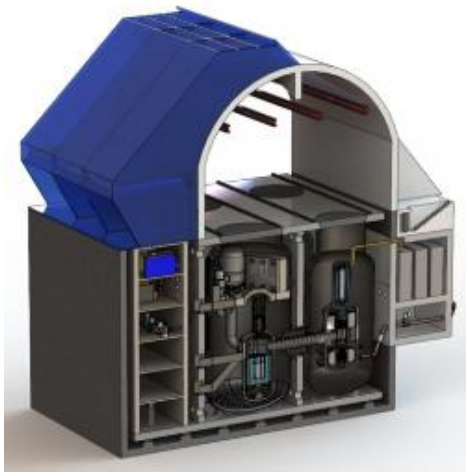
Year	Milestone	Status
2001	Technical and commercial proposal for the two-unit VBER NPP	**Completed**
2002	Preliminary design 1 approved by the Scientific and Technical Board and State Nuclear Supervision Body (GosAtomNadzor)	**Completed**
2004	JSC “Kazakhstan-Russian company “Atomic stations” was established to promote the VBER-300 design.	**Completed**
2006	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	**Completed**
2007-2009	Development of the 100–600 MW VBER plant	**Completed**
2007-2008	R&D for the VBER-460/600 NPP design	**Completed**
2008-2011	Development of the VBER-600/4 NPP based on the heat exchange loop of the increased capacity	**Completed**
2011-2012	Technical and economic optimization of the VBER-600/4 plant	**Completed**
2012-2015	Technical and commercial proposal for the two-unit VBER NPP	**Completed**

HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS

General Atomics, USA
EM2 (Energy Multiplier Module)



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*General Atomics (GA), United States of America*
Reactor type	*HTGR*
Coolant/moderator	*He / N/A*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*500 MW(t) / 265 MW(e) per module, 1060 MW(e) total*
NSSS operating pressure (primary/secondary), MPa	*13.3 / NA*
Core inlet/outlet coolant temperature (°C)	*550 / 850*
Primary circulation	*Driven by the Power Conversion Unit (PCU), no separate pumps*
Fuel type/assembly array	*Uranium Carbide (UC), Hexagonal Assemblies*
Number of fuel assemblies in the core	*85*
Fuel enrichment (%)	*< 17*
Core discharge burnup (GWd/ton)	*140*
Refuelling cycle (months)	*360*
Fuel cycle requirements/approach	*Open cycle*
Reactivity control mechanism	*Control rods*
Approach to safety systems	*Passive Safety Systems*
Seismic design (SSE)	*0.3 g*
Design life (years)	*60*
Plant footprint (m^2)	*5000*
RV height/diameter (m)	*12.5 / 4.6*
RV weight (metric ton)	*280*

1. Introduction

Energy Multiplier Module (EM2) 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. EM2 employs a closed Brayton cycle gas turbine power conversion unit with a Rankine bottoming cycle for 53% net power conversion efficiency, assuming evaporative cooling. EM2 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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

The EM2 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.

(b) 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.

(c) 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.

(d) Reactor Vessel (RV) 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.

(e) 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.

(f) Pressuriser

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

(g) Primary pumps

Primary coolant circulation is managed by the turbo-compressor 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.

(h) Distinguishing features

EM2 core was designed for convert-and-burn operation that lasts 30 years without the need to add or shuffle fuel. The end-of-cycle fuel can be refabricated in a manner that does not separate out heavy metal, permitting reuse in subsequent generations at reduced proliferation risk. Waste problems are mitigated by higher burnup, fuel use in multiple generations, and conversion of existing waste to energy. Economically attractive power costs are achievable through a combination of high efficiency, simplicity of the direct-cycle gas turbine, and relatively small subsystems that can be shop-fabricated and shipped by road to the site.

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) 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.

(m) 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.

(n) 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.

(o) 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.

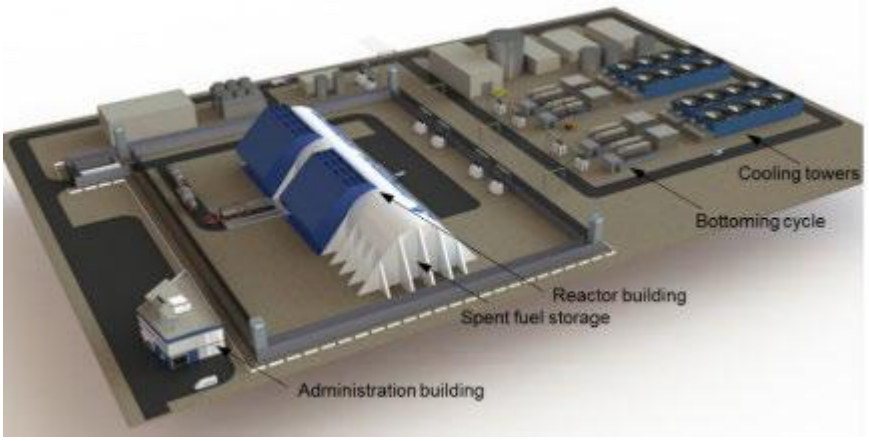
6. Plant Safety and Operational Performances

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.

9. Testing Conducted for Design Verification and Validation

Design verification and validation for the EM2 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

EM2 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 license modernization framework.

11.Fuel Cycle Approach

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

12.Waste Management and Disposal Plan

EM2 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.

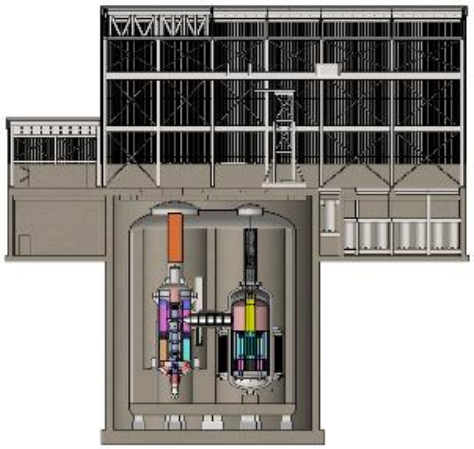
13.Development Milestones

Year	Milestone	Status
2009	Start of design	**Completed**
2013	Pre-conceptual design	**Completed**
2026	Pre-application license review	**Ongoing**

General Atomics, USA
FMR (Fast Modular Reactor)



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MAJOR 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*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*100 MW(t) / 44 MW(e)*
NSSS operating pressure (primary/secondary), MPa	*7 / N/A*
Core inlet/outlet coolant temperature (°C)	*506 / 800*
Primary circulation	*Driven by the Power Conversion Unit (PCU), no separate pumps*
Fuel type/assembly array	*Uranium Dioxide (UO2) / 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*
Fuel cycle requirements/approach	*Open cycle with interim storag0e*
Reactivity control mechanism	*Control rods, Burnable absorbers (B4C)*
Approach to safety systems	*Combination of active and passive systems, no core melt provisions, reactor vessel passive cooling system*
Seismic design (SSE)	*0.3 g*
Design life (years)	*60*
Plant footprint (m^2)	*2,000 m² (reactor building) / 38,000 m² (site)*
RV height/diameter (m)	*11 / 4.6*
RV weight (metric ton)	*230*

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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 Vessel (RV) 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.

(h) Distinguishing features

Helium coolant, SiC composite cladding, dry-cooling heat sink, closed, Brayton cycle, passive safety

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) 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.

(m) 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.

(n) 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.

(o) 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.

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 UO2 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 UO2 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

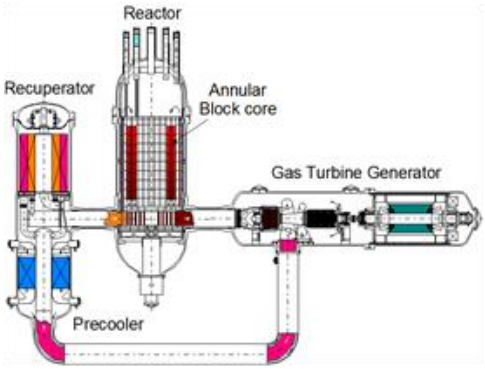
Year	Milestone	Status
2022	Start of design	**Completed**
2026	Concept design complete	**Ongoing**

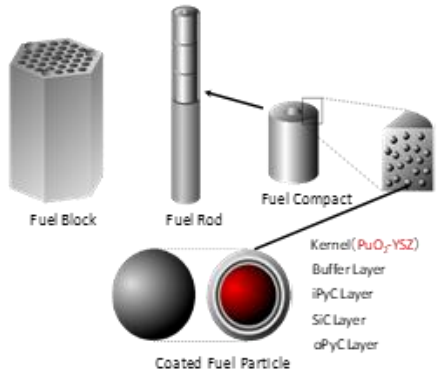
Year	Milestone	Status
2027	Start of pre-licensing vendor design review (in USA)	Planned
2029	Engineering design complete/ Secure necessary licenses (in USA)	Planned
2030	Start construction of a demonstration unit (in USA)	Planned
2036	Commercial operation	Planned

JAEA Consortium, Japan
GTHTR300



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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*
Design status	*Basic design*
Thermal/electrical capacity, MW(t)/MW(e)	*<600 / 100~300*
NSSS operating pressure (primary/secondary), MPa	*7 / 7*
Core inlet/outlet coolant temperature (°C)	*587-633 / 850-950*
Primary circulation	*Forced by gas turbine*
Fuel type/assembly array	*UO2 TRISO ceramic coated particle*
Number of fuel assemblies in the core	*90*
Fuel enrichment (%)	*14*
Core discharge burnup (GWd/ton)	*120*
Refuelling cycle (months)	*48*
Fuel cycle requirements/approach	*Uranium once through (initially)*
Reactivity control mechanism	*Control rod insertion*
Approach to safety systems	*Active and passive*
Seismic design (SSE)	*>0.18g automatic shutdown*
Design life (years)	*60*
Plant footprint (m^2)	*~250 x 250 (4-reactor plant)*
RV height/diameter (m)	*23 / 8*
RV weight (metric ton)	*~1000*

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 850-950oC. 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 Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☒ >550°C

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 m3/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 CO2 emission.

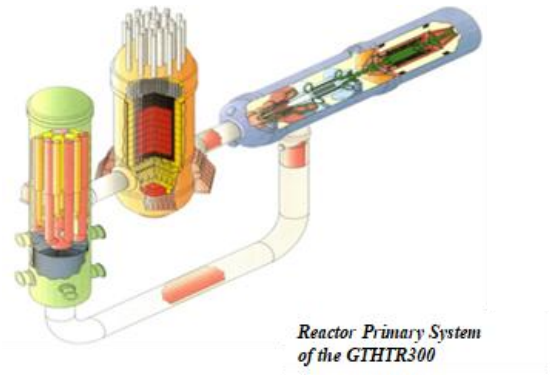
3. Design Philosophy

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) Nuclear Steam Supply System

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.



Reactor Primary System of the 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.

The fuel design is coated fuel particle of less than 1 mm in diameter. Each particle consists of a UO2 kernel coated by four layers of low and high density pyro-carbon and silicon carbide. The all ceramic particle fuel is heat resistant up to 1600oC. Approximately 10 000 particles are packaged into a compact of the size of a thumb. The compacts are then assembled into graphite-clad fuel rods. The fuel rods are inserted into the bore holes of a hexagonal graphite fuel block of about 1 m long and 41 cm across, where the annulus formed between the fuel rod and the bore hole

provides coolant flow channels. The fuel blocks are loaded into the reactor core. The more fuel blocks are placed in the core, the higher the power output of the reactor.

(c) 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.

(d) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

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.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

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 1600oC.
- Reactor helium coolant is chemically inert and thus absent of explosive gas generation or phase change.
- Graphite-moderated reactor core provides negative reactivity coefficient, low-power density, and high thermal conductivity.

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

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) 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.

(o) 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.

6. Plant Safety and Operational Performances

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 Systems

The instrumentation and control system consists of reactor and process instrumentations, control systems, safety protection system, and engineered safety features actuating systems. The six (6) fundamental controls are for turbine bypass, inventory, reactor outlet temperature, turbine inlet temperature, process heat supply rate and IHX differential pressure controls. They are combined in the basic plant control to provide controllability for a variety of transients including loss-of-load and electric load following as shown above.

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 UO2, 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

Year	Milestone	Status
2003	Basic design of GTHTR300 completed	Completed
2005	Design for cogeneration plant GTHTR300C	Completed
2015	Basic design for HTTR-connected gas turbine and H2 plant (HTTR-GT/H2) for system demonstration	Ongoing
2020 ~	HTTR-H2 test plant construction and operation (planned)	Ongoing
2040 ~	Operation of demonstration plant (TBD)	Planned

KAERI, Republic of Korea
HECTAR



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*KAERI, Republic of Korea*
Reactor type	*HTGR*
Coolant/moderator	*He / Graphite*
Design status	*Basic Design*
Thermal/electrical capacity, MW(t)/MW(e)	*90 / 0*
NSSS operating pressure (primary/secondary), MPa	*6.5 / 18*
Core inlet/outlet coolant temperature (°C)	*300 / 750*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*UCO with TRISO in prismatic blocks / hexagonal*
Number of fuel assemblies in the core	*240 fuel blocks*
Fuel enrichment (%)	*<19.75*
Core discharge burnup (GWd/ton)	*>60 [TBC]*
Refuelling cycle (months)	*48*
Fuel cycle requirements/approach	*Open / once-through; HALEU UCO TRISO*
Reactivity control mechanism	*Control rods*
Approach to safety systems	*Combined active and passive system*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*20,000 [TBC]*
RV height/diameter (m)	*≈28.5 / ≈5.2 [TBC]*
RV weight (metric ton)	*≈1100 [TBC]*

1. Introduction

The HECTAR (HElium Cooled Thermal Application Reactor) is a fourth-generation, prismatic-type High-Temperature Gas-cooled Reactor (HTGR) with a thermal output of 90 MWth and a core exit temperature of 750 °C. It is designed to supply high-quality industrial steam, exceeding 500 °C, for processes such as hydrogen production and other chemical applications. The reactor is fuelled by Uranium Oxycarbide (UCO) enriched to High-Assay Low-Enriched Uranium (HALEU). This fuel is used in TRISO fuel compacts embedded in hexagonal graphite blocks, which serve as the fuel blocks. As a modular design, the HECTAR is intended for widespread deployment. The initial "First-of-a-Kind" (FOAK) unit will not fully realize the benefits of modularity; however, these advantages will be achieved with subsequent units.

2. Target Application

☐ Electricity	☒ Hydrogen	☒ Industrial process heat
☐ Baseload	☒HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

3. Design Philosophy

HECTAR is the first HTGR concept developed independently in the Republic of Korea and is built on three principles: inherent safety, design simplicity and modular deployability. Safety is secured by physical characteristics rather than by operator action or AC power — Multiple set of barriers including TRISO coated-particle fuel acting as a functional containment, a graphite core with large thermal inertia and a negative temperature coefficient, and a unit power of 90 MWth low enough for decay heat to be rejected passively to the environment. Economic competitiveness is pursued through a 48-month refuelling interval, a compact 20 000 m² footprint that permits siting next to industrial heat users.

4. Main Design Features

(a) Nuclear Steam Supply System

The NSSS is arranged as three coupled circuits: a closed primary helium loop, a closed secondary steam loop, and an open tertiary loop that delivers steam to the industrial process user. In the primary circuit the reactor vessel and the steam generator vessel are joined by a horizontal coaxial cross-vessel; helium leaves the core at 750 °C and 6.5 MPa, raises steam at 575 °C and 18 MPa in the once-through helical-coil steam generator, and is returned to the core inlet at 300 °C by a single circulator. The closed secondary loop transfers its heat to the open tertiary loop through a steam-to-steam reboiler, so that the process steam delivered to the user never contacts the primary or secondary inventory and two barriers are maintained between the reactor coolant and the industrial site.

(b) Reactor Core

The core is an array of hexagonal graphite blocks loaded with fuel compacts containing UCO TRISO fuel particles and surrounded by graphite reflectors. Fuel is HALEU enriched below 19.75 %, and the moderator and reflector blocks use isotropic nuclear-grade graphite. The core operates on a 48-month refuelling cycle.

(c) Reactivity Control

Reactivity is controlled by control rods inserted through top-mounted control rod drive mechanisms. A diverse and independent reserve shutdown system, based on boronated absorber balls gravity-fed into dedicated core channels, provides a second means of achieving and maintaining subcritical conditions. The strongly negative fuel temperature coefficient acts as a reactor shutdown means that covers anticipated transients without scram.

(d) Reactor Vessel (RV) and Internals

The reactor vessel is a low alloy steel pressure vessel of SA-508/SA-533 class material, held within conventional light water reactor vessel temperature limits by routing the cold helium flow along the inner vessel wall. The internals comprise the graphite core support structure, the metallic core barrel and core support plate, and the upper plenum shroud, arranged so that fuel and replaceable reflector blocks are handled vertically through the top head. Top-head penetrations accommodate the control rod drive mechanisms, the reserve shutdown equipment and refuelling access, while the bottom head connects to the shutdown cooling circuit.

(e) Reactor Coolant System and Steam Generator

The reactor coolant system comprises one electrically driven helium circulator and one once-through helical-coil steam generator, both housed in the steam generator vessel adjacent to the reactor vessel. Feedwater enters at the bottom of the helical bundle and superheated steam is delivered to the closed secondary loop at approximately 18 MPa and above 575 °C, from which heat is passed to the tertiary process steam loop. Because the secondary pressure exceeds the primary pressure, a tube leak results in water ingress into the primary circuit rather than a helium release; this is detected by primary moisture monitors and mitigated by steam generator isolation and dump.

(f) Pressuriser

No pressuriser is required, since the helium coolant is single-phase and non-condensing over the whole operating range. Primary system pressure is established by the helium inventory and is regulated by the helium service system, which also maintains the coolant environmental conditions.

(g) Primary pumps

Primary circulation is provided by a single electric helium circulator mounted on the steam generator vessel, using magnetic bearings and a sealed casing so that no oil or water can be introduced into the primary circuit. The circulator is used during normal operation; on loss of power it coasts down and is isolated by a check valve, after which decay heat is removed by convection and radiation to the passive reactor cavity cooling system (RCCS).

(h) Distinguishing features

HECTAR is the first high-temperature gas-cooled reactor concept designed independently in the Republic of Korea and it is optimised for the delivery of process steam above 500 °C rather than for electricity generation. The inherent safety features and functional containment, of which the TRISO fuel forms part, support a substantially reduced emergency planning zone. The 90 MW(t) module size and the 48-month refuelling interval are selected to match the scale and the outage tolerance of industrial and hydrogen-production users.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

HECTAR combines active and passive systems: active shutdown cooling is provided for investment protection and normal shutdown, while the safety function of residual heat removal is fulfilled entirely by passive means. The reactor cavity cooling system transfers decay heat from the vessel surface by radiation and convection to natural-circulation water-cooled panels, requiring no AC power for pumps, valves or fans.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

Design basis conditions, including loss of forced circulation, pressurised and depressurised conduction cooldown, and station blackout, are managed without active systems: the negative temperature coefficient shuts the reactor down and decay heat is transferred through the core and vessel to the reactor cavity cooling system. Peak fuel temperatures in these events are demonstrated to remain below the TRISO integrity limit, so that fission products are retained within the coated particles.

(m) Safety Approach and Configuration to Manage DEC

Design extension conditions are addressed by the same inherent mechanisms rather than by dedicated severe accident systems, because the low core power density and the large graphite thermal inertia stretch the core heat-up transient over several days. Specific provisions include the diverse reserve shutdown system for anticipated transients without scram, moisture monitoring with automatic steam generator isolation for large water ingress, and a filtered and monitored release path from the reactor building.

(n) Containment System

HECTAR adopts a vented low-pressure confinement rather than a leak-tight high-pressure containment, consistent with the functional containment approach in which the TRISO coated particle is the principal barrier to fission product release. The reactor vessel, cross-vessel and steam generator vessel are installed in a below-grade concrete silo that provides shielding, missile and external hazard protection, and a controlled filtered release path.

(o) Spent Fuel Cooling Safety Approach / System

Discharged fuel blocks are transferred to a shielded interim spent fuel storage facility inside the reactor building, where decay heat is removed by conduction and natural convection to the surrounding structure. After the on-site cooling period the blocks are suitable for direct loading into dry storage casks.

6. Plant Safety and Operational Performances

The plant is designed for a 60-year life with a target availability above 80 %, supported by the 48-month refuelling interval and by on-line maintainability of non-safety systems. The module can be operated in a heat-only mode or in combination with high-temperature steam electrolysis for hydrogen production.

7. Instrumentation and Control Systems

The I&C architecture applies defence in depth through a safety-related reactor protection system, a non-safety plant control system and a diverse actuation capability, implemented on qualified digital platforms with physical and electrical separation between redundant divisions. Plant conditions are monitored by ex-core neutron flux detectors together with helium temperature, pressure, flow and moisture instrumentation, moisture detection being the primary indication of steam generator tube leakage.

8. Plant Layout Arrangement

The nuclear island is arranged around a below-grade reactor building in which the reactor vessel, cross-vessel and steam generator vessel share a common silo, with fuel handling and storage, helium service systems and the main control room in adjacent structures. The process steam interface is located outside the nuclear island, allowing the industrial steam consumer to be sited at a controlled standoff distance. The reference single-module plant is accommodated within a footprint of about 20,000 m² [TBC], and the layout is standardised to allow extension to a multi-module.

9. Testing Conducted for Design Verification and Validation

Design verification and validation are based on KAERI test facilities and on irradiation testing and post-irradiation examination of UCO TRISO fuel and nuclear-grade isotropic graphite. A consolidated test and analysis plan supporting the basic design and the construction permit application is under preparation, covering steam generator and helium circulator component qualification, and a performance test facility for the reactor cavity cooling system.

10.Design and Licensing Status

The design is currently in the Basic Design phase, following completion in 2025 of the conceptual design — the first high-temperature gas-cooled reactor conceptual design carried out independently by the Republic of Korea. Licensing is being pursued under the Korean regulatory framework towards a construction permit application, supported by comprehensive pre-application regulatory studies on topics such as functional containment and the inherent safety characteristics of HTGRs.

11.Fuel Cycle Approach

HECTAR uses an open, once-through fuel cycle based on UCO TRISO fuel compacts in hexagonal graphite blocks, enriched as HALEU to below 19.75 % U-235. Fresh fuel is supplied as fabricated blocks and the core is replaced on a 48-month interval, so that neither on-line refuelling nor on-site fuel fabrication is required.

12.Waste Management and Disposal Plan

Spent fuel blocks are held in the on-site interim storage facility for an initial cooling period and are then transferred to dry storage casks on site, pending national policy on the disposal of graphite-matrix spent fuel.

13.Development Milestones

Year	Milestone	Status
2025	Conceptual Design Complete	**Completed**
2031	Basic Design Complete	**Ongoing**
2033	Construction Start	**Planned**

JSC “Afrikantov OKBM”, Russian Federation HTGR



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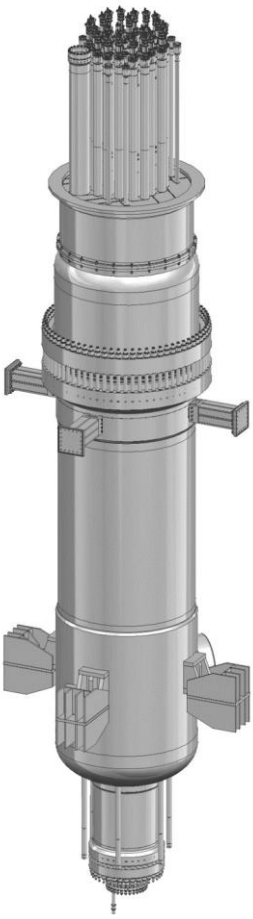


Figure 1: HTGR

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*JSC “Afrikantov OKBM”, Rosatom, Russian Federation*
Reactor type	*HTGR*
Coolant/moderator	*He / Graphite*
Design status	*Detailed Design*
Thermal/electrical capacity, MW(t)/MW(e)	*200 / -*
NSSS operating pressure (primary/secondary), MPa	*4.0 / 4.2*
Core inlet/outlet coolant temperature (°C)	*330 / 850*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*Prismatic graphite fuel blocks / Hexagonal*
Number of fuel assemblies in the core	*~ 370 Fuel blocks in core (FA)*
Fuel enrichment (%)	*< 20*
Core discharge burnup (GWd/ton)	-
Refuelling cycle (months)	*13 - 26*
Fuel cycle requirements/approach	-
Reactivity control mechanism	*Control and Shim Rods of CPS (Control and Protection System) drives*
Approach to safety systems	*Active / Passive*
Seismic design (SSE)	-
Design life (years)	*60*
Plant footprint (m^2)	-
RV height/diameter (m)	*< 17 / < 5.5*
RV weight (metric ton)	-

1. Introduction

The HTGR reactor plant has been developed based on the experience gained by Russian specialists during the development of MHR-T reactor, MHR-100 reactor plant and GT-MHR reactor plant.

The HTGR (200 MWt) is a high-temperature gas-cooled reactor to be used as a source of high-potential heat for industrial processes.

The HTGR reactor plant has been designed and validated since 2021 by order of Rosenergoatom JSC as part of a comprehensive R&D program on development of process solutions for a process power complex with the HTGR reactor plant and a chemical process facility for production of hydrogen and hydrogen-containing products.

2. Target Application

☐ Electricity	☒ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☐ District Heating / Steam	☐ Desalination	☒ >550°C

3. Design Philosophy

4. Main Design Features

(a) Nuclear Steam Supply System

High-potential heat is generated in the HTGR reactor core and transferred to the helium coolant of the primary circuit.

Then heat is transferred from the primary circuit coolant to the secondary circuit helium coolant and then to the steam-water circuit where superheated steam is generated and further transported to the chemical process facility installations.

In normal operation, one HTGR unit of the process power plant supplies superheated steam of the required parameters to one chemical process facility line.

(b) Reactor Core

The reactor core is located in the cavity formed by graphite stack, has a cylindrical configuration (the ratio of height to diameter is ~ 3.5) and is formed by prismatic hexagonal fuel assemblies (Figure 3), arranged into vertical columns.

The main structural material of the reactor core is reactor graphite.

The reactor core includes fuel assemblies, absorbing elements of the backup shutdown system, rods of the control and protection system (CPS), and neutron source.

The fuel assembly is a hexagonal (prismatic) graphite block designed for placement and spacing of fuel compacts and burnable absorber compacts, arranging the coolant circulation, heat transfer from fuel to coolant, and for placement of CPS absorber rods and absorber elements in the reactor core.



Figure 2: Prismatic hexagonal Fuel Assemblies (FA)

The reactor core operates with partial FA relocations in the shutdown reactor.

The HTGR design uses thermo- and radiation-resistant fuel based on micro-particles with ceramic coatings. The fuel composition is based on a fuel element with a UO2 kernel coated by multi layers of pyrocarbon of different density and silicon carbide (TRISO fuel). The micro-particles are uniformly distributed in a solid cylindrical fuel compact made of matrix graphite.

HTGR fuel design has been developed with a wide cooperation of industry and non-industry enterprises, academic science and universities. A HTGR fuel pilot production line has been constructed (design capacity of 250 thousand fuel compacts/year).

Based on R&D results, technological processes have been developed to produce basic materials, micro -particles and fuel compacts. It has been demonstrated that protective coatings retain high efficiency to burnups close to the design fuel burnup levels at normal and abnormal operation temperatures. Operability of the selected HTGR fuel design (also in extreme operation conditions at irradiation temperatures up to 1800 °C) has been confirmed.

(c) Reactivity Control

N/A

(d) Reactor Vessel (RV) and Internals

N/A

(e) Reactor Coolant System and Steam Generator

Primary Circuit

The primary circuit (Figure 3) includes the following main components:

- high-temperature gas-cooled reactor (HTGR);
- main circulation compressor (MCC);
- shutdown reactor cooling system.

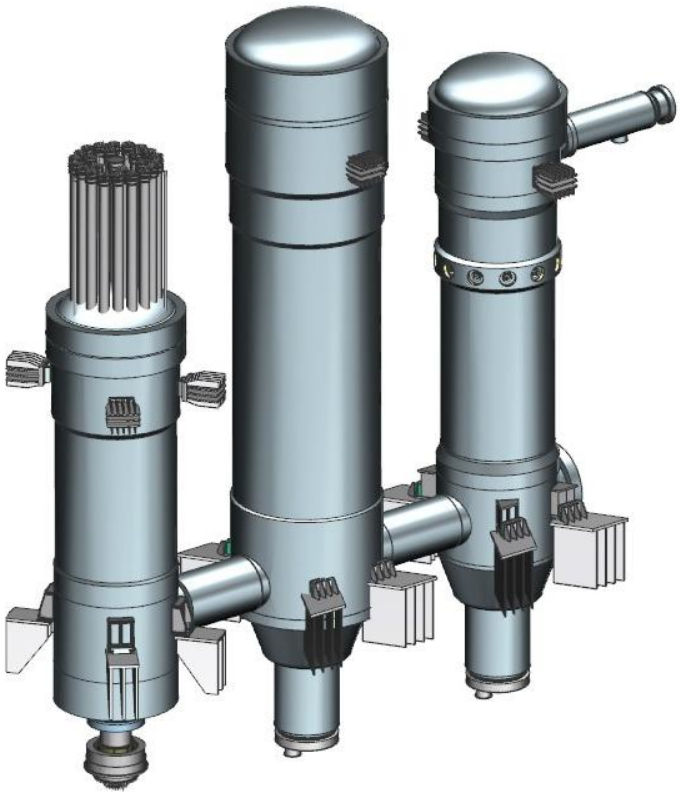


Figure 3: Primary circuit

In the primary circuit, coolant circulation is provided by the MCC as follows: "hot" helium of the primary circuit with a temperature of 850 °C is sent from the reactor to the high-temperature intermediate heat exchanger, where it transfers heat to helium of the secondary circuit, cooling to a temperature of 320 °C. Further, the flow enters the suction cavity of the MCC. From the MCC head, "cold" helium with a temperature of 330 °C and a pressure of 4.0 MPa (abs.) is sent to the reactor.

Secondary Side

The secondary circuit is a combination of the following components: a high-temperature intermediate heat exchanger, a connecting gas duct, a heat transfer unit and a circulation compressor of the secondary circuit.

The heat transfer unit is connected to feedwater pipelines and steam pipelines to transfer steam from the HTGR reactor plant.

(f) Pressuriser

N/A

(g) Primary pumps

N/A

(h) Distinguishing features

The design provides for a reactor cavity air cooling system designed to remove heat from the reactor, reactor support structures and reactor cavity structures during normal operation and abnormal operation conditions, including design basis accidents.

Individual components of the reactor plant are to be manufactured using carbon-carbon composite materials.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The safety of the process power complex as a whole is ensured in accordance with regulatory legal acts and regulations establishing the safety requirements for a nuclear facility and a hazardous production facility. The priority is to ensure nuclear and radiation safety of the nuclear process power plant with the HTGR.

The safety concept of the HTGR reactor plant is based on the principle of defense-in-depth and inherent safety features.

The high safety level of the HTGR reactor plant is mainly provided by effective inherent safety features which include natural negative feedbacks as well as specific processes and design characteristics, thermophysical, chemical and mechanical properties of the materials used, which ensure the stability of the reactor core and prevent fuel melting in any potential accidents at the nuclear process power plant with the HTGR.

The HTGR reactor plant inherent safety features provide for self-limitation of the reactor power, limitation of the primary circuit pressure, exclusion of fuel melting, preservation of the integrity of the primary circuit components, including the reactor vessel.

The nuclear process power plant with the HTGR is designed with specific safety systems which perform safety functions in case of design basis accidents, as well as special technical means for beyond-design-basis accidents management.

The combination of a high inherent safety of the HTGR reactor plant, passive safety systems and special technical means to manage beyond-design-basis accidents allows ensuring a high level of nuclear and radiation safety of the nuclear process power plant with the HTGR corresponding to Generation IV nuclear facilities.

The following is preliminarily confirmed:

- dose loads at the border of the nuclear process power plant with the HTGR under normal operation conditions do not exceed maximum value of 10 μSv/year that is within the quota established by national sanitary rules and standards (SanPiN 2.6.1.24-03) and within the dose limit established by national radiation safety standards (NRB 99/2009);
- dose loads at design basis accidents (8 μSv) do not require practical measures to limit population exposure according to national radiation safety standards NRB 99/2009;
- population exposure doses outside the nuclear process power plant with the HTGR at beyond-design-basis accidents are below the values established by regulatory documentation for radiation accidents and do not require population sheltering and evacuation.

No chemical process facility systems are involved to implement main safety functions of the nuclear process power plant with the HTGR (i.e. emergency reactor shutdown and reactor maintenance in subcritical state, emergency heat removal from the reactor, retention of radioactive substances within the established boundaries) and to manage beyond-design basis accidents.

Abnormal operation of the chemical process facility and impact on the parameters of the steam-water circuit of the nuclear process plant with the HTGR, as well as the most potentially dangerous failures of the chemical process facility and their consequences such as fires or explosions are considered as external impacts that shall not lead to failures of safety-related systems of the nuclear process power plant with the HTGR.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

The containment includes concrete structures with low pressure metal cladding.

In addition to the containment the HTGR reactor plant includes the following physical barriers, such as: fuel kernel, fuel particle coating, fuel compact, graphite block of fuel assemblies, sealed boundary of the primary coolant, biological shielding.

(o) Spent Fuel Cooling Safety Approach / System

The building with spent nuclear fuel and solid radioactive waste provides for its long-term storage at the process power plant site. Handling of spent nuclear fuel and graphite as solid radwaste includes the following processes:

- Conditioning of spent nuclear fuel and reflectors and their transfer to the spent nuclear fuel and solid radwaste handling unit;
- Putting into canisters in the handling unit;
- Moving to the storage.

6. Plant Safety and Operational Performances

The schematic diagram of the process power plant with the HTGR is shown in Figure 4.

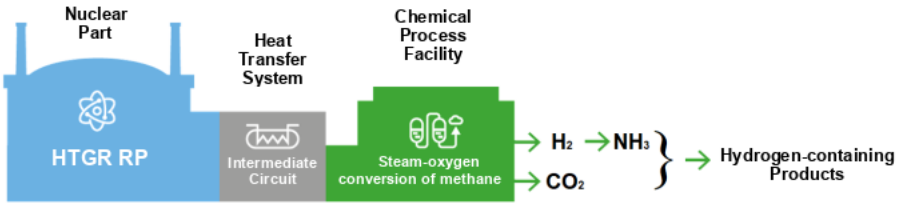


Figure 4: Schematic diagram of the process power plant with the HTGR.

The safety concept of the nuclear process power plant with the HTGR provides that radiation impact on the personnel of the chemical process facility during normal operation and abnormal operation conditions, including design basis accidents, does not lead to excess of established exposure doses, the values of which correspond to the dose limits for the population and do not require protective measures.

7. Instrumentation and Control Systems

An automated control system is provided in the KLT-40S 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. Plant Layout Arrangement

The process power complex with the HTGR reactor plant includes two facilities with process connections: a nuclear process power plant with the HTGR and a chemical process facility.

The nuclear process power plant with the HTGR is a nuclear facility which safety is regulated by federal rules and regulations in the field of atomic energy use, and Federal Law of 21.11.1995 No. 170-FZ "On the Use of Atomic Energy."

The boundaries of the nuclear process power plant with the HTGR are:

- a water pipeline from the water treatment system of the power process complex;
- secondary quick-acting valves located on the pipeline supplying steam to the chemical process facility;
- secondary quick-acting valves located on the pipeline supplying feedwater from the chemical process facility;
- devices for connection to the power transmission line to supply power to the process power plant with the HTGR.

The process power plant with the HTGR is designed to withstand the following external impacts:

- aircraft crash;
- design-basis earthquake;
- maximum design earthquake;
- air shock wave;

The chemical process facility is a hazardous production facility which safety is regulated by federal rules and regulations in the field of industrial safety of hazardous production facilities and Federal Law of 21.07.1997 No. 116-FZ "On Industrial Safety of Hazardous Production Facilities."

9. Testing Conducted for Design Verification and Validation

Irradiation experiment results confirmed the operability of pre-irradiated HTGR fuel (laboratory batches) at temperatures of up to 1800 °C.

10.Design and Licensing Status

The final design of the HTGR reactor plant has has been completed (thermal power: - 200 MW, helium temperature at the core outlet: 850 °C).

A HTGR fuel pilot production line has been constructed (design capacity of 250 thousand fuel compacts per year).

Irradiation experiment results confirmed the operability of pre-irradiated HTGR fuel (laboratory batches) at temperatures of up to 1800 °C.

Comprehensive R&D work is underway to develop and validate materials to be used (graphite, heat-resistant alloy, composite materials, etc.) and to optimize process solutions.

In 2030, it is planned to make a decision to start the investment phase of the project.

Estimated construction time:

- Development of design documentation is not more than 36 months from the date of th decision to start the investment phase of the project.
- The period of construction of the nuclear process power plant with the HTGR and chemica process facility and commissioning of a first-of-the-kind nuclear process power plant with the HTGR and chemical process facility is no more than 84 months from the date of issue of the construction permit.

11.Fuel Cycle Approach

N/A

12.Waste Management and Disposal Plan

N/A

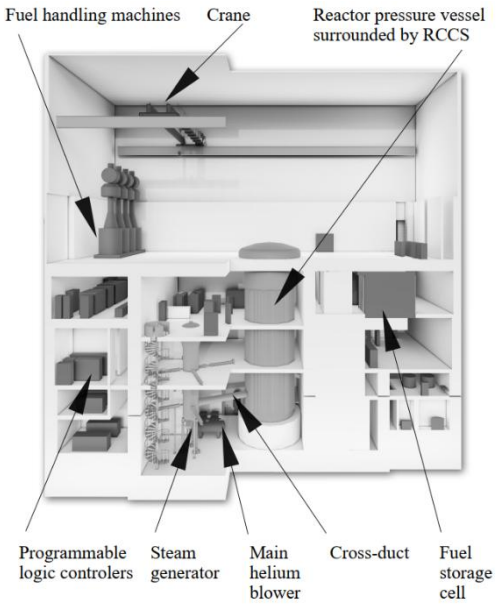
13.Development Milestones

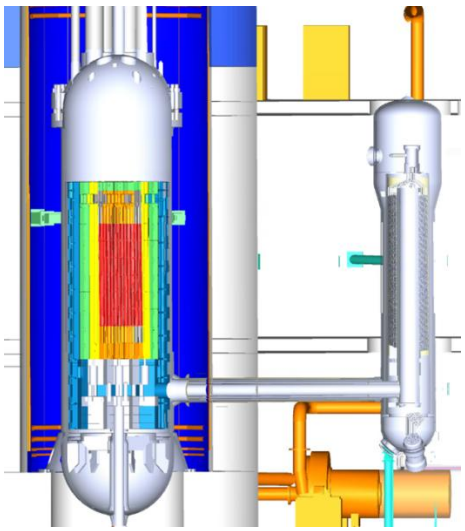
Year	Milestone	Status
2021	Start of activities	Completed
2021	HTGR RP preliminary design	Completed
2021	Production of laboratory samples of HTGR fuel, start of reactor testing of HTGR fuel laboratory samples	Completed
2023	Final design of HTGR coated fuel particle and fuel compact	Completed
2023	Final design of HTGR RP	Completed
2024	Development of HTGR pilot fuel manufacturing technology, manufacturing of pilot production line	Completed
2025	A pilot demonstration test facility for hydrogen production (capacity of up to 150 Nm3/h) was put into operation	Completed
2025	Completion of in-pile tests and post-irradiation studies of HTGR fuel laboratory samples (including tests of irradiated samples of HTGR fuel in modes which were typical of emergency ones)	Completed

NCJB, Poland
HTGR-POLA



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*National Centre for Nuclear Research (NCBJ), Poland*
Reactor type	*HTGR*
Coolant/moderator	*Helium / Graphite*
Design status	*Basic Design*
Thermal/electrical capacity, MW(t)/MW(e)	*30 / 11.5*
NSSS operating pressure (primary/secondary), MPa	*4.0 / 13.8*
Core inlet/outlet coolant temperature (°C)	*325 / 750*
Primary circulation	-
Fuel type/assembly array	*UO2 TRISO/ graphite prismatic blocks*
Number of fuel assemblies in the core	*114*
Fuel enrichment (%)	*8.9 and 9.1*
Core discharge burnup (GWd/ton)	*40*
Refuelling cycle (months)	*20*
Fuel cycle requirements/approach	-
Reactivity control mechanism	*B4C control rods, reserved B4C pellets*
Approach to safety systems	*Inherent safety, passive and active systems, RCCS ≥ 72h*
Seismic design (SSE)	*0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*25,000*
RV height/diameter (m)	*16 / 4.08*
RV weight (metric ton)	*200*

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 estimated a demand of 6.5GWt power and concluded that Poland should build a fleet of 30-40 industrial HTGRs with a thermal output of 180 MWt 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 heat for targeted practical applications.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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 MWe, superheated steam with a maximum output of 25 t/h and 2.9 MWe, and communal heat with a maximum output of 16.5 MWt. and 9.0 MWe.

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 MWt reactor design.

4. Main Design Features

(a) Nuclear Steam Supply System

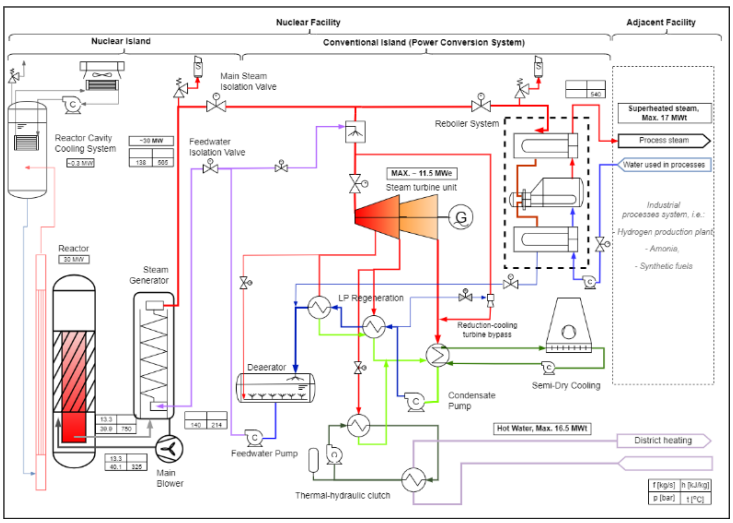


Figure 1. Simplified diagram of the HTGR-POLA NSSS

The NSSS consists of a reactor pressure vessel, steam generator, and helium blower. It operates at 30 MWt, with helium coolant transporting heat from the reactor core to the steam generator. The core's coolant temperature increases to 750℃, with a 4.0 MPa coolant pressure. Heat is transferred to the secondary circuit, generating superheated steam (565℃, 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.

(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 RSS). The active core height is 3.6 m, and its equivalent diameter is 1.66 m. TRISO fuel with HALEU kernels enriched in U-235 is used, with fuel compacts having seven packing-factor levels (13.0%, 14.8%, 15.5%, 16.5%, 17.5%, 18.0%, and 19.0%) inserted into dedicated fuel 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.

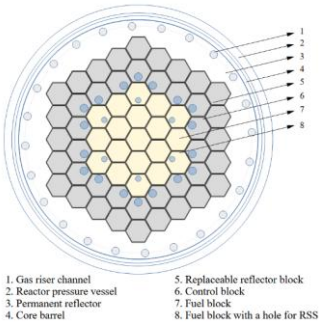


Figure 2. Reactor core configuration

(c) Reactivity Control

The reactor design uses two independent reactivity control systems: Control Rods (CR) and Reserve Shutdown System (RSS). 18 CR, using B4C as neutron poison, are used for active operational control. If a CR becomes inoperable, the RSS is activated. The RSS uses cylindrical pellets (B4C/C) released from hoppers into six channels within the active core. Burnable poison rods (B4C/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 Vessel (RV) 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 lower mixing chamber and passes through the rising channels in the permanent reflector to 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.

(h) Distinguishing features

5. Safety Features

(i) 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. The Reactor Cavity Cooling System (RCCS) is a redundant system that operates actively during normal plant operation

and passively under accident conditions, providing reliable reactor cavity cooling and decay heat removal without the need for operator actions or external power.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

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.

(m) Safety Approach and Configuration to Manage DEC

(n) 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.

(o) 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. The final report for this stage, titled 'Technical Description of the Research High-Temperature Gas-Cooled Nuclear Reactor' received official ministerial approval on 20 May 2025 In 2025, NCBJ conducted a technical dialogue with the National Atomic Energy Agency (PAA) on the HTGR-POLA project to present reactor design information and support early regulatory engagement, with the objective of facilitating alignment with national licensing requirements ahead of the formal application process. The subsequent phase includes continued development of the Preliminary Safety Analysis Report (PSAR) and associated safety documentation, as well as preparatory activities supporting the licensing process, including site-related investigations. In parallel, advancement of the detailed design work is planned to proceed.

11.Fuel Cycle Approach

The HTGR-POLA is designed for open fuel cycle with TRISO fuel. The primary fuel enrichment is 8.9% and 9.1% of 235U, the maximum burnup is currently limited to 37 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.

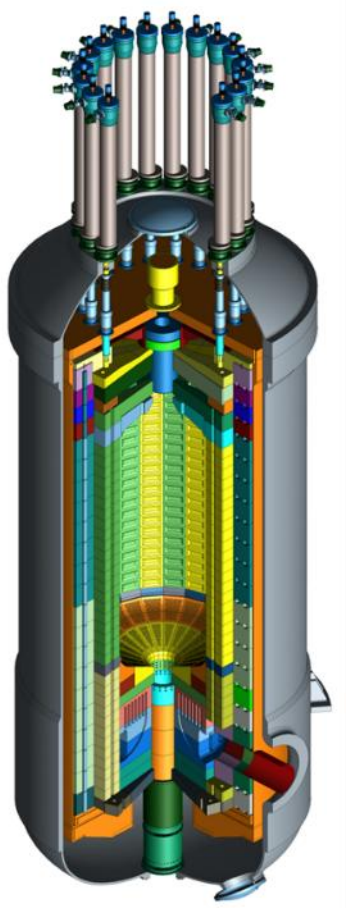
13.Development Milestones

Year	Milestone	Status
2012-2019	Preliminary studies and technological innovation (using available materials).	**Completed**
2019-2021	Pre-conceptual design phase and technology validation	**Completed**
2021-2022	Conceptual Design Phase	**Completed**
2022-2024	Basic Design Phase	**Completed**
2025	Technical dialogue with the National Atomic Energy Agency (PAA)	**Completed**
2026-2028	Detailed Design Phase	**Ongoing**
2029-2030	Licensing Phase	**Planned**
2031-2034	Construction	**Planned**
2035	Commissioning	**Planned**

STL Nuclear (Pty) Ltd., South Africa HTMR100



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*STL Nuclear (Pty) Ltd, South Africa*
Reactor type	*HTGR*
Coolant/moderator	*Helium, graphite*
Design status	*Basic design*
Thermal/electrical capacity, MW(t)/MW(e)	*100 / 35 single module plant, 1000 / 350 multi module plant*
NSSS operating pressure (primary/secondary), MPa	*4 / 0.4 / 16*
Core inlet/outlet coolant temperature (°C)	*300 / 650*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*TRISO particles in pebbles; LEU/Th*
Number of fuel assemblies in the core	*~160 000; around 125 to 150 pebbles/day throughput*
Fuel enrichment (%)	*Between 9.95% and 19.95%*
Core discharge burnup (GWd/ton)	*80-90*
Refuelling cycle (months)	*Online fuel loading and multi-pass recirculation*
Fuel cycle requirements/approach	*Various options (see below)*
Reactivity control mechanism	*Control Rods in the reflector*
Approach to safety systems	*Passive*
Seismic design (SSE)	*0.3 g (generic site), 0.5 g under consideration)*
Design life (years)	*40*
Plant footprint (m^2)	*5 000 (buildings only)*
RV height/diameter (m)	*15.7 / 5.6 (flange outer diameter)*
RV weight (metric ton)	*155*

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.

The 1000 MWt 10 module configuration represents a high-value generation asset that bridges the gap between traditional baseload nuclear power and the dynamic, flexible energy delivery required by future grids. By prioritizing system redundancy, leveraging proven Concentrated Solar Power (CSP) thermal storage technology, and utilizing

advanced Printed Circuit Heat Exchanger (PCHE) interfaces, the proposed configuration provides a path toward sustainable, dispatchable, and economically competitive clean power generation.

2. Target Application

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☒ <200°C
☒ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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 intermediate salt loop is designed to provide firm, dispatchable electricity through TES thermal buffering between the nuclear island and two Rankine trains and is ideal to integrate with variable renewable energy systems. 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 and allows flexibility in protection against external events.

The 10 module configuration can deliver 1000 MWt integrated with a 16-hour molten salt storage system that creates a "thermal buffer" that fundamentally decouples steady-state nuclear heat production from the variable demands of the electricity market. This allows the nuclear islands to operate at a consistent 100% capacity factor, maximizing the return on the nuclear capital investment, while the power block follows daily demand profiles.

4. Main Design Features

(a) Nuclear Steam Supply System

The helium coolant from each of the ten reactor modules exits at 650°C and passes through high-efficiency PCHEs. The heated molten salt (60% $NaNO_3$, 40% KNO_3) is collected into a common manifold. A portion of the salt flow is directed to the 16 000 MWhth TES (Hot Tank), which acts as a buffer. From the hot salt reservoir, two parallel heat salt-to-steam generators feed the dual Rankine cycle power trains. This structural design ensures that the system maintains a high-capacity factor and superior grid response capability.

(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) Reactivity Control

Eighteen neutron absorber rods are provided in graphite sleeves inside the graphite side reflector blocks. The absorber rods can be operated independently as a group or as sub-groups, as required by the reactor operating control system. A control rod consists of several rod absorber material segments, pinned together to form articulating joints. The segments consist of sintered B_4C absorber material, sandwiched between an inner and an outer tube segment. The inner tube segment allows cooling helium gas to flow fromthe 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.

(d) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

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 PCHE. Electric blowers are used to move the helium within the reactor pressure boundary. The PCHE acts as intermediate heat exchanger and transfers thermal energy to the secondary solar salt loop, where redundant energy is stored in salt TES tanks. The hot salt tanks feed the salt-to-steam generators, which converts the thermal energy to high quality steam in the tertiary loop which in turn drives the steam turbine. Both the molten salt loop and steam cycle utilise conventional, commercially proven CSP technology.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

(i) 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 introducting the secondary salt loop which acts as barrier between the primary helium loop and tertiary water/steam loop. The secondary salt loop is also operated at lower pressure than both the helium and steam loops, which prevents ingress of salt into the helium loop and steam cycle. The secondary salt loops effectively act as a neutronic decoupler between the transients of the steam cycle and reactor reactivity control, thereby enabling optimal and safe reactor operation amidst variable grid demands.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

The HTMR100 manages Design Basis Conditions (DBC) - such as a loss of coolant or heat sink - relying heavily on inherent physical laws and passive features. It uses a low power density (3.6 MW/m³), a negative temperature

reactivity coefficient, and TRISO particle fuel that safely retains fission products below 1600°C without active system action.

(m) Safety Approach and Configuration to Manage DEC

The HTMR-100 manages Design Extension Conditions (DEC) through an inherent and passive safety approach. Its configuration relies on low power density, TRISO ceramic-coated particle fuel, a massive graphite moderator, and an uninsulated reactor vessel that radiates decay heat directly to a passive cavity cooling system, preventing core damage without operator action or active pumps.

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 extension conditions.

(n) 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.

(o) 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.

6. Plant Safety and Operational Performances

The 1000 MWt 10 module HTMR100-hybrid system serves as a versatile bridge between traditional nuclear baseload and the flexibility required by modern grids. By using a 16-hour molten salt "thermal buffer," the system decouples constant nuclear heat production from fluctuating market demand. This allows the reactors to maintain a 100% capacity factor—optimizing capital returns—while the power block independently follows daily load profiles. The system can also be adapted to accept electrical energy inputs from external variable renewable energy sources via the TES system, further enhancing its flexibility to serve a volatile grid demand.

7. Instrumentation and Control Systems

The Automation System (ATS) comprises that group of safety and non-safety C&I systems that provide automated protection, control, monitoring and human-system interfaces. The three specific systems in the HTMR100 system structure that 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 PCHE intermediate heat exchanger 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.

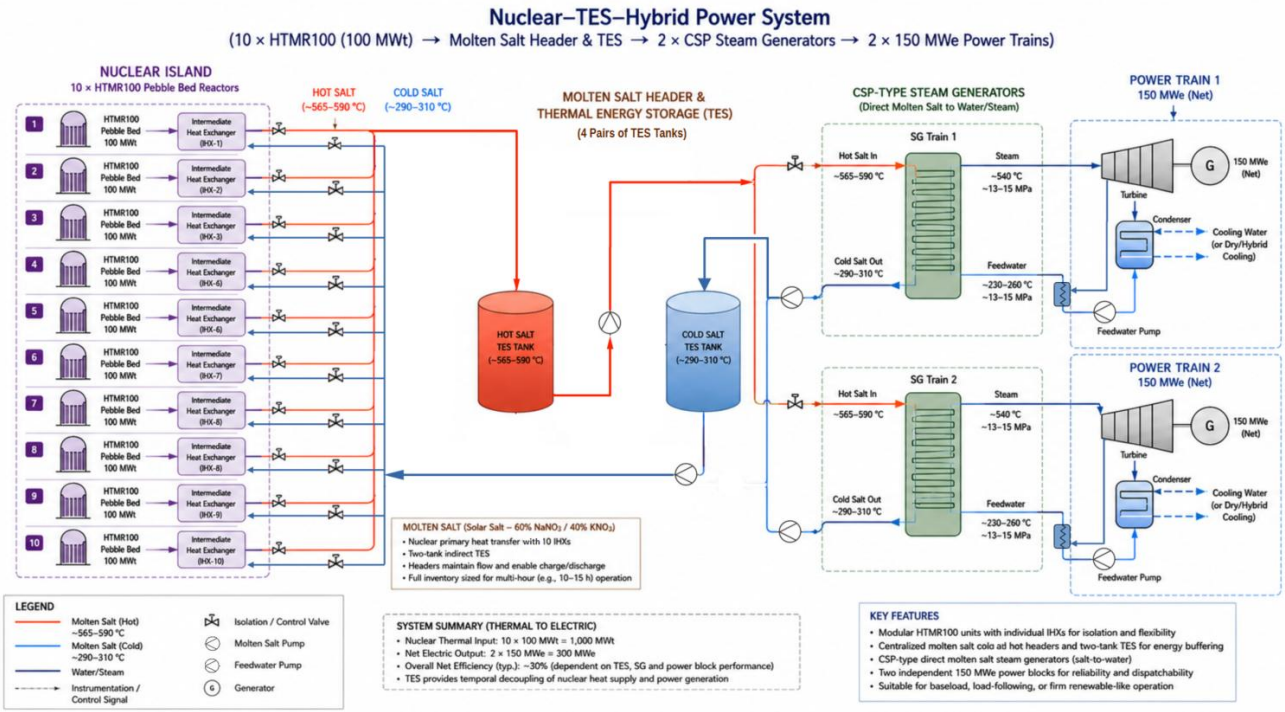


Figure 1: 10 HTMR100 Module configuration

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 Hi-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 Hi-cask to a low energy spent fuel concrete cask (Low-Cask); iv) The Low-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 unloadin g machine (failed fuel separator), may be required in the lifetime of the core. During 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.

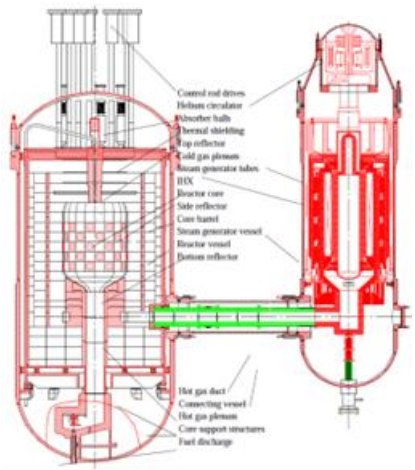
13.Development Milestones

Year	Milestone	Status
2012	Project Started	**Completed**
2020	Preparation for pre-licensing application	**Completed**
2021	Conceptual design	**Completed**
2022	Basic design started	**Ongoing**
2028	Detailed design completed	**Planned**
By 2030	First concrete poured	**Planned**

Tsinghua University, China
HTR-10



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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*
Design status	*Operational*
Thermal/electrical capacity, MW(t)/MW(e)	*10 / 2.5*
NSSS operating pressure (primary/secondary), MPa	*3 / 4*
Core inlet/outlet coolant temperature (°C)	*250 / 700*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*Spherical elements with TRISO particles fuel (UO2 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*
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.*
Reactivity control mechanism	*Control rod insertion/ negative temperature feedback*
Approach to safety systems	*Combined active and passive*
Seismic design (SSE)	*3.3 m/s2*
Design life (years)	*20 (test reactor)*
Plant footprint (m^2)	*100x130*
RV height/diameter (m)	*11.1 / 4.2*
RV weight (metric ton)	*167*

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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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

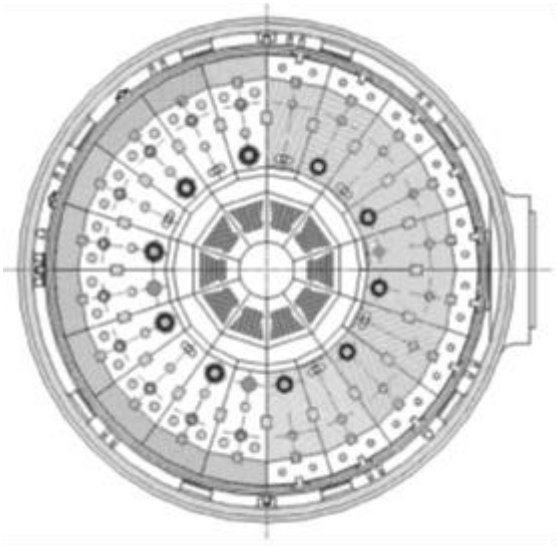
The primary pressure boundary consists of reactor pressure vessel, steam generator pressure vessel and hot gas duct pressure vessel which connects the above two vessels. This arrangement can make the maintenance and inspection of the components easier and mitigate the accident result of water ingress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

4. Main Design Features

(a) Nuclear Steam Supply System

(b) Reactor Core

The reactor core volume is 5m3, 1.8 m in diameter and the mean height is 1.97 m. About 27 000 spherical fuel elements with 60 mm in diameter will be filled up in the reactor core, the enrichment of fuel is 17% and the mean discharge burn up is designed to be 80 000 MWd/tU. The reactor core is entirely constructed by graphite materials, no metallic components are used in the region of the core. At the funnel bottom of the reactor core, there is a fuel-element discharging tube with a diameter of 500 mm and a length of 3.3 m. At the tube end the special fuel discharge facility singularise the fuel to be unloaded through a 65 mm diameter pipe that penetrates the reactor pressure vessel.



(c) 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.

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 (B4C) is used as the neutron absorber. Each control rod contains five B4C 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 B4C 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.

(d) Reactor Vessel (RV) 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.

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.

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 m3.

(e) Reactor Coolant System and 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.

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.

(f) Pressuriser

(g) Primary pumps

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.

(h) Distinguishing features

To verify and demonstrate the technical and safety features; and to establish an experimental base for process heat applications

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.

(i) Engineered Safety System Approach and Configuration

(j) 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.

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

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.

(o) Spent Fuel Cooling Safety Approach / System

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 m2. 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. Testing Conducted for Design Verification and Validation

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.

10.Design and Licensing Status

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

11.Fuel Cycle Approach

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

12.Waste Management and Disposal Plan

To be included in the national plan of test facilities.

13.Development Milestones

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

JAEA, Japan
HTR-50S



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*JAEA*
Reactor type	*Prismatic HTGR*
Coolant/moderator	*Helium / graphite*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*50/17.2 (max)*
NSSS operating pressure (primary/secondary), MPa	*4 (abs)/12.5 (abs)*
Core inlet/outlet coolant temperature (°C)	*Phase I: 325 / 750*
Primary circulation	*Phase II: 325 / 900*
Fuel type/assembly array	*Forced by gas circulator*
Number of fuel assemblies in the core	*UO2 TRISO ceramic coated particle*
Fuel enrichment (%)	*33 fuel element / fuel block*
Core discharge burnup (GWd/ton)	*5.9 / 9.4*
Refuelling cycle (months)	*49*
Fuel cycle requirements/approach	*730 days*
Reactivity control mechanism	*N/A*
Approach to safety systems	*Control rod insertion*
Seismic design (SSE)	*Active and passive*
Design life (years)	*TBD*
Plant footprint (m^2)	*40*
RV height/diameter (m)	*TBD*
RV weight (metric ton)	*13.8*

1. Introduction

HTR50S is 50 MWth 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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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 Vessel (RV) 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 and Steam Generator

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.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) 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 conditions and the analysis results demonstrated that evaluation items for the safety analysis met the evaluation criteria.

(m) 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 characteristics 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 extension conditions which considers failures of engineered safety features.

(n) Containment System

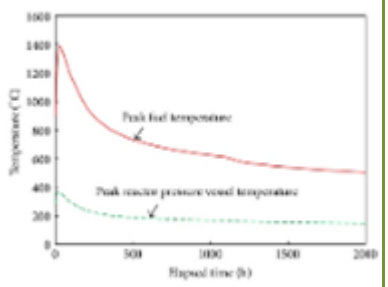
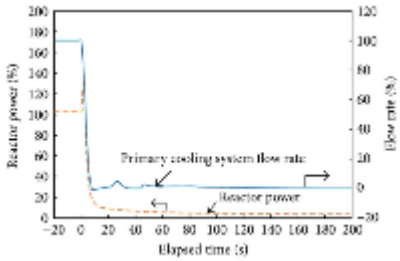
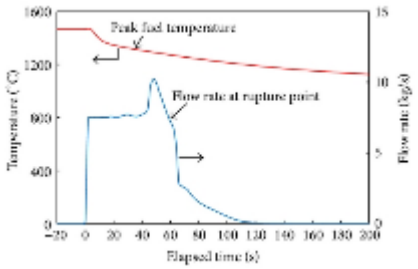
The reactor containment vessel is one of the engineering safety facilities in the HTR50S, which is designed to prevent the release of radioactive materials to the outside and to ensure the safety of the general public and workers in the vicinity of the reactor facility. Isolation valves are installed 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.

(o) 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.

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 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.

		
Depressurised accident behavior	SG tube rupture accident behavior Reactor power, reactor coolant flow rate	SG tube rupture accident behavior Peak fuel temperature, steam flow rate at rupture point

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



Figure 1. Bird' eye view

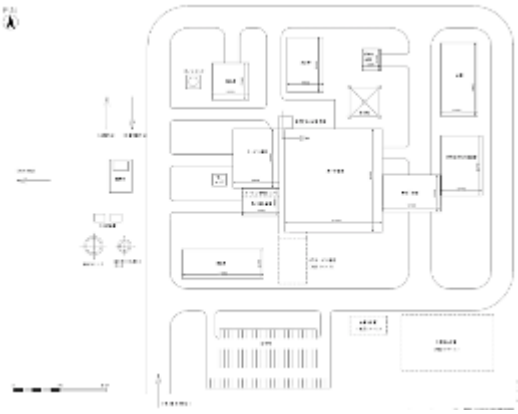


Figure 2. Site layout

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.

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.

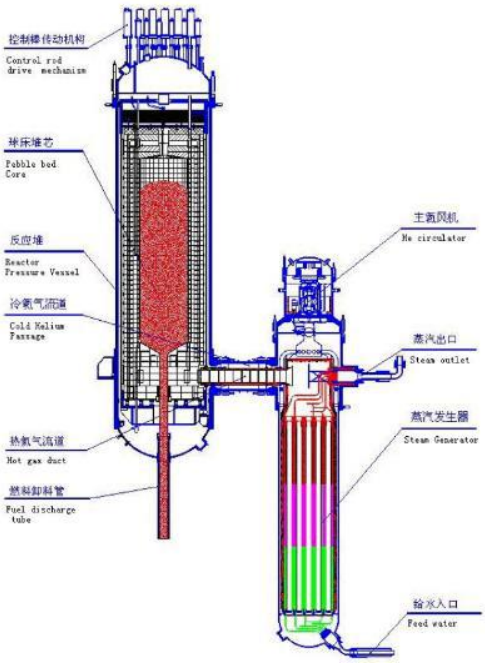
13.Development Milestones

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

Tsinghua University, China
HTR-PM



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*INET, Tsinghua University, People's Republic of China*
Reactor type	*Modular pebble bed HTGR*
Coolant/moderator	*Helium/graphite*
Design status	*In operation*
Thermal/electrical capacity, MW(t)/MW(e)	*2 × 250 / 210*
NSSS operating pressure (primary/secondary), MPa	*7 / 13.25*
Core inlet/outlet coolant temperature (°C)	*250 / 750*
Primary circulation	*Forced circulation*
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*
Fuel cycle requirements/approach	*LEU, open cycle, spent fuel intermediate storage at the plant*
Reactivity control mechanism	*Control rod insertion*
Approach to safety systems	*Combined active and passive*
Seismic design (SSE)	*0.2 g*
Design life (years)	*40*
Plant footprint (m^2)	*256 100*
RV height/diameter (m)	*25 / 5.7 (inner)*
RV weight (metric ton)	*800*

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

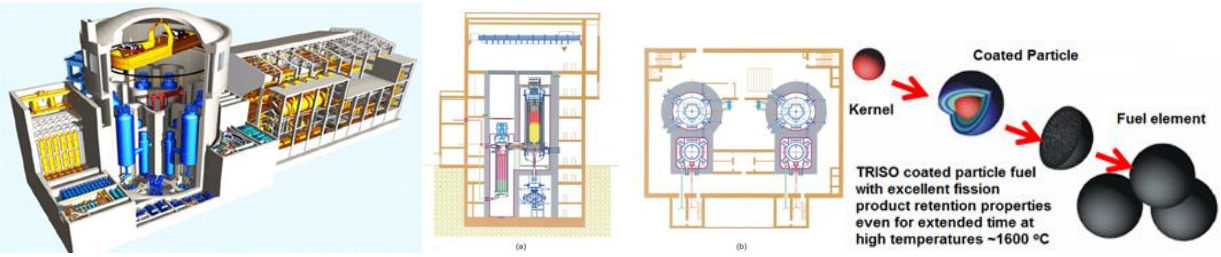
☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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) Nuclear Steam Supply System

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) Reactor Core

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 235U. 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.

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.

(c) 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 B4C.

(d) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

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.

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.

(f) Pressuriser

N/A

(g) Primary pumps

N/A

(h) Distinguishing features

Inherent safety, no need for offsite emergency measures

5. Safety Features

(i) Engineered Safety System Approach and Configuration

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.

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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

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.

(o) Spent Fuel Cooling Safety Approach / System

6. Plant Safety and Operational Performances

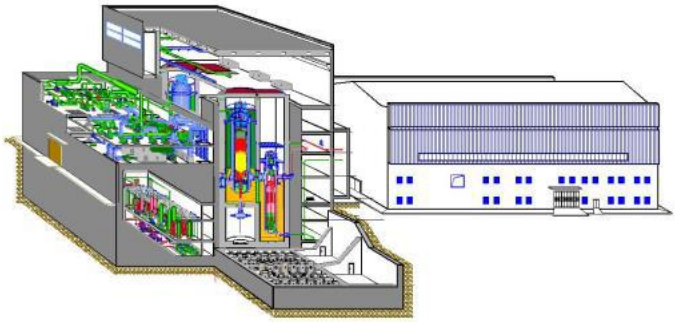
The HTR-PM demonstration power plant is under final commissioning test phase. Due to online fuel loading mode of HTR-PM, better availability factor can be expected compared with other power plants operating in a mode of periodic fuel loading.

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.



HTR-PM plant layout

9. Testing Conducted for Design Verification and Validation

N/A

10.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

11.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).

12.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.

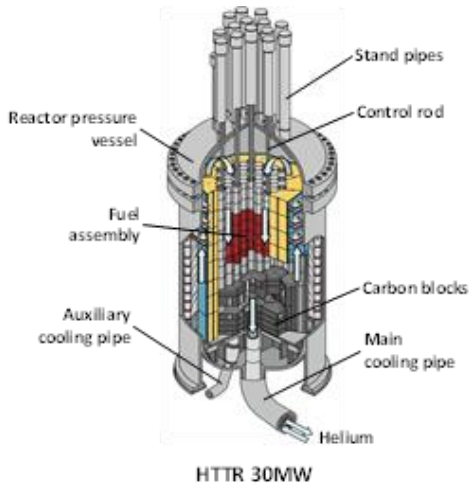
13.Development Milestones

Year	Milestone	Status
2001	Launch of commercial HTR-PM project	Completed
2004	Standard design of HTR-PM started	Completed
2006	HTR-PM demonstration power plant approved as one of National Science and Technology Major Projects	Completed

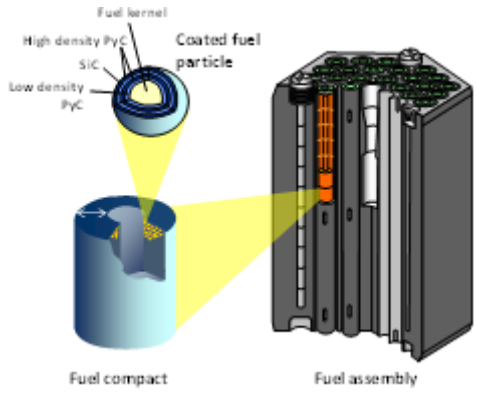
Year	Milestone	Status
2006	Huaneng Shandong Shidaowan Nuclear Power Co., Ltd, the owner of the HTR-PM, established by the China Huaneng Group, the China Nuclear Engineering Group Co. and Tsinghua University	Completed
2006-2008	Basic design of HTR-PM completed	Completed
2009	Assessment of HTR-PM PSAR completed	Completed
2012	First Pour of Concrete of HTR-PM	Completed
2013	Fuel plant construction started	Completed
2014	Qualification irradiation tests of fuel elements completed	Completed
2015	Civil work of reactor building finished	Completed
2016	RPV and core barrel etc. delivered, installation of main components ongoing	Completed
2017-Q42020	Fuel plant achieved expected production capacity	Completed
2021.8	Startup commissioning test of primary circuit	Completed
2021.9	Operation License	Completed
2021.12.20	First criticality	Completed
2022	Grid connection	Ongoing

JAEA, Japan
HTTR

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HTTR 30MW



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*JAEA in cooperation with MHI, Toshiba, IHI, Hitachi, Fuji Electric, NFI, Toyo Tanso, Japan*
Reactor type	*Prismatic HTGR*
Coolant/moderator	*Helium / graphite*
Design status	*In operation*
Thermal/electrical capacity, MW(t)/MW(e)	*30*
NSSS operating pressure (primary/secondary), MPa	*4*
Core inlet/outlet coolant temperature (°C)	*395 / 850 (950 max.)*
Primary circulation	*Forced by gas circulators*
Fuel type/assembly array	*UO2 TRISO ceramic coated particle*
Number of fuel assemblies in the core	*150*
Fuel enrichment (%)	*3 – 10 (6 avg.)*
Core discharge burnup (GWd/ton)	*22 (33 max.)*
Refuelling cycle (months)	*660 EFPD*
Fuel cycle requirements/approach	*N/A*
Reactivity control mechanism	*Control rod insertion*
Approach to safety systems	*Active*
Seismic design (SSE)	*> 0.7m/s2 automatic shutdown*
Design life (years)	*~20 (Operation time)*
Plant footprint (m^2)	*~200m × 300m*
RV height/diameter (m)	*13.2 / 5.5*
RV weight (metric ton)	*N/A*

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 Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

The objectives of HTTR are to:

- Establish and upgrade the technological basis for the advanced HTGR;
- Perform innovative basic research in the field of high temperature engineering; and
- 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.

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.



Cutaway view of the HTTR

4. Main Design Features

(a) Nuclear Steam Supply System

(b) Reactor Core

The HTTR employs the TRISO (Tri-structural isotropic)-coated fuel particles (CFPs) with UO2 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.

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

(c) Reactivity Control

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 B4C. 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 B4C/C pellet. When the RSS is activated, the hopper is opened and the B4C/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) Reactor Vessel (RV) and Internals

(e) Reactor Coolant System and Steam Generator

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

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.

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 CFPs of HTTR have excellent heat-resistant property which can bear very high temperature condition over 2200℃ 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;

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

(i) Engineered Safety System Approach and Configuration

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

(o) Spent Fuel Cooling Safety Approach / System

6. Plant Safety and Operational Performances

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. Testing Conducted for Design Verification and Validation

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

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.

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.

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.

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.

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.

10.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.

11.Fuel Cycle Approach

12.Waste Management and Disposal Plan

13.Development Milestones

Year	Milestone	Status
1969-1984	Conceptual design (4 years); System integrity design (6 years) and Basic design (3 years)	Completed
1985-1990	Detail design (3 years); Application and permission of construction (1 year)	Completed
2001-2002	Construction	Completed
2004	First criticality	Completed
2007-2010	Reactor outlet coolant temperature of 850oC; Safety demonstration test	Completed
2014	Reactor outlet coolant temperature of 950oC	Completed
2020	850oC/30 days operation; 950oC/50 days operation and Safety demonstration tests	Completed
2021	Conformity review on the New Regulatory Requirements start toward resume operation	Completed
2022-2024	Permission toward the restart of HTTR	Completed

JIMMY ENERGY SAS, France
JIMMY



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*JIMMY ENERGY SAS, France*
Reactor type	*HTGR*
Coolant/moderator	*Helium / Graphite*
Design status	*Detailed Design*
Thermal/electrical capacity, MW(t)/MW(e)	*60 / N/A*
NSSS operating pressure (primary/secondary), MPa	*1.5 / 3.0*
Core inlet/outlet coolant temperature (°C)	*300 / 700*
Primary circulation	*Forced*
Fuel type/assembly array	*UCO TRISO particles packed into pellets*
Number of fuel assemblies in the core	*N/A*
Fuel enrichment (%)	*9.8 - 19.75*
Core discharge burnup (GWd/ton)	*110*
Refuelling cycle (months)	*Long-life core: no on-site refuelling during operation*
Fuel cycle requirements/approach	*Refueling every 5 to 10 years*
Reactivity control mechanism	*B4C control rods and B4C neutronic poisons*
Approach to safety systems	*Full passive*
Seismic design (SSE)	*Zone 4*
Design life (years)	*10 to 20*
Plant footprint (m^2)	*6000 (entire installation)*
RV height/diameter (m)	*5/5*
RV weight (metric ton)	*15*

1. Introduction

Jimmy is a high-temperature gas-cooled reactor with a thermal power of 60MW(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 CO2 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

☐ Electricity	☒ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☒ <200°C
☐ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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 CO2 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) Nuclear Steam Supply System

The NSSS consists of a helium-cooled primary circuit transferring heat to a pressurized CO2 secondary circuit through a primary heat exchanger. The reactor is designed for heat supply only and does not include an electricity conversion system.

(b) Reactor Core

The core is composed of fuel assemblies located within a graphite reflector. Fuel assemblies are intended to be assembled in a factory and transported using existing fuel transportation package concepts. The fuel uses HALEU UCO TRISO particles packed into pellets and designed within the AGR programme envelope to achieve the targeted burnup.

(c) Reactivity Control

Reactivity is controlled by independent B4C control rods, complemented by B4C neutronic poisons crossing the core or the reflector. The approach supports independent and redundant shutdown capability.

(d) Reactor Vessel (RV) and Internals

The reactor vessel is 5x5m and relies on operating experience from HTGRs. Jimmy has announced a strategic partnership with Onet Technologies, a subsidiary of the French group Onet, for the production of its reactor vessel.

(e) Reactor Coolant System and Steam Generator

The primary coolant system is a pressurized helium circuit with forced circulation provided by a helium blower on the cold branch. The circuit divides and reunites before and after the core pressure tubes. The secondary CO2 circuit is connected to the primary circuit through a plate heat exchanger and transfers heat to the industrial customer through a site-specific secondary exchanger. Jimmy does not directly produce steam; steam, if required, is produced on the customer side using the delivered heat.

(f) Pressuriser

Not applicable in the sense of a water-cooled pressuriser. The primary system is a helium circuit operated at nominal pressure conditions.

(g) Primary pumps

Primary circulation is forced by a helium blower. The secondary CO2 loop circulation is also forced.

(h) Distinguishing features

Distinguishing features include a compact and low-weight design, simplified manufacturing and assembly, flexible operation for industrial heat users, intrinsic passive safety, and reliance on technologies with prior industrial experience.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

Jimmy safety relies primarily on HTGR intrinsic characteristics: passive decay heat removal, robust TRISO fuel particles and favourable reactivity coefficients. Engineered safety functions emphasize redundancy and high independence of control rods, with passive behaviour for key safety demonstrations.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

For Design Basis Conditions, the design approach is based on maintaining fuel integrity and removing decay heat without reliance on complex active safety systems. In normal operation, heat is removed through the industrial heat sink. In accidental conditions, the graphite core provides thermal inertia and the low power density supports passive cooling while keeping fuel temperatures below design limits.

(m) Safety Approach and Configuration to Manage DEC

For Design Extension Conditions, the safety approach continues to rely on the same passive features: TRISO retention capability, graphite thermal inertia, low power density and negative reactivity feedback. The objective is to limit the potential for significant radioactive releases through a defence-in-depth strategy.

(n) Containment System

The containment concept follows defence in depth and includes multiple barriers: TRISO particle coating layers, the graphite core, the primary system and the building containment. The building configuration also contributes to protection against external hazards and security-related aggressions.

(o) Spent Fuel Cooling Safety Approach / System

Spent fuel cooling is passive. Analyses indicate that spent fuel from the Jimmy reactor does not require temporary on-site storage in a spent fuel pool. At the end of the operating period, the core/module is removed and replaced, avoiding on-site refuelling during operation.

6. Plant Safety and Operational Performances

The design objective is to achieve a large radioactivity release frequency below 1E-7 per reactor-year. The generator is intended for reliable industrial heat supply, with flexible integration of one or several units according to site heat demand.

7. Instrumentation and Control Systems

The instrumentation and control system is under design. It is intended to follow conventional nuclear safety principles, including appropriate independence, redundancy and monitoring of key reactor and heat transport parameters. No major disruptive I&C technology is planned.

8. Plant Layout Arrangement



The plant layout is based on a compact building of approximately 30m x 30m x 30m, with an overall installation footprint of about 6000 m2. Thick walls support protection against external aggressions. Few systems penetrate the building boundary, mainly the reactor cavity cooling system and the secondary system connecting to the industrial heat line. Maintenance and installation access is mainly from the roof.

9. Testing Conducted for Design Verification and Validation

Jimmy philosophy is to rely on technologies that have already been tested or industrialized. Verification and validation activities therefore focus on confirming the behaviour of well-modelled components and systems rather than on a broad technology demonstration programme. Validation tests are planned or performed as required to support detailed design, qualification and licensing.

10.Design and Licensing Status

- FOAK:
  - Safety options reviewed in 2022 by the French safety authority
  - Site permit for FOAK plant under review by the French Safety Authority
- Fuel assembly plant
  - Site permit under review by the French Safety Authority
- Industrial platform (non-nuclear)
  - Building permit and environmental authorizations obtained

11.Fuel Cycle Approach

The fuel cycle approach is designed to simplify operation and maximize safety. On-site refuelling is planned during the operating life every 5 to 10 years. Jimmy also 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 radioactive waste during normal operation. Potential operational waste is limited, with helium loop leaks filtered and stored if they occur. At end of life, radioactive waste is collected and disposed of in accordance with applicable regulations, relying on existing French waste management infrastructure with minimal adaptations.

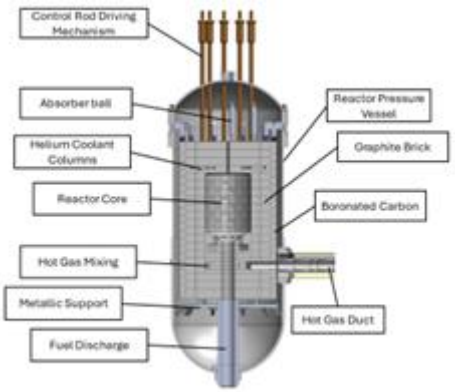
13.Development Milestones

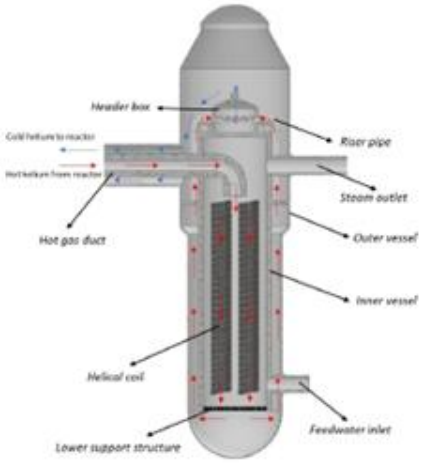
Year	Milestone	Status
2021	Company creation.	**Completed**
2022	Two successive funding rounds: €2.2m followed by €15m. basic design completed and peer reviewed. First industrial client feasibility study completed.	**Completed**
2023	Jimmy becomes the largest recipient of the France 2030 “innovative nuclear reactors” call for projects, securing record State support of €32m. first long-lead item orders placed.	**Completed**
2024	Site permit application filed for FOAK plant. Le Creusot industrial platform announced	**Completed**
2025	In-depth technical & cost review for FOAK plant. Site permit application filed for assembly plant. Jimmy obtains the building permit to establish its 12-hectare industrial platform in Le Creusot (Saône-et-Loire, France).	**Ongoing**
2026 to date	Jimmy is again selected by France 2030, as part of Phase 2 of the “innovative nuclear reactors” programme, securing €40m. The company also announces a €40m fundraising round and a contract with Onet Technologies for the manufacturing of its reactor vessel	**Planned**

BRIN - ITB, Indonesia
PeLUIt-40



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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*
Design status	*Basic design*
Thermal/electrical capacity, MW(t)/MW(e)	*30MWt / 10MWe*
NSSS operating pressure (primary/secondary), MPa	*3 / 6*
Core inlet/outlet coolant temperature (°C)	*250 / 750*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*Spherical elements with coated particle fuel*
Number of fuel assemblies in the core	*27 000*
Fuel enrichment (%)	*17*
Core discharge burnup (GWd/ton)	*80*
Refuelling cycle (months)	*On-line refuelling*
Fuel cycle requirements/approach	*LEU, open cycle, spent fuel intermediate storage at the plant*
Reactivity control mechanism	*Control rod and small absorber sphere*
Approach to safety systems	*Combined active and passive*
Seismic design (SSE)	*0.26g*
Design life (years)	*40*
Plant footprint (m^2)	*~24 000*
RV height/diameter (m)	*11.1 / 4.2 (inner)*
RV weight (metric ton)	*180*

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.

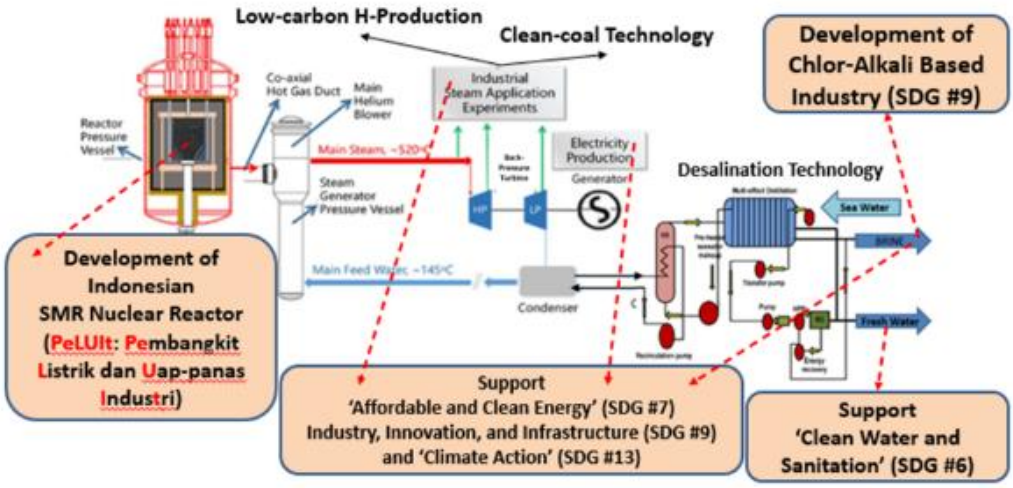


Figure 1: PeLUIt Schematic System and Targeted Impact on Sustainable Development Goals

2. Target Application

☐ Electricity	☒ Hydrogen	☐ Industrial process heat
☐ Baseload	☒HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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 Figure 1 of PeLUIt.

3. 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.

4. Main Design Features

(a) Nuclear Steam Supply System

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.

(b) Reactor Core

Fuel Characteristics

Fuel elements are spherical ones. Every fuel element contains 5 g of heavy metal. The equilibrium core has a 17% enrichment of 235U. 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.

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.

(c) 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.

(d) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

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.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

Inherent safety, no need for offsite emergency measures.

5. Safety Features

The PeLUIt incorporates the established safety features of pebble bed type 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.

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

N/A

(m) Safety Approach and Configuration to Manage DEC

N/A

(n) Containment System

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.

(o) Spent Fuel Cooling Safety Approach / System

N/A

6. 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.

7. 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.

8. Plant Layout Arrangement

N/A

9. Testing Conducted for Design Verification and Validation

N/A

10.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.

11.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.

12.Waste Management and Disposal Plan

N/A

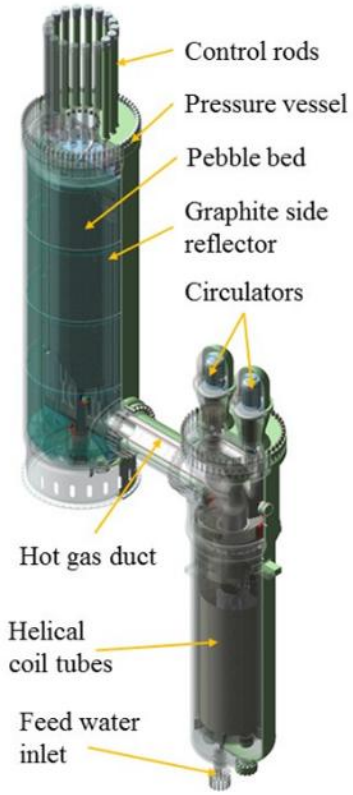
13.Development Milestones

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

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



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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*
Design status	*Under construction*
Thermal/electrical capacity, MW(t)/MW(e)	*200 / 82.5*
NSSS operating pressure (primary/secondary), MPa	*6.0 / 16.5*
Core inlet/outlet coolant temperature (°C)	*260 / 750*
Primary circulation	*Forced helium circulation*
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*
Fuel cycle requirements/approach	*Uranium once through (initially)*
Reactivity control mechanism	*Thermal feedback & control rods*
Approach to safety systems	*Passive*
Seismic design (SSE)	*0.5g*
Design life (years)	*60*
Plant footprint (m^2)	*340m x 385m (4 reactor modules with 4 turbines)*
RV height/diameter (m)	*16.4 / 4.88*
RV weight (metric ton)	*274*

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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☒ Desalination	☐ >550°C

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) Nuclear Steam Supply System

(b) Reactor Core

The Xe-100 core comprises ~220 000 graphite pebbles fuel elements each containing ~18 000 UCO TRISO coated particles. The core is graphite moderated with online refueling capability. The advantage of online refuelling is that the core excess reactivity is maintained at below 2% which means that no burnable poisons are needed to ensure that the reactor reactivity remains within safe shutdown limits at all times. This also improves the neutron economy of the core and helps the Xe-100 to achieve an average burnup of 165 000 MWd/tHM. At full power approximately 173 fresh pebbles are added daily, and a similar number are also removed as spent fuel.

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

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 UO2 (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) 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 B4C 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.

(d) Reactor Vessel (RV) and Internals

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

(e) Reactor Coolant System and Steam Generator

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

The intrinsic safety characteristic of the plant is guaranteed by a relatively low power density of 4.8 MW/m3, 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.

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

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.

(o) Spent Fuel Cooling Safety Approach / System

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;

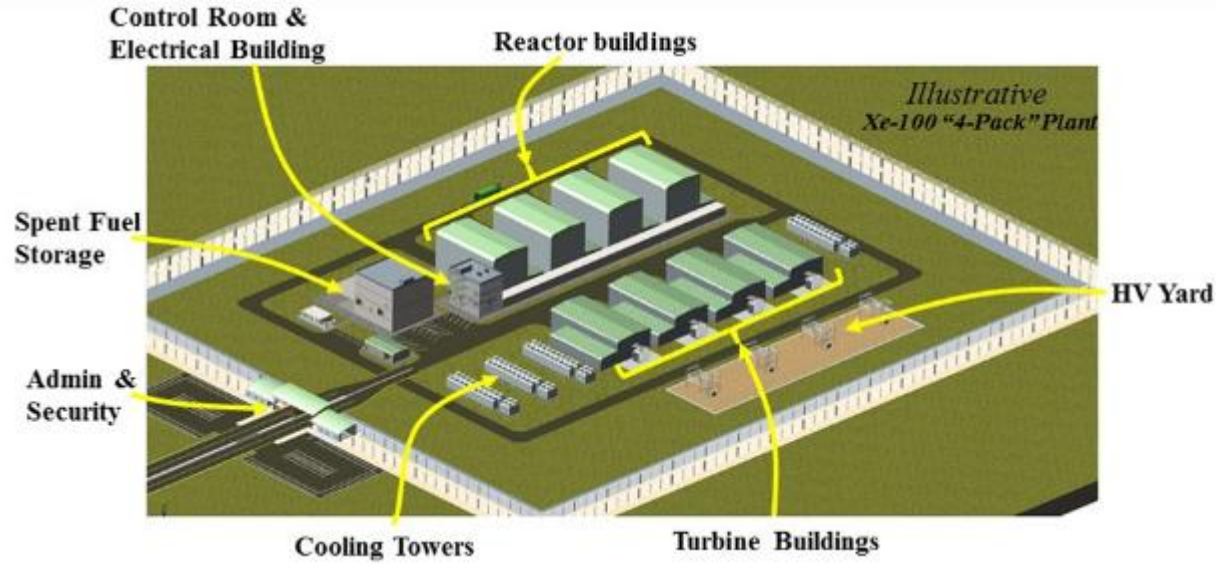
Helium – Chemically and radiologically inert helium is an effective heat transport fluid. Moreover, it does not influence the neutron balance. Helium allows for very high coolant temperatures;

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.

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. Testing Conducted for Design Verification and Validation

10.Design and Licensing Status

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

11.Fuel Cycle Approach

12.Waste Management and Disposal Plan

13.Development Milestones

Year	Milestone	Status
2019	Conceptual Design Development Complete	**Completed**
2021	Basic Design Development Complete	**Completed**
2021	Applications submitted to the U.S. Nuclear Regulatory Commission	**Completed**
2025-	Start of Construction	**Ongoing**

LIQUID METAL COOLED FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS

ARC Clean Technology, United States of America ARC-100



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*ARC Clean Technology, Inc., United States of America*
Reactor type	*LMFR (pool type)*
Coolant/moderator	*Sodium / N/A*
Design status	*Basic Design*
Thermal/electrical capacity, MW(t)/MW(e)	*286 / 100*
NSSS operating pressure (primary/secondary), MPa	*Not pressurized*
Core inlet/outlet coolant temperature (°C)	*355 / 510*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*Metal fuel (U-10%Zr alloy) based on high assay low enriched uranium (HALEU)*
Number of fuel assemblies in the core	*99*
Fuel enrichment (%)	*13.1*
Core discharge burnup (GWd/ton)	*77*
Refuelling cycle (months)	*240*
Fuel cycle requirements/approach	*Metallic HALEU / Open Cycle*
Reactivity control mechanism	*Control Rods, Shut-off Rods and inherent negative reactivity feedback*
Approach to safety systems	*Passive*
Seismic design (SSE)	*0.3 PGA*
Design life (years)	*60*
Plant footprint (m^2)	*54,000*
RV height/diameter (m)	*15.6 / 7.6*
RV weight (metric ton)	*600*

1. Introduction

ARC Clean Technology was founded in 2006 to take the world's most proven advanced reactor design beyond the laboratory and into the real world, delivering the reliable, clean power required by modern industry at scale. The ARC-100 is a 100 MWe sodium-cooled fast reactor refining the underlying technology of the EBR-II which operated successfully at Argonne National Laboratory for over 30 years.

Headquartered in Washington, DC with operations in Canada, ARC builds proven nuclear science into deployable infrastructure. The ARC-100 is factory-fabricated, modular, and delivers 100 MWe of baseload power for rapid deployment. That timeline matches the build schedules of data center campuses and industrial expansions.

ARC is an awardee of the U.S. Department of Energy's Advanced Reactor Demonstration Program (ARDP) and is pursuing licensing under the newly finalized Part 53 framework.

In Canada, ARC is working with New Brunswick Power for deployment of the ARC-100 at the Point Lepreau site and with Nucleon Energy for deployment in Alberta. In 2023, NB Power submitted a License to Prepare Site application to the Canadian Nuclear Safety Commission (CNSC) for the ARC-100 project, and in 2025 ARC successfully completed Phase 2 of the CNSC's Vendor Design Review process. ARC is the only advanced reactor technology in Canada to have achieved both milestones.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

The ARC-100 produces clean electricity for utilities and data centers, as well as heat for industrial applications including clean fuels (hydrogen and ammonia), chemical production, mining, refining, water treatment and desalination, and the creation of life-saving medical isotopes.

3. Design Philosophy

The design philosophy of the ARC-100 is simplicity, less complexity, and scalability:

- Small footprint
- Exponentially fewer parts
- Increased use of standard materials
- No pressure vessel required
- 70-80% of area and equipment is non-nuclear related
- Passive safety replaces active systems
- Repeatability through simplicity (singular unit, 2-pack, 4-pack, 6-pack)

4. Main Design Features

(a) Nuclear Steam Supply System

The nuclear steam supply system of the ARC-100 includes the following:

- Reactor core and fuel assemblies
- Reactor control system
- Reactor vessel system consisting of a reactor vessl shell, core internal structures, top plate and the guard vessel
- Heat transport and auxiliary system consisting of the Primary, Intermediate and steam generaotor systems
- Passive emergency core cooling and heat removal system consisiting of the Direct Reactor Vessel Auxiliary Cooling System (DRACS) and Reactor Vessel Auxiliary Cooling System (RVACS)
- Containment system
- Fuel handling and storage system consisting of an In-vessel Transfer Machaine (IVTM), spent fuel storage locations within the Reactor Vessel shell, an intra-building transfer cask, and dry spent fuel storage modules.

(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 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 Vessel (RV) 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 196 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 and Steam Generator

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.

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.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

(i) Engineered Safety System Approach and Configuration

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

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

(j) Decay Heat Removal System

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

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

The RVACS always operates in a passive state, RVACS removes the reactor's decay heat through the Reactor Vessel and Guard Vessel walls by radiation and convection to naturally circulating air outside the Guard Vessel without

exceeding structural temperature limits. The DRACS is composed of three units operating in a natural convection mode. Heat exchanger loops using NaK as an intermediate transfer heat from the cold sodium pool to air heat exchangers in which the hot air vents to the atmosphere. DRACS can be either passive or active using forced air convection

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) 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.

(o) 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.

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 volage as the fail-safe shutdown systems are designed to operate without electricity.

8. Plant Layout Arrangement

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.

Balance of Plant

The steam turbine is a non-reheat, air cooled, single shaft, single casing turbine with separate HP/IP/LP sections. 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

Canada. In 2025, ARC Clean Technology successfully completed the Canadian Nuclear Safety Commission Vendor Design Review II process with no fundamental barriers to licensing identified. Based on the regulatory requirements applicable to new build projects in Canada, in 2023 a license to prepare site application and an environmental assessment application were submitted for an ARC-100 unit at Point Lepreau, New Brunswick, Canada.

United States. ARC is a participant in the U.S. Department of Energy's Advanced Reactor Demonstration Program (ARDP), supporting accelerated SMR development, and is pursuing licensing in US under the newly finalized Part 53 framework issued by the U.S. Nuclear Regulatory Commission (NRC) in 2026.

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 Plan

The low quantities of solid, liquid, and gaseous radioactive waste resulting from operations will be handled and processed in a responsible and safe manner 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

Year	Milestone	Status
2020	Conceptual Design	**Completed**
2023	Preliminary Design	**Completed**
2023	License to Prepare Site application submitted	**Completed**
2025	CNSC Vendor Design Review Phase II	**Ongoing**
2026-2028	Detailed design	**Ongoing**
Lates 2020s	License to Construct	**Planned**

Oklo Inc.,
United States of America
Aurora Powerhouse Product Line



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Oklo Inc., United States of America*
Reactor type	*LMFR*
Coolant/moderator	*Light metal / N/A*
Design status	*Under construction*
Thermal/electrical capacity, MW(t)/MW(e)	*200 / 75*
NSSS operating pressure (primary/secondary), MPa	*Not pressurized*
Core inlet/outlet coolant temperature (°C)	*600 / 700*
Primary circulation	*Natural circulation*
Fuel type/assembly array	*Metal fuel*
Number of fuel assemblies in the core	-
Fuel enrichment (%)	*19.75*
Core discharge burnup (GWd/ton)	-
Refuelling cycle (months)	*< 120*
Fuel cycle requirements/approach	*HALEU / Closed-loop (Recycling)*
Reactivity control mechanism	-
Approach to safety systems	*Hybrid*
Seismic design (SSE)	*Report generic design basis ground motion (e.g., PGA) or SSE level*
Design life (years)	-
Plant footprint (m^2)	*< 65,000*
RV height/diameter (m)	*- / -*
RV weight (metric ton)	-

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 Application

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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. Design Philosophy

The guiding idea behind Aurora-INL is to make advanced nuclear power easier to build, easier to repeat, and more economically competitive by designing for simplicity from the start.

4. Main Design Features

(a) Nuclear Steam Supply System

Single loop, industrial condensing steam turbine

(b) Reactor Core

Metal-fueled fast reactor with uranium-zirconium metal fuel, including U-10Zr

(c) Reactivity Control

(d) Reactor Vessel (RV) and Internals

A small and simple double-vessel system designed to be fabricated from readily available stainless steel.

(e) Reactor Coolant System and Steam Generator

Near-atmospheric-pressure sodium/liquid-metal operation and a passive air-cooled reactor vessel auxiliary cooling system (RVACS). Steam generator is sodium/water u-tube design.

(f) Pressuriser

Near-atmospheric pressure, non-pressurized, therefore this does not apply for a non-LWR.

(g) Primary pumps

(h) Distinguishing features

Up to 75 MWe, sodium/liquid-metal-cooled fast reactor, metal fuel, up to 10 years before refueling, non-pressurized / near-atmospheric operation, passive air cooling via RVACS, owner-operator / long-term PPA model, and a fuel strategy that includes DOE-managed recovered EBR-II material for the first core plus future recycled-fuel capability.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

Passive-dominant / inherent-and-passive framing. RVACS uses passive air cooling and “maintains long-term cooling capability indefinitely without additional human actions,” with coolant kept at near-atmospheric pressure and well below sodium’s boiling point.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

Low-pressure sodium operation, temperatures below sodium boiling, inherent/passive safety, and passive air cooling for long-term heat removal.

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

(o) Spent Fuel Cooling Safety Approach / System

6. Plant Safety and Operational Performances

Aurora is self-stabilizing, self-controlling, cooled by natural forces, and “walk-away safe” with 95+% reliability.

7. Instrumentation and Control Systems

Aurora is designed to rely on automation and inherent safety features so personnel act more like monitors than operators during normal operations. There are limited SR I&C systems (just a reactor trip system).

8. Plant Layout Arrangement

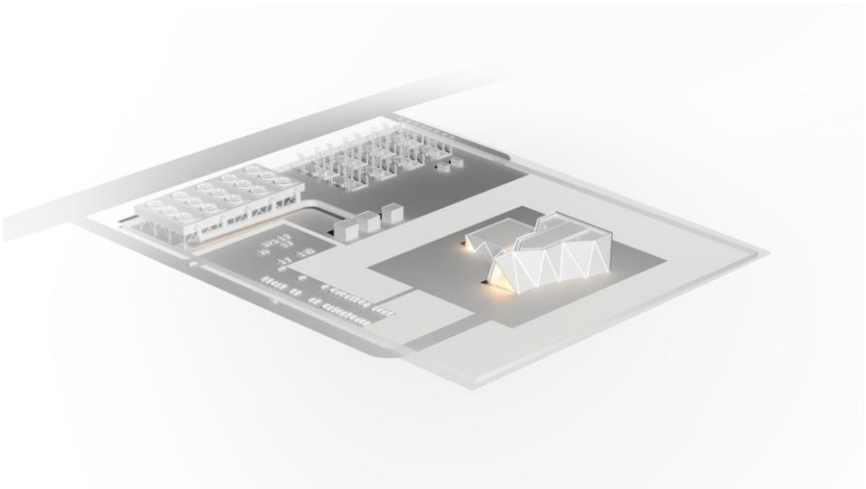


Figure 1. Plant Layout of Oklo's Aurora-INL

The reactor is situated below grade and housed adjacent to the monitoring room and other operations-oriented spaces. Other site elements, including the power conversion system, air-cooled condensers, parking, and transmission infrastructure, are separate structures. The arrangement is also well suited to multi-unit deployment for efficiency and scalability, because portions of the supporting site infrastructure can be shared across units.

9. Testing Conducted for Design Verification and Validation

Full-scale fuel-assembly flow testing of a prototypical fuel assembly at Argonne under a DOE GAIN voucher. This work generated experimental data to validate computer models, demonstrate manufacturing parameters, and advance the fuel-assembly design toward production; including testing pressure drop and flow distribution across operating conditions. EBR-II heritage tests—including loss-of-heat-sink without scram and loss-of-flow without scram demonstrations underlie Aurora powerhouse’s passive/inherent-safety narrative.

10.Design and Licensing Status

The Aurora powerhouse at INL, known as the Aurora-INL, broke ground in September 2025. The Aurora-INL project is now beginning the steps for DOE authorization under the RPP, following the process outlined in DOE Order 1271, with an anticipated transition to NRC licensing prior to commercial operation.

11.Fuel Cycle Approach

Metal-fueled fast reactor, first core backed by DOE-awarded EBR-II-derived material, fresh HALEU as a supply path, and longer-term recycling as the strategic destination.

12.Waste Management and Disposal Plan

13.Development Milestones

Year	Milestone	Status
2016	Began pre-application interactions with the U.S. NRC and ongoing pre-application process for Aurora powerhouses.	Completed
2017	Demonstrated prototype fuel fabrication.	Completed
2018	Conducted thermal testing.	Completed
2019	Safeguards Information Protection and Handling Plan approved.	Completed
2020	Combined license application submittal to the U.S. Nuclear Regulatory Commission and acceptance for review (currently being revised).	Completed
2021	Began pre-application for fuel recycling.	Completed
2021	Partnership with Centrus announced.	Completed
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.	Completed
2023	Began pre-application process for fuel fabrication.	Completed
2024	Safety Design Strategy for the Aurora Fuel Fabrication Facility approved by DOE.	Completed
2024	Completed successful end-to-end demonstration of advanced fuel recycling process, advancing commercial-scale recycling facility.	Completed
2024	Began trading on the New York Stock Exchange under the new ticker symbol “OKLO”.	Completed
2025	Breaks ground on first Aurora powerhouse.	Completed
2025	Advances licensing with completion of NRC readiness assessment.	Completed
2025	Selected for three projects under U.S. Department of Energy’s reactor pilot program.	Ongoing
2026	Announces agreement with Meta Platforms, Inc. that advances plans to develop a 1.2 GW power campus in Pike County, Ohio.	Planned
2026	Announces planned joint venture with Centrus to advance nuclear fuel services in Ohio.	Planned
2026	Announces U.S. Department of Energy approval for Nuclear Safety Design Agreement of Aurora powerhouse at Idaho National Laboratory.	Planned

Blue Capsule Technology, France
BLUE CAPSULE



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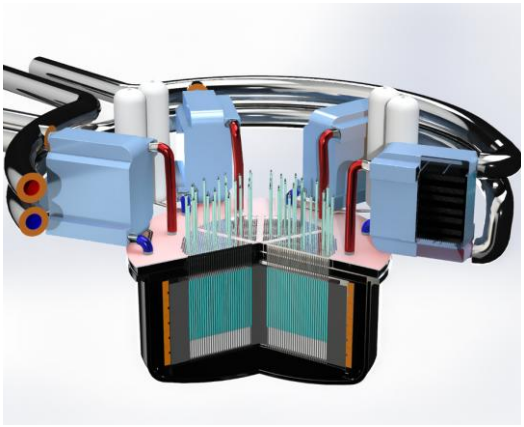


Figure 1: Blue Capsule's reactor core

MAJOR 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*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*170MW(t) gross/60MW(e)*
NSSS operating pressure (primary/secondary), MPa	*0,1 sodium / 0,5 air*
Core inlet/outlet coolant temperature (°C)	*400 °C / 750°C*
Primary circulation	*Natural*
Fuel type/assembly array	*Prismatic, TRISO with UO2 kernel*
Number of fuel assemblies in the core	*6570 (10 assemblies per fuel column)*
Fuel enrichment (%)	*4.95% in 235U*
Core discharge burnup (GWd/ton)	*Approx 35 GWd/t*
Refuelling cycle (months)	*online refuelling*
Fuel cycle requirements/approach	*Open cycle*
Reactivity control mechanism	*Control rods*
Approach to safety systems	*Passive safety (natural coolant circulation, TRISO failure temperature significantly higher than sodium boiling point), ultimate air coolant*
Seismic design (SSE)	*Design is based on Cadarache/Marcoule envelope spectrum. Sensitivity to the EPR2 spectrum is planned*
Design life (years)	*60*
Plant footprint (m^2)	*1,800 - 27,000m2 whole plant*
RV height/diameter (m)	
RV weight (metric ton)	

1. Introduction

Blue Capsule is a 170 MWth (150 MWth net capacity) reactor that allows onsite co-location with hard-to-abate industries. It is designed to maximise the production of industrial-grade heat in the form of hot air at 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 at up to 60 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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☐ Desalination	☒ >550°C

Blue Capsule targets three key end users:

- Decarbonizing existing industrial sites using a "plug-in" approach, with high-temperature steam up to 650°C,
- Supporting “multi-vector”-type end users with co-generation capabilities, and
- Integrating into “pre-heating” end user processes like glass, steel, or the fabrication of ceramics.

Low-temperature applications such as district heating or low-grade steam can be provided thanks to the heat recovery steam generator exhaust air.

Electricity is cogenerated via a back-pressure steam turbine tailored to the end user’s steam requirements, with the option of full steam expansion in electricity-only operation when heat is not required.

3. Design Philosophy

Building on operational feedback from the nuclear safety research reactor and Hallam reactor programs, the Blue Capsule reactor uses sodium as the core coolant and graphite as the moderator. The concept relies on a set of design choices that together enhance safety, operability, and cost competitiveness. The high thermal conductivity of sodium enables strong heat transfer coefficients, allowing the reactor core to be efficiently cooled by natural circulation in the primary loop. This provides a simple and inherently robust capability for decay heat removal, which is effective under both normal operation and abnormal conditions, while minimizing reliance on active systems.

In addition, the high-mass density of sodium, compared with gaseous coolants such as pressurised helium in HTR concepts, results in high thermal inertia of the primary system. This leads to slow system transients, offering extended response times for operators, and facilitating reactor control during abnormal events. The high boiling point of sodium further enables operation at near-atmospheric pressure, removing the need for a pressuriser while allowing the delivery of high-temperature heat to industrial users.

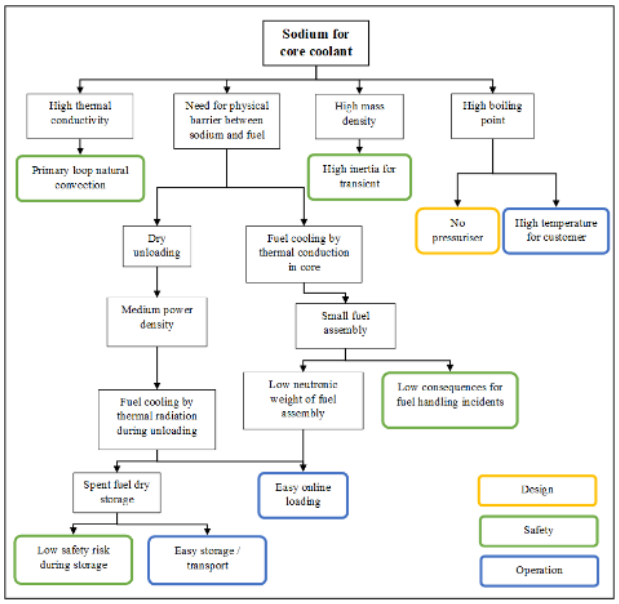


Figure 2: Options taken by the Blue Capsule designers, leading to design, safety and operation benefits.

From a cost perspective, the low-pressure primary system constitutes a major advantage. Given that the primary circuit is not pressurised, structural requirements are significantly reduced, resulting in limited use of steel and no requirement for large, forged components (in contrast with pressurised water or gas-cooled designs). As a result, the overall steel mass is substantially reduced. Compared with the HTTR, the specific steel inventory is approximately an order of magnitude lower on a per-megawatt-thermal basis. This has a direct, and favorable, impact on aspects such as capital cost, manufacturing complexity, and supply chain constraints.

As regards fuel and operations, the requirement for a physical barrier between sodium and fuel drives a fuel design that is based on conductive heat transfer between dry fuel assemblies and sodium-facing barriers. This enables the segmentation of the fuel column into compact, low-mass fuel assemblies. Such a configuration limits the consequences of fuel-handling incidents and supports dry unloading, dry spent fuel storage, and simplified transport logistics, with reduced safety risks during backend operations.

Finally, Blue Capsule’s competitiveness – notably compared with industrial heat sources such as natural gas – rests on its cost-driven, low-pressure design philosophy and its plug-in integration with industrial sites. Together, this means

combining high-temperature heat delivery, simplified safety systems, reduced use of materials, and streamlined operations.

4. Main Design Features

(a) Nuclear Steam Supply System

The heat from the assemblies is evacuated by way of 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. Blue Capsule's 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 for end users. This versatility is achieved by using an "open" energy conversion cycle, similar to a CCGT plant, with an open or semi-open air loop coupled with a Rankine steam cycle.

(b) Reactor Core

The Blue Capsule reactor core is similar to the core of gas-cooled HTRs. The fuel compacts in Blue Capsule's core contain TRISO particles with UO2 kernels (enrichment at 4.95% of 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 sleeves that separate the fuel element from sodium flow, in order to protect the graphite from interaction with sodium. Thus, Blue Capsule reactor fuel is in a dry zone, separated from the sodium flow.

(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. The nuclear power is intended to remain constant, with any variability in the final product being managed through the energy conversion island and/or 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 Vessel (RV) and Internals

The reactor core is enveloped by a non-pressurized vessel ("inner vessel"). 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 above 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 and Steam Generator

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 finned tube type, with tubes arranged in a serpentine and air flowing transversally through the tube bank. This design has the best operational history for sodium-air heat exchangers and minimizes the risk of leaks thus maximising reactor availability. The heat exchangers' design and positioning – as close as possible to the core, minimizing 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.

(h) Distinguishing features

5. Safety Features

(i) Engineered Safety System Approach and Configuration

Blue Capsule takes advantage of the favorable properties of sodium as a coolant, and of TRISO particles, both in terms of safety and performance. Using liquid sodium as a coolant allows for a maximum fuel temperature of 1250°C under normal operation, and to keep the fuel temperature below the TRISO failure limit under 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 cold source, and because no water is involved, sodium-water reaction is avoided by design. The Blue Capsule safety and security approach is therefore summarised as: eliminating the generalised loss of the first barrier; eliminating core uncovering; no risk of sodium release into the atmosphere; no risk of depressurisation, and; fuel qualification in accidental conditions covered by normal operations.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

First, 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, providing strong physical stability. Furthermore, the compact design of the primary circuit with the positioning of the heat exchangers just above the core – thus no pressurisation – leads to a simplification of the required 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 number of redundancies to achieve significant reliability. Concerning external threats, and especially considering co-location with the industrial customer's site, 1/3 of the reactor building is placed at subterranean level to enhance 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.

(m) Safety Approach and Configuration to Manage DEC

Blue Capsule – and in particular the reactor core and safety systems – is being designed so that potential seismic events are considered.

(n) Containment System

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

(o) 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 fuel particules, 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 for dry interim storage in racks, cooled only 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 general fuel failure being impossible in practice.

6. Plant Safety and Operational Performances

The capacity factor will be over 90% (due to online refuelling), the gross thermal power output is 170 MWth, and the outlet temperatures are 750°C primary and secondary: 700°C air, 650°C steam, and 60 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 the energy conversion island and/or 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.

8. Plant Layout Arrangement



Figure 3: 3D rendering of the nuclear and energy conversion islands alone.

The layout of the plant buildings is based on two main principles: proximity between the reactor building and the industrial processes it serves, and the sharing of utilities and support functions. For multiple capsules located on the same site, all functions not housed within individual reactor buildings are intended to be shared. These include electrical supply, utility networks, support facilities, fire protection, physical security, and spare parts storage. Non-protected areas will accommodate conventional systems (secondary circuit, turbine, alternator, compressors, etc.), as well as temporary construction and heavy maintenance activities. Road access is assumed to enable the delivery of large equipment.

For a single unit and no mutualized equipment, the estimated land footprint of the reactor building is 1,800m2 and 27,000m2 for the whole plant.

The building houses all equipment required for reactor operation, including HVAC systems, electrical power supply, mechanical handling systems, and fuel management systems. It also provides enhanced protection against external hazards, such as seismic events and aircraft impacts.

9. Testing Conducted for Design Verification and Validation

The project's development is based on a series of experimental facilities, test programs or projects, in parallel to the reactor project. These are designed to:

- Validate the industrial feasibility of key components in representative environments;
- Strengthen national and European industrial ecosystems, by anticipating the ramp-up linked to mass production;
- Reduce technical and industrial risks prior to the launch of the first-of-a-kind reactor (FOAK).
- Increase the technology readiness level (TRL) of certain components
- Industrialize or reindustrialize certain processes, including through the construction of experimental facilities.

- Make significant advances regarding fuel (TRISO particles, compacts, assemblies); this will be conducted within the framework of Blue Capsule Technology cooperation with Framatome, as well as the ongoing component testing ("mini PoC") program.

The main steps are as follows:

- The ELISE experimental loop, on a reduced scale in flow but on a scale close to 1 in height, will test the manufacture of mechanical components intended to operate in a high-temperature sodium environment, including: purification systems, heat exchangers, sensors, valves, materials, and sealing solutions. Start of operation is due by the end of 2026.
- The laboratory test campaigns, which began in 2025, will confirm the feasibility of manufacturing the fuel elements (graphite/SiC barrels, TRISO compacts) as well as the key components of the core.
- The Non-nuclear program (NNP) will finalize the consolidation of the supply chain through the manufacture of all critical components and systems. It will also be used to qualify assembly, handling and operating processes in representative environments.

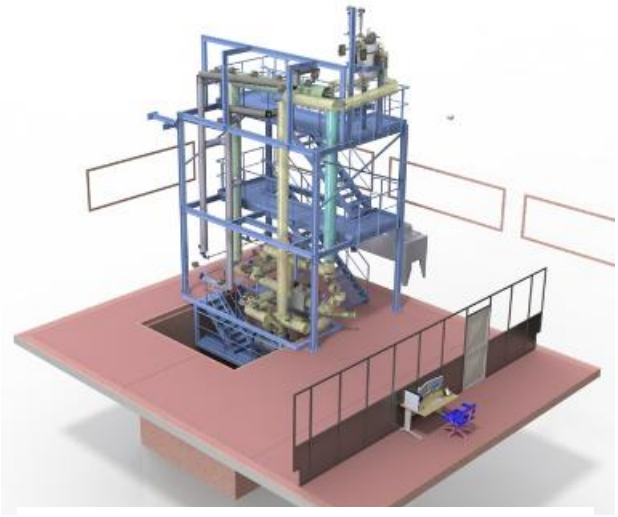


Figure 8: 3D rendering of the ELISE experimental loop.

10.Design and Licensing Status

Blue Capsule conducted the sketch and feasibility study phase in 2022-2023, with the Conceptual Design phase starting in early 2024 (and due to finish mid-2026). The reactor benefits from mature technological components and successful licensing experiences in France and internationally. The design will be validated through an experimental program (see Section 9).

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 by French regulators. This enables 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 the Phenix, Superphenix and Astrid projects.

Since April 30, 2025, Blue Capsule has been engaged in a preparatory review phase with the ASNR (French nuclear safety authority), consisting of a series of technical meetings focused on the project's safety approach and associated challenges. The first two meetings addressed TRISO fuel (qualification and safety aspects) and the sodium coolant (chemistry and safety aspects). The next meeting will cover the safety methodology implemented for the project. Upon completion of this preparatory phase, Blue Capsule plans to submit a safety options report (SOR) in 2027, followed by an application for authorization to create a nuclear facility (DAC), including a preliminary safety report, in 2029.

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 Technology is already considering options for nuclear fuel reprocessing at the preliminary design phase. Since HTR/TRISO fuels are excellent at "burning" materials, Blue Capsule Technology is also exploring options for using MOX-based TRISO, and other options for treating and recycling spent fuels.

Framatome and Blue Capsule Technology have developed a shared vision aimed at ultimately supporting the creation of a self-sufficient supply chain for TRISO fuel for HTR reactors, particularly for Blue Capsule's reactor project. This technological platform will be sited in Romans-sur-Isère, France, and will comprise:

- the TRISO particle pilot line
- the compacts pilot line
- the Blue Capsule fuel assembly workshop.

The TRISO fuel fabrication technology platform will be capable of producing uranium-235 enrichments of up to 20% (HALEU; high-assay low-enriched uranium). However, Blue Capsule's initial target enrichment level will be less than 5%, which will enable the project to operate independently of the HALEU enrichment chain in the short term.

12.Waste Management and Disposal Plan

The management of nuclear waste will be based on the following principles:

- sorting at source according to the nature of the waste;
- collection within the facility using collection points (drums, bins, etc.);
- characterization and identification of waste based on the use of nuclear measurement techniques (spectrometry, etc.) and conventional methods (sampling, weighing, etc.); and,
- packaging in a container approved by the waste destination facility.

Solid nuclear waste produced by the Blue Capsule facility will be classified using the following criteria:

- radiological characteristics (activity, types of primary emitters);
- physico-chemical nature (incinerable, compactable, containing chemical compounds); and
- geometric characteristics determining the type of packaging,

For fuel, the preferred solution at this stage is open-cycle storage in a dedicated facility. In this regard, ROBATEL is a partner in the project and is jointly developing, with Blue Capsule Technology a cask designed for the long-term storage of TRISO fuel. This cask will be filled with Blue Capsule reactor spent fuel and stored pending a final repository or reprocessing pathway.

In this regard, experiments have been carried out to recycle TRISO fuel. The Blue Capsule study planned with academic and industrial partners will provide a precise answer to the question of the compatibility of Blue Capsule spent fuel recycling with existing and future fuel recycling facilities.

As concerns graphite, the presence of large quantities (fuel element structures plus core moderator elements) creates challenges in terms of decommissioning, radiological classification and the waste management pathway. However, Blue Capsule Technology designs its systems with a view to minimizing the use of materials that are potentially problematic from a waste management perspective and optimizing the waste management pathway.

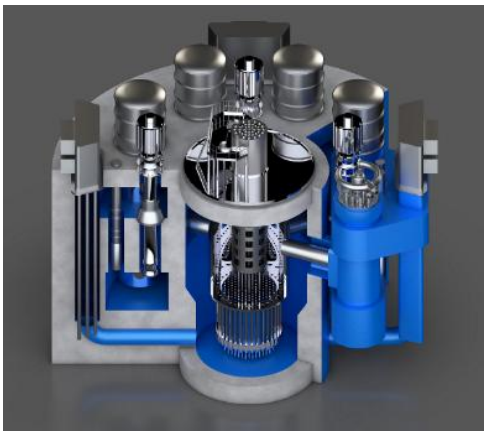
13.Development Milestones

Year	Milestone	Status
2022-2023	Concept description (using previously developed patents).	**Completed**
2024-2026	Conceptual design phase (and pre-licensing)	**Ongoing**
2027	ELISE commissioning date	**Planned**
2027	Site selection and FOAK contractualization	**Planned**
2026-2028	Basic design phase	**Planned**
2028-2031	Detailed design phase	**Planned**
2031-34	Manufacturing phase	**Planned**
2031	FOAK First concrete poured	**Planned**
2035	FOAK Commissioning date	**Planned**

NIKIET, Russian Federation
BREST-OD-300



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*NIKIET, Russian Federation*
Reactor type	*Liquid metal cooled fast reactor*
Coolant/moderator	*Lead*
Design status	*Under construction*
Thermal/electrical capacity, MW(t)/MW(e)	*700 / 300*
NSSS operating pressure (primary/secondary), MPa	*0.1 / 17 – 18.5*
Core inlet/outlet coolant temperature (°C)	*420 / 535*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*Mixed uranium plutonium nitride*
Number of fuel assemblies in the core	*169*
Fuel enrichment (%)	*up to 14.5*
Core discharge burnup (GWd/ton)	*61.45*
Refuelling cycle (months)	*36 – 78*
Fuel cycle requirements/approach	*Closed fuel-cycle. It uses nitride of depleted U with Pu*
Reactivity control mechanism	*Reactivity compensation (RC), emergency protection (EP) and automatic control (AC) members*
Approach to safety systems	*Passive*
Seismic design (SSE)	*VII-MSK 64*
Design life (years)	*30*
Plant footprint (m^2)	*80 × 80*
RV height/diameter (m)	*17.5 / 26*
RV weight (metric ton)	*27 000*

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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) Nuclear Steam Supply System

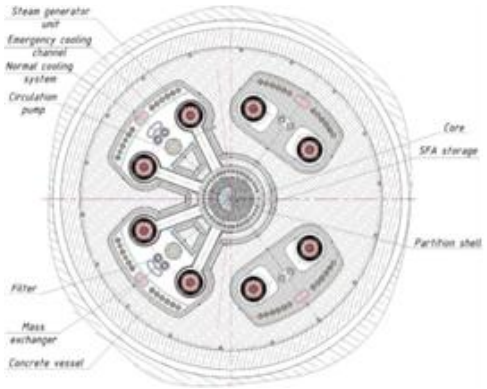


Figure 1. BREST-OD-300 Layout

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.

(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/cm3) 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.

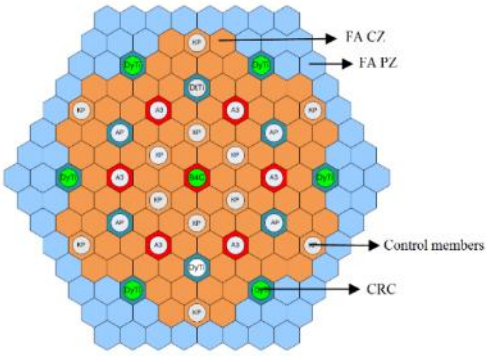


Figure 2. BREST-OD-300 Reactor Core Map

(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 Vessel (RV) 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 and Steam Generator

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

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.

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

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) Containment System

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

(o) 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 -55oC and 37oC respectively. The system is passive.

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 3.2·10-8 1/year.

7. Instrumentation and Control Systems

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

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

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.

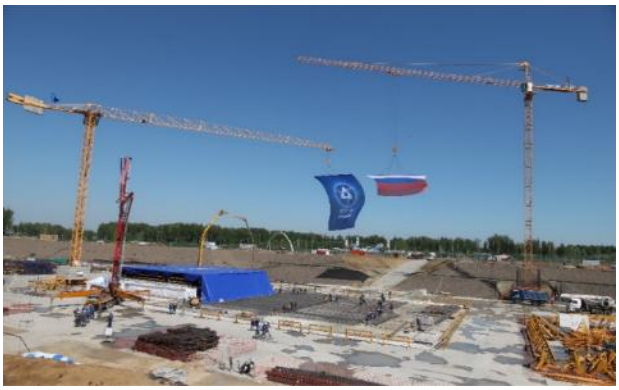


Figure 3. Pouring the first concrete into the foundation of the power unit with the BREST-OD-300 reactor

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.

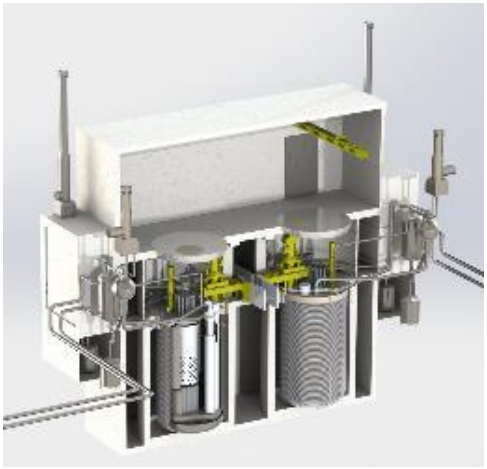
13.Development Milestones

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

Hexana, France
HEXANA



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*HEXANA, France*
Reactor type	*AMR*
Coolant/moderator	*Sodium/none*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*2 x 400 MWth*
NSSS operating pressure (primary/secondary), MPa	*2 x 150 MWe*
Core inlet/outlet coolant temperature (°C)	*N/A*
Primary circulation	*400-530*
Fuel type/assembly array	*Forced convection*
Number of fuel assemblies in the core	*MOX*
Fuel enrichment (%)	*80*
Core discharge burnup (GWd/ton)	*<28% Pu*
Refuelling cycle (months)	*110*
Fuel cycle requirements/approach	*12 to 18*
Reactivity control mechanism	*N/A*
Approach to safety systems	*Control rods*
Seismic design (SSE)	*Diversified with passive*
Design life (years)	*N/A*
Plant footprint (m^2)	*60 years*
RV height/diameter (m)	*15000*
RV weight (metric ton)	*14/9*

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

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

At present, 43% of global CO2 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 MWth 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 CO2, 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, CO2 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 gCO2/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 tailor-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 H2 per day for a twinned configuration. When compared to H2 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 H2 than the PWR. This value reaches +50% if thermal coupling cannot be achieved with the PWR.

4. Main Design Features

(a) 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.

(b) 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.

(c) 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 becomes too hot, without I&C. That kind of device was studied during the ASTRID project.

(d) Reactor Vessel (RV) 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).

(e) 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).

(f) Pressuriser

No pressure in a Sodium Fast Reactor.

(g) 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.

(h) Distinguishing features

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) 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.

(m) 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.

(n) 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.

(o) 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.

6. 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.

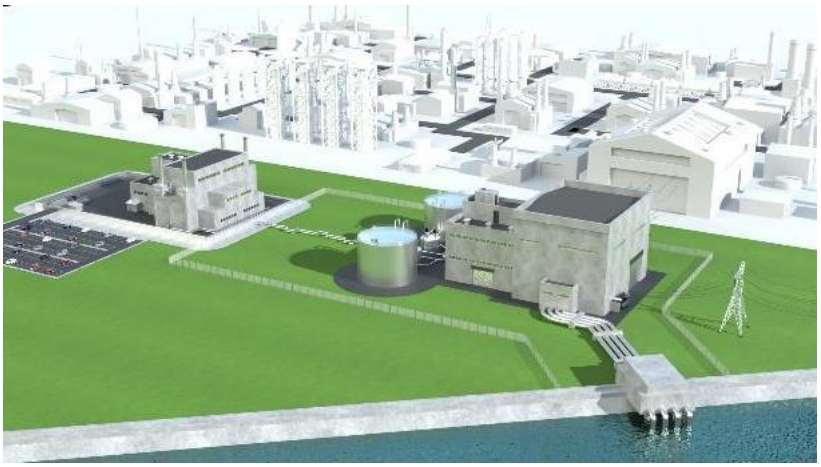
7. Instrumentation and Control Systems

I&C systems developed for 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.

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

8. Plant Layout Arrangement

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



9. 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.

10.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, several security cases are planned concerning non-proliferation issues and physical protection.

11.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 CO2 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 2.5gCO2éq/kWh, which is unbeatable.

12.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.

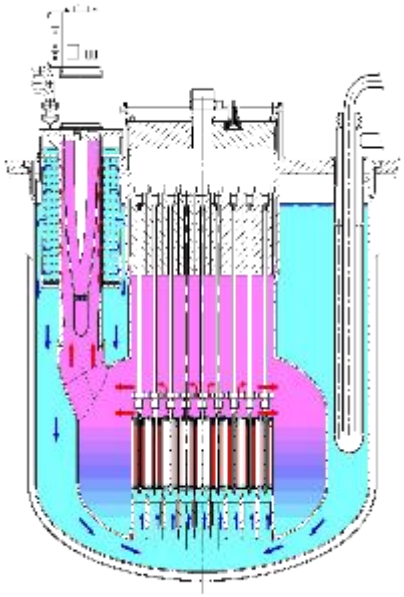
13.Development Milestones

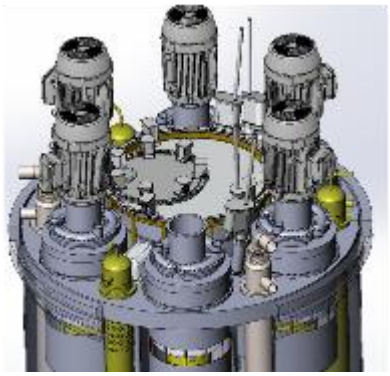
Year	Milestone	Status
2020-2023	Began pre-application interactions with the U.S. NRC and ongoing pre-application process for Aurora powerhouses.	**Completed**

Year	Milestone	Status
2024-2026	Demonstrated prototype fuel fabrication.	Ongoing
2026-2028	Conducted thermal testing.	Planned
2028-2030	Safeguards Information Protection and Handling Plan approved.	Planned
2030	Combined license application submittal to the U.S. Nuclear Regulatory Commission and acceptance for review (currently being revised).	Planned

NewCleo, Italy/France
LFR-AS-200

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Newcleo, Italy/France*
Reactor type	*Lead-cooled Fast Reactor (LFR)*
Coolant/moderator	*Lead (Pb)/None*
Design status	*Conceptual design*
Thermal/electrical capacity, MW(t)/MW(e)	*480 / 200*
NSSS operating pressure (primary/secondary), MPa	*0.01 / 18*
Core inlet/outlet coolant temperature (°C)	*420 / 530*
Primary circulation	*Forced circulation (6 pumps)*
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"*
Fuel cycle requirements/approach	
Reactivity control mechanism	*N/A*
Approach to safety systems	*Control rods, burnable absorbers (B4C)*
Seismic design (SSE)	*Passive and active, no core melt*
Design life (years)	*0.3g*
Plant footprint (m^2)	*60*
RV height/diameter (m)	*1,100 m² (nuclear island)*
RV weight (metric ton)	*6.2 m / 6.3 m at flange*

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

☒ Electricity	☒ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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.

(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 Vessel (RV) 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.

(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.

(h) Distinguishing features

Amphora-shaped inner vessel, lead coolant, no intermediate loops, compact design.

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) 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.

(m) 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.

(n) 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.

(o) 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

Year	Milestone	Status
2016	Completion of pre-conceptual design (by Hydromine Nuclear Energy S.àR.L)	**Completed**
2017-2019	Completion of conceptual design	**Completed**

Year	Milestone	Status
2022	Hydromine Nuclear Energy S.àR.L. is incorporate in newcleo for a fast development programme. Design restarted after an idle phase	Ongoing
2026	Start of pre-licencing discussions in France	Ongoing

TerraPower, LLC, United States of America The Natrium™ Project



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*TerraPower, LLC USA*
Reactor type	*Fast neutron spectrum*
Coolant/moderator	*liquid metal sodium*
Design status	*Under construction*
Thermal/electrical capacity, MW(t)/MW(e)	*840 / 345*
NSSS operating pressure (primary/secondary), MPa	*Low pressure/atmospheric*
Core inlet/outlet coolant temperature (°C)	*540*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*Metallic Uranium Alloy HALEU fuel/hexagonal*
Number of fuel assemblies in the core	*N/A*
Fuel enrichment (%)	*Maximum of 19.75%*
Core discharge burnup (GWd/ton)	*150 GWd/ton*
Refuelling cycle (months)	*18-24*
Fuel cycle requirements/approach	*N/A*
Reactivity control mechanism	*control rod drive mechanism*
Approach to safety systems	*passive*
Seismic design (SSE)	*>0.3g*
Design life (years)	*60*
Plant footprint (m^2)	*N/A*
RV height/diameter (m)	*20.4/11.6*
RV weight (metric ton)	*2450*

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.

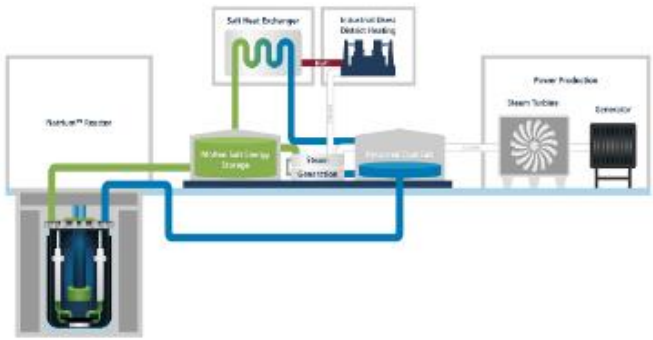
2. Target Application

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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 Vessel (RV) 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.

(h) Distinguishing features

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) 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.

(m) 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.

(n) 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.

(o) 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. Plant Safety and Operational Performances

7. 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.

8. 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.

9. 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.

10. 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.

11. 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.

12.Waste Management and Disposal Plan

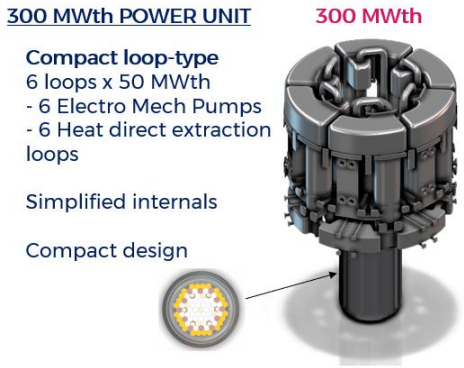
The Project will generate solid low-level radioactive wastes through normal plant operation, including anticipated operational occurrences. These wastes will be addressed via a case-by-case treatment and the disposal process will be tailored to the waste stream. Spent fuel will remain in the spent fuel 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.

13.Development Milestones

Year	Milestone	Status
2008-2016	Preliminary studies and technological innovation (using previously developed patents).	**Completed**
2017-2019	Pre-conceptual design phase and technology validation	**Completed**
2019-2023	Conceptual Design Phase (and preparation for pre-licensing)	**Completed**
2023-2025	Basic Design Phase	**Completed**
2025~	Commercialisation	**Ongoing**
2025-2026	Detailed Design Phase	**Ongoing**
2026	Target first concrete for the FOAK plant	**Planned**

Otrera New Energy, France
OTRERA 300

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Otrera New Energy, France*
Reactor type	*SFR / N/A*
Coolant/moderator	*Sodium*
Design status	*Basic Design Phase*
Thermal/electrical capacity, MW(t)/MW(e)	*Otrera ONE: 200MWth => 70 to 90 Mwe + heat, Commercial flagship: Reactor Unit 300 MWth => 110 Mwe + 180 MWth @100°-150°*
NSSS operating pressure (primary/secondary), MPa	*0.1 MPa / 18 MPa*
Core inlet/outlet coolant temperature (°C)	*400°C / 550°C*
Primary circulation	*Compact loop*
Fuel type/assembly array	*MOX / Hexagonal*
Number of fuel assemblies in the core	*53*
Fuel enrichment (%)	*MOX: 17% Pu. HALEU: below 20 % U235*
Core discharge burnup (GWd/ton)	*100 GWd/ton*
Refuelling cycle (months)	*84 to 120 months (two-batch management, reshuffle every 42 to 60 months)*
Fuel cycle requirements/approach	*Closed fuel cycle; multi-recycling of plutonium; MOX recycling; HALEU full-compatible.*
Reactivity control mechanism	*Control rods (CRDMs) and passive shutdown rods*
Approach to safety systems	*Inherent, passive and simplified active systems; practical elimination of large or early releases, with a level-4 core catcher; primary sodium fire eliminated by design; no sodium-water interface in the commercial configuration*
Seismic design (SSE)	*Envelop SSE, covered by deterministic site specific SDD levels, under RFS 2001-01 standard*
Design life (years)	*60 years, with a further 80-year design objective*
Plant footprint (m^2)	*< 60 000 m² for NPP,< 3500 m² for Reactor building,< 8000 m² for Nuclear Island*
RV height/diameter (m)	*< 12 m / <5.5 m*
RV weight (metric ton)	*< 100 metric tons*

1. Introduction

OTRERA New Energy, a spin-off from CEA, designs, develops and deploys Generation IV advanced modular reactors (AMR) for decarbonized electricity and heat.

The commercial design, OTRERA 300, is a sodium-cooled fast reactor (SFR) of 300 MW(t) delivering 110 MW(e) together with up to 180 MW(t) of process heat between 100 and 150°C. A twin-unit plant delivers 220 MW(e) and 360 MW(t). The heat is recovered at the rejection stage of the conversion cycle, so cogeneration does not reduce the electrical output.

The primary system is a compact loop architecture built from six standardized 50 MW(t) modules. Each module contains one electromagnetic pump and one compact sodium-to-nitrogen heat exchanger, and transfers heat directly to a closed nitrogen Brayton cycle. There is no intermediate sodium circuit. Removing it removes the sodium-water interface from the plant and reduces component count, building volume and sodium inventory.

OTRERA ONE, the industrial precursor, is a 200 MW(t) unit delivering 70 to 90 MW(e). It uses four modules of the same 50 MW(t) design, with an intermediate sodium circuit and a Rankine cycle, a configuration already qualified on Phénix and Superphénix. It de-risks the program and serves as the optimization bay for the direct sodium-to-nitrogen conversion of the commercial fleet. The same 50 MW(t) module is therefore qualified once and used in both configurations.

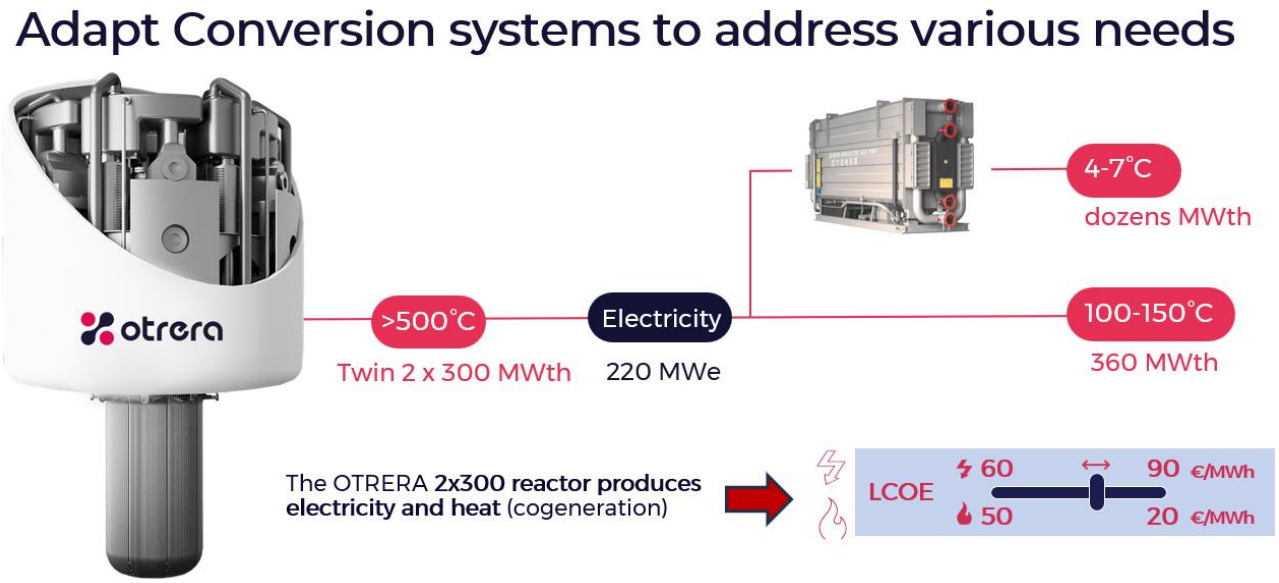
The technology builds on more than 60 years of French development and operation of sodium fast reactors: Rapsodie, Phénix, Superphénix and the ASTRID project. The remaining work is industrial and regulatory rather than scientific, and no separate experimental reactor is required.

Key features of OTRERA 300:

- Direct sodium-to-nitrogen Brayton conversion, without intermediate sodium circuit.
- Simultaneous delivery of 110 MW(e) and 180 MW(t), the heat being taken from the cycle heat rejection. A trigeneration option converts the 150°C heat into chilled water down to 4°C through absorption chillers, for datacentres and industrial processes. A high-temperature option delivers heat up to 550°C when electrical output is traded for heat.
- Six standardized 50 MW(t) modules, shop-manufactured and tested before delivery, common to OTRERA ONE and OTRERA 300.
- Patented iso-breeding core, operated over 7 to 10 years, with a reshuffle every 3.5 to 5 years, compatible with MOX (Pu below 20%) and with HALEU (below 20% U235).
- A back end that is already in operation: oxide fuel compatible with the industrial treatment and recycling route operated in France, with no new facility required.
- Three-barrier confinement, the third barrier being a double-walled metallic enclosure under inert gas, which decouples the confinement function from protection against external hazards. The civil structures act as a physical protection shell, and the ventilation system maintains an additional dynamic confinement between the third barrier and this shell.
- Formal pre-licensing under way with the French safety authority (ASNR) since July 2026.
- EDF and several leading companies of the French nuclear industry are shareholders; OTRERA is a France 2030 laureate

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☒HTSE - electrolysis	☒ <200°C
☒ Load-following	☒PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C



OTRERA 300 is designed for siting close to industrial users. The balance between electricity and heat is adjustable during operation. In the trigeneration configuration, the reactor also supplies chilled water down to 4°C for data centres and industrial cooling.

3. Design Philosophy

The design philosophy of OTRERA 300 is to obtain a compact and modular architecture that combines a high safety level with simplified systems and competitive capital cost. The design aims at high availability, at manufacturability, and at deployment close to industrial end users.

The main architectural choices reduce operational complexity, maintenance requirements and the associated costs. They are made possible essentially by an optimized core and the six independent loops per unit design.

- A core designed for iso-breeding, producing approximately as much fissile material as it consumes, and managed in two batches over a cycle of 7 to 10 years. The long cycle reduces the number of fuel handling operations over the plant life and improves the use of the fuel resource.
- A safety architecture that combines inherent characteristics, passive systems and simplified redundant active systems. The reactor block is placed below ground level and the primary sodium is enclosed in a double-walled metallic enclosure under inert gas, which eliminates the primary sodium fire by design and separates the confinement function from the protection against external hazards.
- A modular approach to construction, operation and maintenance, based on:
    - subassemblies of the reactor unit manufactured and tested in the shop before delivery;
    - fuel handling equipment shared between the units of a same reactor building. In a twin configuration, and up to three units in a single reactor building, the fuel handling machine, the handling hood and the associated tooling are common to all units. Handling equipment is therefore procured, qualified and maintained once per building rather than once per unit, and is designed so that a drop of a subassembly is precluded by the handling device itself;
    - maintenance of the components operating outside sodium, in particular pumps and heat exchangers, designed for accessibility and replacement;
    - dedicated access and procedures for the components operating in sodium;
    - hardened instrumentation and signal transmission qualified for the sodium environment;
    - skid-mounted auxiliary and supporting systems;
    - a general layout based on established civil engineering methods.
- A capital cost driven by component count and by component simplification. The six identical 50 MW(t) modules replace the small number of large and dedicated components of a pool-type architecture, and the direct sodium-to-nitrogen conversion removes the intermediate sodium circuit and its associated systems. The resulting component count and heat exchanger cost are for instance significantly below those of the latest ASTRID pool type reference design.

- A design compatible with the closure of the fuel cycle. The core supports the multi-recycling of plutonium and the use of recycled fissile materials. For countries operating a nuclear fleet, OTRERA can be integrated into the existing fuel cycle infrastructure and contributes to reducing the dependence on external supplies of fuel and fissile materials.

4. Main Design Features

(a) Nuclear Steam Supply System

The OTRERA 300 unit delivers 300 MW(t) to a closed nitrogen Brayton cycle operating at 18 MPa. Heat is transferred directly from the primary sodium, which is cooled from 550°C to 400°C, to the nitrogen, which is heated from 300°C to 530°C, through six compact sodium-to-gas heat exchangers in counterflow arrangement. There is no intermediate sodium circuit and no steam generator in this configuration.

The unit delivers approximately 110 MW(e) together with up to 180 MW(t) of controllable heat between 100 and 150°C. This heat is taken at the rejection stage of the Brayton cycle, so its delivery does not reduce the electrical output. The balance between electricity and heat is adjusted during operation according to end-user demand. When electrical output is traded for heat, the unit can instead deliver heat up to 550°C by direct exchange with the primary system, for higher temperature industrial processes.

In the OTRERA ONE configuration, four modules of the same 50 MW(t) design feed an intermediate sodium circuit, a steam generator and a Rankine cycle, delivering 70 to 90 MW(e).

(b) Reactor Core

The core is a breakeven design based on a hexagonal array of 53 fuel assemblies, each containing 217 fuel pins. It is managed in two batches, with a reshuffle every 3.5 to 5 years and a total irradiation period of 7 to 10 years at nominal power.

The average discharge burnup is approximately 100 GWd/tHM, with a maximum cladding damage below 120 dpa. Both values are within the range already reached on oxide fuel in Phénix

The core accommodates two fuel options. MOX with a plutonium content below 20% is the reference and supports the multi-recycling of plutonium described in section 11. HALEU oxide below 20% U235 provides an alternative supply route for early deployment and for markets without access to recycled material. Both options use the same assembly design and the same core geometry.

(c) Reactivity Control

Reactivity is controlled by control rod drive mechanisms, which provide operational reactivity control during the cycle and contribute to the shutdown function. The burnup reactivity swing is limited by the iso-breeding core design, which reduces the reactivity worth to be compensated over the cycle.

A second and independent shutdown system is provided by passive shutdown rods installed within the core. This system does not rely on control rod drive mechanisms, on external power or on operator action.

Each of the two systems is designed to bring the reactor to a safe shutdown state independent of the other.

(d) Reactor Vessel (RV) and Internals

The reactor vessel incorporates a cold collector, the core region and a hot collector. Primary sodium enters from the lower part of the vessel, flows upward through the core and is collected above it. The core is supported by a dedicated core support plate, while the internals and upper structures are supported by the vessel and its deck.

The primary sodium operates at approximately 400°C at the core inlet and 550°C at the core outlet, with a nominal flow rate of approximately 7 200 m³/h. The vessel operates at near-atmospheric pressure and is therefore not designed as a pressure-retaining component.

The resulting vessel is less than 12 m high, less than 5.5 m in diameter and weighs less than 100 t. These dimensions allow shop manufacturing of the vessel and factory testing of large subassemblies, with transport by road or waterway and limited fabrication work on site.

(e) Reactor Coolant System and Steam Generator

Reactor Coolant System and Energy Conversion Interface

The primary coolant system consists of six standardized 50 MW(t) modules, each integrating one electromagnetic sodium pump and one compact sodium-to-gas heat exchanger. The six modules are identical and hydraulically independent, so that a module can be isolated and its components handled after shutdown without affecting the integrity of the primary system.

The whole of the primary system, including the six modules, is contained within a double-walled metallic enclosure maintained under inert gas. This enclosure forms the third confinement barrier and prevents any contact between primary sodium and oxygen.

This modular architecture, based on six standardized modules, contributes to the compactness, simplification and maintainability of the reactor and facilitates the factory manufacturing and testing of key components.

Steam Generator

The OTRERA 300 Brayton commercial configuration has no steam generator. Heat is transferred directly from the primary sodium to the nitrogen of the Brayton cycle, so there is no sodium-water interface in the plant and the sodium-water reaction is eliminated by design.

In the OTRERA ONE configuration, an intermediate sodium circuit transfers the heat of the four primary loops to a sodium-to-water steam generator feeding a Rankine cycle. This arrangement, including the steam generator technology, its leak detection systems and the protection of the intermediate circuit against a tube failure, was qualified during the operation of Phénix and Superphénix and is used without major modification of its operating principles. Design work has focused on compactness and on the ease of out-of-sodium operations.

(f) Pressuriser

Not applicable. The primary system operates at near-atmospheric pressure, and the cover gas is maintained slightly above atmospheric pressure. A passive surge tank is provided on each of the six heat extraction loops, between the pump and the heat exchanger, to accommodate the thermal expansion of the sodium.

(g) Primary pumps

Six electromagnetic pumps circulate the primary sodium, one per loop. Electromagnetic pumping removes all rotating mechanical components in contact with sodium, together with their bearings, seals and shaft penetrations through the primary boundary, which are among the main sources of maintenance and of leakage paths on mechanical primary pumps.

An electromagnetic pump has no mechanical inertia. The flow coastdown required on loss of electrical supply is therefore obtained through the pump power supply arrangement, which is designed to match the thermal inertia of the primary system and the actuation of the passive decay heat removal systems.

(h) Distinguishing features

Direct sodium-to-nitrogen Brayton conversion without intermediate circuit. 110 MW(e) and 180 MW(t) delivered simultaneously, without loss of electrical output. Closed fuel cycle on facilities already in industrial operation.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The safety approach of OTRERA 300 relies first on the physical characteristics of the design, then on passive systems, and finally on simplified and redundant active systems.

The primary system operates at near-atmospheric pressure, which removes the driving force for a fast loss of coolant and allows the primary boundary to be designed without pressure-retaining constraints. The margin between the nominal sodium temperature and its boiling point exceeds 300°C, which gives the core a large thermal inertia and a slow response to transients. The reactor block is placed below ground level, which decouples the protection against external hazards, aircraft impact and missiles from the confinement function.

The core is arranged as a basket with transverse and ascending flow paths, which reduces the consequences of a subassembly inlet blockage. A large sodium inventory is maintained in the upper hot collector above the core, so that the core remains covered even under a postulated large break in a primary loop.

The primary sodium is entirely enclosed in a metallic containment enclosure under inert gas. A primary sodium fire is therefore eliminated by design, since no oxygen is available at any point of the primary system. In commercial configuration, heat is transferred directly from sodium to nitrogen, so there is no sodium-water interface anywhere in the plant. In the OTRERA ONE configuration, an intermediate sodium circuit separates the primary sodium from the steam generator, following the arrangement qualified on Phénix and Superphénix.

(j) Decay Heat Removal System

Decay heat is removed from the core, and from the spent subassemblies stored at the core periphery, by three diverse systems:

- two independent and passive sodium loops, with heat exchangers immersed in the primary system, rejecting heat by natural circulation;

- one passive oil circuit surrounding the safety vessel in the reactor pit;
- one active oil circuit connected to the primary sodium purification system.

Under abnormal transients or postulated accidents, the non-safety active cooling systems act as the first line of defense. Their failure actuates the passive systems, and the plant reaches and maintains a safe shutdown state for an unlimited period without electrical power, make-up gas or operator action.

Spent subassemblies stored in the peripheral positions are immersed in the primary sodium and are therefore covered by these same systems, with no dedicated safety function. The on-site storage drum, which receives subassemblies after further decay, has its own passive cooling.

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

Design Basis Conditions are managed by the two reactivity control systems and by the decay heat removal systems described above. Reactivity is controlled by the control rod drive mechanisms, and an independent set of passive shutdown rods provides a diverse shutdown capability that does not depend on the control rod drive mechanisms or on external power. The negative reactivity feedback of the core and the thermal inertia of the primary sodium inventory above the core keep temperature excursions within the design margins during operational and incidental transients.

(m) Safety Approach and Configuration to Manage DEC

Design Extension Conditions are managed by the passive decay heat removal systems, by the sodium inventory above the core, and by the third containment barrier. For conditions involving core degradation, the design includes a core catcher located below the core support structure, considered as a level 4 defense in depth measure, which retains and cools degraded fuel inside the primary vessel. The objective of the safety demonstration is the practical elimination of situations that could lead to large or early radioactive releases, so that off-site emergency measures are not required beyond the site boundary.

(n) Containment System

Radioactive material is confined by three barriers: the fuel cladding, the primary sodium boundary, and a dedicated metallic containment enclosure. The third barrier is a double-walled structure under inert gas that encloses the whole of the six heat extraction loops. It consists of an upper and a lower section, each made of a primary enclosure surrounded by a complete secondary enclosure. It confines the primary sodium in the event of a loss of integrity of the primary boundary and contributes to the confinement of radioactive materials under severe conditions.

The confinement function is separated from the protection against external hazards. The metallic enclosure provides confinement; the civil structures of the reactor building provide protection against external hazards. This separation allows the requirements on the reactor building walls and civil work to be set independently from the confinement requirements, while increasing the confinement performance. The civil structures act primarily as a physical protection shell against external hazards, and the ventilation system maintains an additional dynamic confinement between the third barrier and this shell.

(o) Spent Fuel Cooling Safety Approach / System

Spent subassemblies are first moved to peripheral storage positions inside the reactor vessel, where they remain immersed in the primary sodium and are cooled by the primary system itself. Their decay heat is therefore removed by the same passive systems as those that cool the core, with no dedicated safety system and no additional heat sink.

After sufficient decay, subassemblies are transferred under inert gas to a dedicated storage drum located on site. Decay heat is removed from the drum by natural convection, without electrical power or operator action. The heat load at transfer is a small fraction of the load handled inside the reactor vessel, which sets the required cooling capacity of the drum at a low level.

There is no spent fuel pool and no water inventory associated with fuel handling or storage at any stage. The loss of pool cooling and pool boil-off sequences that govern the design of water-cooled reactors do not apply. Handling operations are performed with equipment designed so that a drop of a subassembly is precluded by the handling device itself.

Subassemblies are washed and packaged when they are prepared for shipment to the treatment plant.

6. Plant Safety and Operational Performances

A capacity factor of 97% is a design objective, to be reached once fleet operating maturity is achieved. The core is managed in two batches over a cycle of 7 to 10 years, with a reshuffle every 3.5 to 5 years. Refueling and reshuffling

operations are therefore the only scheduled fuel outages over the cycle, which removes the refueling outage that occurs every 12 to 24 months in water-cooled reactors and governs their availability.

Maintenance is designed together with the plant. The six heat extraction loops are identical and independent, so a loop can be isolated, and its components handled after shutdown without affecting the integrity of the primary system. Components operating outside sodium, particularly the gas cycle machinery, are accessible during operation. Components operating in sodium are handled through dedicated procedures and equipment, after drainage and transfer to a specialized facility. Fuel handling equipment is shared between the units of the same reactor building.

Specialized maintenance operations on sodium components are of the same nature as dismantling operations and use the same tooling and facilities. Maintenance therefore builds the operating experience and qualifies the equipment that will later be used for decommissioning. The approach draws on the maintenance and decommissioning experience of both the French PWR fleet and the French sodium fast reactors.

7. Instrumentation and Control Systems

OTRERA 300 uses digital instrumentation and control systems. The reactor is operated from a control room located in the electrical building, with centralized monitoring of the main reactor and plant parameters.

Beyond the functions common to any nuclear plant, the design implements the instrumentation specific to a sodium fast reactor, developed and operated on the previous French reactors: sodium leak detection on the primary and, where applicable, intermediate circuits; thermocouples at the outlet of each fuel subassembly for the detection of local flow reduction; delayed neutron detection for the early identification of a cladding failure; and continuous monitoring of the sodium chemistry through the purification circuits. This instrumentation is qualified for operation in sodium and under the temperature and radiation conditions of the primary system.

The absence of pressure in the primary system allows a continuous monitoring approach based on leak detection before break, rather than on periodic inspection of a pressure boundary.

The I&C architecture supports continuous monitoring, early detection of abnormal operating conditions and the management of maintenance activities. Centralized monitoring also allows operating data to be aggregated and analyzed across an OTRERA fleet, in support of predictive maintenance.

8. Plant Layout Arrangement



Site footprint: less than 60 000 m² for the plant, less than 8 000 m² for the nuclear island, less than 3 500 m² for the reactor building in a twin configuration of two 300 MW(t) units.

The reactor block is located below ground level. The confinement of radioactive materials is provided by the metallic enclosure, and the civil structures of the reactor building provide the shell protection against external hazards. The requirements applicable to the reactor building can therefore be set independently from the confinement requirements.

The safety demonstration aims at the practical elimination of the situations that could lead to large or early radioactive releases, so that off-site emergency measures beyond the site boundary are not required. This objective supports siting close to industrial end users.

9. Testing Conducted for Design Verification and Validation

The verification and validation program distinguishes between what French sodium fast reactor experience has already qualified and what is specific to the OTRERA architecture and therefore has to be qualified again.

The core, the MOX fuel, the sodium technology, the steam generator of the OTRERA ONE configuration and the general behavior of a sodium primary system are supported by the operation of Rapsodie, Phénix and Superphénix and by the ASTRID design studies. No demonstration of the fundamental principles of the technology is required, and no separate experimental reactor is needed.

Three components carry the technological risk of the design and are the subject of a dedicated qualification program:

- The electromagnetic primary pump, which replaces the mechanical pumps used in previous French sodium fast reactors. Manufacturability, electromagnetic behavior and sodium flow performance are being tested on full-scale prototypes in a sodium loop; PEMDYN, the CEA electromagnetic pump test facility at Cadarache, provides the reference experimental basis for the OTRERA pump
- The compact heat exchanger, in its sodium-to-sodium version for OTRERA ONE and in its sodium-to-nitrogen version for OTRERA 300. Thermal performance, mechanical behavior under thermal transients, leak detection and maintainability are tested at full scale under representative operating conditions;
- The control rod drive mechanisms, tested in a sodium loop in their series-production configuration, to qualify the reactivity control and shutdown functions.

The program is conducted at increasing levels of representativeness. Scaled water models characterize and validate the sodium flow distribution and the thermal-hydraulic behavior of the reactor block. Full-scale prototypes are then tested on sodium loops under representative temperature and flow conditions. Experimental results are fed back into the design so that assumptions and numerical models are validated against measurement rather than against calculation alone.

The objective of the program is to bring the components specific to the OTRERA architecture to the maturity required before the construction of OTRERA ONE, the remaining maturity levels being reachable only in a reactor.

10. Design and Licensing Status

OTRERA is in Stage 3 of the French regulatory process, the pre-licensing review of the fundamental safety options. On 30 July 2026, OTRERA submitted to the French Nuclear Safety and Radiation Protection Authority (ASNR) a formal request for an opinion on selected fundamental safety options adopted for the OTRERA ONE project. This submission marks the entry of the project into the formal pre-licensing process.

The safety approach, the core physics, the MOX fuel and the sodium technology are supported by the operation of Rapsodie, Phénix and Superphénix and by the associated regulatory precedent. The work remaining before deployment is industrial and regulatory rather than scientific. OTRERA ONE is therefore developed as an industrial precursor and not as an experimental reactor, its qualification programme being limited to the components specific to the OTRERA architecture, as described in section 9.

Following the opinion of ASNR on the fundamental safety options, OTRERA will complete the detailed design and the associated safety demonstration. The next regulatory stage is Stage 4, the assessment of the application for authorization to create (Demande d'Autorisation de Création). This stage covers the full technical assessment of the design, the assessment of the proposed site, and the assessment of the organizational and operational capabilities of the applicant.

11. Fuel Cycle Approach

The OTRERA fuel cycle relies on facilities that are already in industrial operation and requires no new plant to be built.

Spent subassemblies decay first in the peripheral storage positions of the reactor vessel, then in the on-site storage drum described in section 5. They are then washed, packaged and transported to the Orano La Hague plant, where uranium and plutonium are separated from the fission products and the minor actinides. The recovered plutonium, the reprocessed uranium and the depleted uranium available from the front end of the existing fleet are used to fabricate new fuel. No fissile material is sent to waste.

This route is what makes fast reactors a resource technology rather than only a waste technology. A fast spectrum fissions the fertile uranium 238 that a thermal spectrum leaves unused, so an SFR fleet extracts from a given quantity

of uranium roughly two orders of magnitude more energy than a water-cooled fleet, and does so from depleted and reprocessed uranium rather than from newly mined and enriched material. France already holds several hundred thousand tonnes of depleted uranium, which is enough to supply a fast reactor fleet for centuries without any further mining.

The core is also compatible with HALEU oxide below 20% U235. This second option is not a fallback: it allows deployment on the target schedule and in markets where recycled material is not available, using the same assembly design and the same core geometry. The transition from HALEU to MOX requires no modification of the reactor.

12.Waste Management and Disposal Plan

The OTRERA waste management approach relies on existing industrial routes and on the French national framework for radioactive materials and waste (Act of 28 June 2006 and successive PNGMDR plans). Each stream is directed to an operating or licensed national facility, and no new back-end route has to be created for the deployment of the reactor. To this end, OTRERA fuel assemblies are designed for compatibility with the shearing and dissolution route of the existing treatment plant.

Operational waste is limited by design. The primary system is non pressurized and uses no boron or chemical shim, so the plant does not produce the borated effluents and evaporator concentrates that dominate the operational waste of a water-cooled reactor. Sodium purification media (cold traps) and cover gas treatment media are conditioned as low or intermediate level waste. Liquid effluents are treated online on the installation and then transferred to a centralized hot facility for treatment.

Specialized maintenance operations are carried out periodically on the sodium components of the reactor blocks after drainage of sodium and transfer to a decentralized hot facility. Residual sodium is eliminated by controlled hydrolysis, using the process qualified by CEA and implemented at industrial scale for the 5 500 t of Superphénix sodium and for the 1 500 t of Phénix sodium in the NOAH facility at Marcoule. Sodium is converted into sodium hydroxide, which is immobilized in a cement matrix and managed as very low- or low-level waste. The compact loop architecture of OTRERA reduces the sodium inventory per unit, well below the pool type reactors of the previous generation.

Spent fuel is MOX oxide fuel with no sodium bonding and is therefore directly compatible with the existing aqueous treatment route without any dedicated preconditioning step. Core management is by single or dual batch over a 7-to-10-year cycle, and with an average discharge burnup of 100 GWd/t. After decay in the peripheral storage positions and in the on-site storage drum, spent fuel is transported to the Orano La Hague plant. This route is not a projection: more than 500 Phénix subassemblies, about four and a half cores, were reprocessed at the Marcoule pilot plant and at La Hague, 3.3 t of the recovered plutonium was refabricated into Phénix fuel, and several subassemblies completed three successive cycles of reprocessing, refabrication and irradiation. Moreover, OTRERA fuel design is already approaching full compatibility with the existing reprocessing routes of ORANO used for Ex-PWR MOX currently operated in France by EDF.

Ultimate waste is consequently limited to fission products and minor actinides, vitrified in the standard canisters already produced by the French back end and destined to the Cigéo deep geological repository, currently under licensing. Because plutonium is fissioned instead of being degraded, multirecycling in a fast reactor generates about four times fewer minor actinides per unit of electricity than plutonium recycling in a water-cooled reactor. Heterogeneous transmutation of minor actinides, as an upside for backend management has been demonstrated in Phénix and studied for the ASTRID program but is not required by the OTRERA design.

Decommissioning is anticipated from the design stage. The specialized maintenance operations performed periodically during operation are of the same nature as dismantling operations and use the same facilities. The reactor block is compact and located below grade, which limits the volume of activated material at end of life. The inherent architecture with 6 heat extraction loops per unit simplifies D&D as well as calls for reusable D&D tools and means.

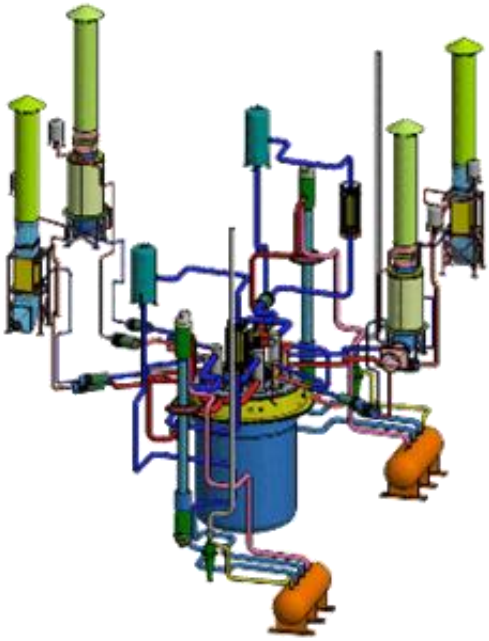
13.Development Milestones

Year	Milestone	Status
2022-2023	Preliminary studies and technological innovation (using previously developed patents).	**Completed**
2023-2024	Pre-conceptual design phase and technology validation	**Completed**
2024-2025	Conceptual Design Phase (and preparation for pre-licensing)	**Completed**
2026-2027	Basic Design Phase	**Ongoing**
2027-2030	Detailed Design Phase and Site-specific Licensing	**Planned**

Year	Milestone	Status
2030-2033	Construction of FOAK plant	Planned
2034	Commissioning and first criticality of FOAK	Planned

NOAK studies expected to start in 2028 for a firs concrete by 2036.

KAERI, Republic of Korea
SALUS-100



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*KAERI, Republic of Korea*
Reactor type	*LMFR*
Coolant/moderator	*Na / N/A*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*267 / 100*
NSSS operating pressure (primary/secondary), MPa	*0.12 / 1.2*
Core inlet/outlet coolant temperature (°C)	*360 / 510*
Primary circulation	*Forced circulation*
Fuel type/assembly array	*U-Zr with FC92 cladding*
Number of fuel assemblies in the core	*112*
Fuel enrichment (%)	*≤ 19.0*
Core discharge burnup (GWd/ton)	*73.0*
Refuelling cycle (months)	*240*
Fuel cycle requirements/approach	*TBD*
Reactivity control mechanism	*Control rods*
Approach to safety systems	*Combined active and passive system*
Seismic design (SSE)	*0.3*
Design life (years)	*60*
Plant footprint (m^2)	*TBD*
RV height/diameter (m)	*16.6 / 10.5*
RV weight (metric ton)	*239.0*

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1. Introduction

SALUS-100, currently under development by KAERI, is an innovative small modular reactor (SMR) design based on the sodium-cooled reactor technology with fast neutron spectrum with an electrical output of 100 MWe. This innovative design, derived from the previous TRU transmutation burner SFR concept, represents a significant step forward in establishing a new model for SMRs and offers diverse applications, including on– and off-grid electricity generation, district heating, and industrial heat supply. The reactor features of SALUS-100 incorporate an integrated pool-type primary system configuration and a long fuel-cycle core concept with a metallic alloy nuclear fuel. Its design concept prioritizes preventing severe accidents by capitalizing on the inherent safety and reliability features achieved through emphasizing passive design characteristics.

The paper describes the engineering design and safety analyses of SALUS-100, highlighting advanced design concepts in reactor core, fuel, materials, and systems. Additionally, the key research and development endeavors, focusing on vital areas such as enhancing system design capability, validating designs, demonstrating safety, and innovating in metal fuel technology, are briefly discussed, emphasizing the potential of SALUS-100 in revolutionizing the SMR market and utilizing efficient nuclear energy in the future.

2. Target Application

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

3. Design Philosophy

SALUS-100 is a 100 MWe pool-type sodium-cooled fast reactor intended for electricity generation and heat supply. Its design philosophy combines a long-life U-10Zr metallic-fuel core, approximately 20 years of operation without refueling, and a 60-year plant design life. The reactor builds upon the established design methods and validation experience of the PGSFR, giving preference to proven technologies and requiring appropriate validation of advanced technologies.

Safety is based on defense in depth, with emphasis on preventing severe accidents through inherent and passive features. The low-pressure sodium coolant, high thermal inertia of the pool-type primary system, metallic-fuel characteristics, and negative reactivity feedback support autonomous responses to abnormal conditions. Passive safety is applied as far as practicable to reduce dependence on electrical power and operator action. Simplicity, adequate design margins, redundancy, fail-safe characteristics, maintainability, and minimized radioactive releases are incorporated to enhance safety, reliability, operability, and economic performance.

4. Main Design Features

(a) Nuclear Steam Supply System

The SALUS-100 primary heat transport system (PHTS) places every component and its liquid-sodium coolant inside a single reactor vessel. By eliminating external primary piping, the design creates an integral pressure boundary, sharply reducing loss-of-coolant-accident risk and, thanks to the large sodium pool, damping temperature swings during transients. Core heat is carried upward to intermediate heat exchangers (IHXs), which in turn warm secondary sodium for power conversion; decay heat can still be removed by natural circulation because the IHXs sit above the core. Vertical sodium pumps supply forced flow in normal operation and include flywheels that coast down long enough to keep core temperatures within limits after a power loss.

A surrounding containment vessel provides reserve volume to hold any leaked sodium while keeping the core covered and the heat-removal path intact. Separate decay-heat exchangers (DHXs) immersed in the cold pool transfer residual heat to sodium-to-air coolers outside the vessel, giving a passive, independent sink. Sodium levels and internal baffles are arranged to prevent gas entrainment, protect pump NPSH, and preserve flow paths to both IHXs and DHXs under all conditions. In short, the integral pool-type layout, elevated IHXs, passive DHXs, and protected pump design collectively deliver robust heat removal and inherent safety with minimal external interfaces.

(b) Reactor Core

The SALUS-100 core is a pool-type, once-through driver adapted from the PGSFR but re-engineered to run 20 years without shuffling. Cycle extension is achieved by (i) lowering rated power to 267 MWth (which halves power density), (ii) tripling heavy-metal inventory, and (iii) fostering in-situ breeding with zoned enrichment. Each fuel assembly now holds 169 thicker metallic pins (U-10 wt % Zr, 1.45 m long) on a wider pitch (~+32 %), while the overall radial envelope of the PGSFR (Ø ≈ 2.1 m) is retained so pumps and IHXs need no relocation.

The active core is divided into inner / middle / outer zones enriched at 9.0 / 12.7 / 19.0 wt % U-235; the inner zone also uses axial enrichment zoning to speed fissile build-up. Pins are clad in FC92 with a chromium diffusion barrier, bonded by sodium, kept at 75 % smear density, and equipped with an enlarged 2.05 m plenum to hold fission gas over two decades. Surrounding the 112 fuel assemblies are 78 reflectors, 54 steel shields and nine control assemblies, giving an active height of 1.45 m.

Monte-Carlo depletion analyses with McCARD, benchmarked to BFS-84-1 metal-fuel experiments, predict a reactivity swing under 500 pcm (< 1 $) for the full 6 840 EFPD (≈ 90 % CF) mission. Average / peak linear powers fall to 88 / 150 W cm^{-1}, and the average discharge burn-up reaches 73 MWd kg^{-1} (local peak 108 MWd kg^{-1}), preserving ample thermal, mechanical and safety margins throughout the 20-year campaign.

(c) Reactivity Control

The SALUS-100 reactor relies on inherent nuclear reactivity feedback mechanisms to self-regulate reactor power and ensure core stability. By utilizing a U-10wt%Zr metallic alloy fuel with high thermal conductivity, the operating fuel temperature remains relatively low during normal operation, which results in a smaller fuel temperature reactivity coefficient and less rapid reactivity changes during transients. When fuel or coolant temperatures rise, prompt negative reactivity feedback is immediately induced primarily through the Doppler effect and fuel axial expansion. Furthermore, net negative reactivity is maintained through a combination of cladding axial expansion, sodium coolant density changes, core radial expansion, and the thermal expansion of control rod drivelines that inserts control rods deeper into the core.

Active reactivity control and emergency shutdown functions are provided by two independent, diverse shutdown systems comprising a total of nine boron carbide (B4C) control rod assemblies clad in Type 316 stainless steel. The primary reactivity control system utilizes six control rod assemblies driven by electric servomotors and ball-screw mechanisms to handle fine axial positioning for reactor power regulation, startup, and burnup compensation. Upon an emergency shutdown signal or loss of electrical power, the electromagnets holding the primary control drivelines de-energize, allowing the control assemblies to drop freely into the core under gravity as a fail-safe measure. The secondary reactivity control system consists of three control rod assemblies that provide redundant shutdown capability and feature a passive Self-Actuated Shutdown System (SASS). The SASS employs a bimetallic thermal expansion device that automatically unlatches the secondary control rods when the core outlet coolant temperature exceeds a designated limit, dropping them by gravity to prevent sodium boiling even if active systems fail.

Normal operational reactivity is managed by the Reactor Regulating System (RRS) and the Digital Rod Control System (DRCS), while safety protection is governed by the Plant Protection System (PPS) alongside an independent Diverse Protection System (DPS) designed to mitigate Anticipated Transients Without Scram (ATWS) and common cause failures. Mechanical safety features strictly limit reactivity insertion by allowing only one control rod assembly to be withdrawn at a time at controlled speeds, supported by a Rod Stop System (RSS) to prevent rod ejection accidents. Additionally, a lower damper at the bottom of the control assembly reduces the drop velocity at the end of a scram to mitigate mechanical impact and preserve structural integrity.

(d) Reactor Vessel (RV) and Internals

The SALUS-100 reactor vessel is a Type 316 stainless steel structure that contains the primary sodium coolant and cover gas, forms the primary coolant boundary, and supports the reactor core and reactor internal structures. The vessel is approximately 10.5 m in diameter, 16.6 m high, and 50 mm thick, with a design temperature of 520°C and a design pressure of 0.2 MPa. Its upper surface is welded to the lower surface of the reactor head, while the loads of the core and internal structures are transferred to a forged core-support flange in the lower vessel head. Except for this flange and temporary shipping restraints, the vessel has no attachments or penetrations. It is surrounded by a containment vessel, and the annular space between the two vessels is filled with nitrogen and continuously monitored for sodium leakage. The spacing is selected so that, even if sodium leaks from the reactor vessel, the remaining coolant level is sufficient to keep the core covered. The same annular space also provides access for remote in-service inspection of the vessel.

The reactor internal structures support the core, instrumentation, in-vessel equipment, piping, fuel-transfer equipment, and stored fuel assemblies. They also establish coolant-flow paths for both forced and natural circulation and distribute sodium to the individual core assemblies. Their principal components include the core support structure, inlet plenum, core shroud, former ring, inner vessel, fuel transfer port, core shielding structures, inlet pipes, upper internal structure, and upper shielding structure. The inlet plenum distributes low-temperature sodium uniformly to the core, while the core support structure and former ring maintain the vertical and horizontal positions of the fuel assemblies. The inner vessel separates the hot and cold sodium pools and reduces thermal gradients during transients. Radial and upper shielding structures containing B4C limit neutron irradiation of the reactor vessel and internal structures and reduce activation of the intermediate sodium. The vessel and internal structures are designed for a 60-year service life and are evaluated for pressure, temperature, mechanical, seismic, irradiation, and sodium-environment effects.

(e) Reactor Coolant System and Steam Generator

The SALUS-100 reactor coolant system consists of the Primary Heat Transport System (PHTS) and the Intermediate Heat Transport System (IHTS). The pool-type PHTS contains the radioactive primary sodium and its major components within the reactor vessel, thereby eliminating large external primary piping and reducing the potential for a loss-of-coolant accident. At full power, the PHTS removes 267 MWt from the core while maintaining core inlet and outlet temperatures of 360°C and 510°C, respectively. Two mechanical primary sodium pumps circulate the coolant through the core and four intermediate heat exchangers (IHXs). Each pump discharges through two inlet pipes connected to the inlet plenum and includes a flywheel to provide coastdown flow following a loss of pump power. The IHXs are shell-and-tube, counter-flow heat exchangers that transfer heat from the radioactive primary sodium to the

non-radioactive intermediate sodium. Their thermal centers are positioned above the reactor core to support decay-heat removal by natural circulation when the primary pumps are unavailable.

The IHTS comprises two identical loops arranged symmetrically around the reactor vessel. Each loop contains two IHXs, one annular linear induction pump, one expansion tank, and one steam generator. At full power, each loop circulates approximately 649.6 kg/s of sodium between hot- and cold-leg temperatures of 482°C and 322°C. The intermediate sodium pressure is maintained above the primary sodium pressure across the IHX tubes so that any tube leakage is directed from the IHTS toward the PHTS. Two steam generators, one for each IHTS loop, are installed in an auxiliary building outside the reactor building. Each unit is a vertical shell-and-tube, once-through, counter-flow heat exchanger with straight tubes and a heat-transfer capacity of 133.3 MWt. Intermediate sodium flows on the shell side, while feedwater enters the lower header and flows upward inside 769 tubes, becoming superheated steam before reaching the upper steam header. The steam generator includes tube-support plates, flow-distribution structures, feedwater orifices to suppress flow instability, and a bellows to accommodate differential thermal expansion. A dedicated sodium-water reaction pressure relief system protects the IHTS and reactor coolant boundary against excessive pressure resulting from a steam-generator tube leak.

(f) Pressuriser

N/A – It is not required for SALUS-100

(g) Primary pumps

SALUS-100 uses two vertical, single-stage centrifugal mechanical pumps to circulate primary sodium through the reactor core and intermediate heat exchangers. The pumps are arranged symmetrically in the cold pool between the inner vessel and reactor vessel and are supported by the reactor head. Each pump outlet branches into two internal pipes connected to the core inlet plenum. At the design point, each pump has a rated capacity of 49.68 m^3/min, a developed head of 23.37 m, and a rotational speed of 517 rpm, with a sodium inlet temperature of 360 °C. Because the pumps operate near atmospheric pressure, their elevation and surrounding flow passages are designed to provide adequate net positive suction head and prevent cavitation and vortex inflow. Each pump is equipped with a flywheel to maintain coastdown flow following loss of power. A pony motor, rated at approximately 10% of the main-motor capacity, provides about 10% of rated flow for low-flow operation. Pump speed, motor condition, and vibration are continuously monitored. The pumps are not credited with a safety function after an accident, because post-shutdown core cooling is provided by natural sodium circulation in conjunction with the decay heat removal system.

(h) Distinguishing features

SALUS-100 is a 100 MWe sodium-cooled fast reactor distinguished by its long fuel cycle, integrated pool-type configuration, metallic fuel, and emphasis on inherent and passive safety. Its U-10Zr-fueled core is designed for approximately 20 years of operation without refueling or fuel-assembly shuffling. The long cycle is achieved by reducing the core power density, increasing the heavy-metal loading, and applying three radial enrichment zones. The reactor is intended for multipurpose use, including on-grid and off-grid electricity generation, district heating, and industrial heat supply.

The reactor core, four intermediate heat exchangers, two primary sodium pumps, and primary coolant are contained within the reactor vessel. This arrangement eliminates large external primary piping, minimizes the possibility of coolant loss from a primary pipe break, and provides substantial thermal inertia. An inner vessel separates the hot and cold sodium pools, while a surrounding containment vessel retains leaked sodium at a level sufficient to keep the core covered. The metallic fuel provides high thermal conductivity and negative reactivity feedback during temperature increases. Decay heat is removed by four independent trains comprising two passive and two active systems, with the passive trains relying on the natural circulation of sodium and air. SALUS-100 also incorporates a passive self-actuated shutdown system and is designed to maintain a safe state for at least 72 hours following a station blackout without operator action. Its design largely retains proven PGSFR technologies and validation methods while adapting the core, fuel, power level, and operating conditions for long-cycle power generation.

5. Safety Features

(i) Engineered Safety System Approach and Configuration

The engineered safety system approach of SALUS-100 follows defense in depth, giving priority to accident prevention while providing independent measures to control design-basis accidents and mitigate severe-accident consequences. The principal engineered safety feature is the safety-grade Decay Heat Removal System (DHRS), which removes core decay heat when the normal heat-removal path through the intermediate heat transport system and steam generators is unavailable. The DHRS consists of four independent trains arranged for redundancy, diversity, and independence. Two trains are passive decay heat removal systems that use the natural circulation of sodium and the natural draft of air, while the other two are active systems that use forced airflow. The system is designed so that two

available trains can bring the reactor to a safe shutdown state within 72 hours after reactor trip and primary-pump shutdown.

Safety-system actuation is provided by the Plant Protection System, which comprises the Reactor Protection System and the Engineered Safety Feature Actuation System. When monitored plant parameters reach their setpoints, the protection system initiates reactor trip and DHRS actuation. The actuation signals are generated using two-out-of-four coincidence logic and transmitted to the Engineered Safety Features-Component Control System, whose redundant controllers issue commands to the safety components. A Diverse Protection System provides separate reactor-trip and decay-heat-removal actuation functions against common-cause failures. For severe-accident conditions, the fully passive Reactor Vault Cooling System removes heat from the containment-vessel wall by naturally induced airflow. A self-actuated shutdown system and the sodium-water reaction pressure relief system provide additional protection against failure to shut down and steam-generator tube leakage, respectively.

Accordingly, the safety-system configuration established for SALUS-100 is primarily defined at the reactor and nuclear steam supply system level. Plant-wide implementation of these safety functions was not developed through separate BOP and architect-engineering activities within the project; instead, the relevant plant interfaces and supporting provisions were intended to build upon the established PGSFR design basis.

(j) Decay Heat Removal System

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

For SALUS-100, conditions within the design basis are managed through a graded classification comprising anticipated operational occurrences and two classes of design-basis accidents. Events occurring at frequencies of at least 10^{-2} per reactor-year are classified as anticipated operational occurrences, while Design Basis Accident Classes 1 and 2 cover frequency ranges from 10^{-2} to 10^{-4} and from 10^{-4} to 10^{-6} per reactor-year, respectively. Their consequences are evaluated using deterministic safety analysis with conservative assumptions and event-specific acceptance criteria for fuel integrity, cladding damage, core coolability, sodium boiling, structural integrity, containment leakage, and off-site dose.

Protective actions are initiated primarily by the safety-grade Plant Protection System, which consists of the Reactor Protection System and the Engineered Safety Feature Actuation System. The Reactor Protection System continuously monitors reactor power, coolant flow, and core temperatures and automatically inserts the control rods when a trip setpoint is reached. The actuation system initiates the required engineered safety features, including decay-heat removal, using two-out-of-four coincidence logic. A Diverse Protection System provides an independent reactor-trip and decay-heat-removal actuation path against anticipated transients without scram and common-cause failures, although safety analyses for anticipated operational occurrences and design-basis accidents credit the Plant Protection System.

The principal engineered safety feature for design-basis accidents is the Decay Heat Removal System. It comprises two passive and two active trains that are physically and electrically independent. The passive trains remove heat through natural circulation of sodium and air, while the active trains use forced airflow. Even with a single failure, the system is designed to cool the reactor to a safe shutdown state within 72 hours while maintaining fuel and structural temperatures within their specified limits.

(m) Safety Approach and Configuration to Manage DEC

SALUS-100 treats events with estimated frequencies below 10^{-6} per reactor-year as Design Extension Conditions and classifies them according to whether significant fuel degradation or fuel melting occurs. The DEC approach supplements the design-basis provisions with inherent safety characteristics, passive systems, complementary design features, and accident-management measures intended to prevent accident progression and mitigate radiological consequences. The principal preventive features arise from the pool-type configuration and U-10Zr metallic fuel, including high thermal inertia, a large sodium-boiling margin, high fuel thermal conductivity, and negative reactivity feedback from fuel-temperature and structural-expansion effects.

Anticipated transients without scram are assessed through unprotected transient overpower, unprotected loss-of-flow, and unprotected loss-of-heat-sink events. These analyses examine whether inherent reactivity feedback and decay-heat removal can maintain a safe state when the normal reactor-trip function is unavailable. The secondary shutdown system also incorporates a self-actuated shutdown device that responds passively to an increase in core outlet temperature and releases control rods for gravity insertion.

Decay heat can be removed through the independent passive and active trains of the Decay Heat Removal System. For severe-accident conditions, the Reactor Vault Cooling System serves as a fully passive complementary feature. It transfers heat from the containment-vessel wall to the atmosphere through naturally induced airflow and requires no actuation signal. The DEC assessment also considers local core-flow blockage, station blackout with loss of passive

decay-heat removal and reactor-vault cooling, and a reactivity anomaly combined with failure to scram and station blackout. The severe-accident evaluation assumes in-vessel retention and credits the inherent behavior of metallic fuel in limiting further accident progression. At the documented design stage, parts of this evaluation were based on corresponding PGSFR metal-core analyses.

(n) Containment System

The SALUS-100 containment boundary comprises a steel containment vessel (CV) in the lower region and a liner-plate-clad concrete reactor building in the upper region. The CV, built of 2¼Cr-1Mo steel and bolted to the surrounding concrete, follows ASME B&PV Sec III Div 5 high-temperature rules because it must confine any sodium leaking from the RV at temperatures above 350 °C; an annular gap slightly larger than the RV diameter permits ROV in-service inspection.

The reactor building encloses all safety-related systems, its internal liner completing the containment boundary and blocking radioactive release. Designed to remain intact during design-basis and severe accidents—including aircraft impact—it shares a seismically isolated common basemat with the auxiliary building, limiting earthquake response and preserving leak-tightness under both normal and accident conditions.

(o) Spent Fuel Cooling Safety Approach / System

Spent fuel is initially stored inside the reactor vessel and cooled while submerged in primary sodium. Decay heat is removed through the normal heat transport path or the decay heat removal system until passive air cooling becomes feasible. The fuel is then cleaned, sealed in inert-gas canisters, and transferred to dry storage cooled by natural air circulation. Storage temperature and radioactivity are monitored. No separate SALUS-100-specific design work for the external spent-fuel cooling and storage system was performed within the project.

6. Plant Safety and Operational Performances

The SALUS-100 applies the standard defense-in-depth concept but, in light of Fukushima, gives heightened emphasis to preventing and mitigating severe accidents through maximised inherent and passive safety. As a sodium-cooled, metal-fuelled fast reactor, it adopts the inherent features typical of SFRs:

- Low-pressure liquid sodium with a wide boiling margin;
- A pool-type primary system whose high thermal inertia and atmospheric pressure eliminate LOCA;
- Strong negative reactivity feedback from Doppler, fuel axial expansion and axial expansion of control-rod drive lines;
- Retention of a coolable geometry because fuel damage begins at relatively low temperature.

The metal fuel itself adds a further margin. Its high thermal conductivity keeps fuel temperature and radial gradients low, so Doppler swings are modest and less control reactivity is required. The resulting slow, mild transients give operators a longer grace period, while the large power-to-melt margin below the 1 250 °C melting point enhances overall plant resilience.

7. Instrumentation and Control Systems

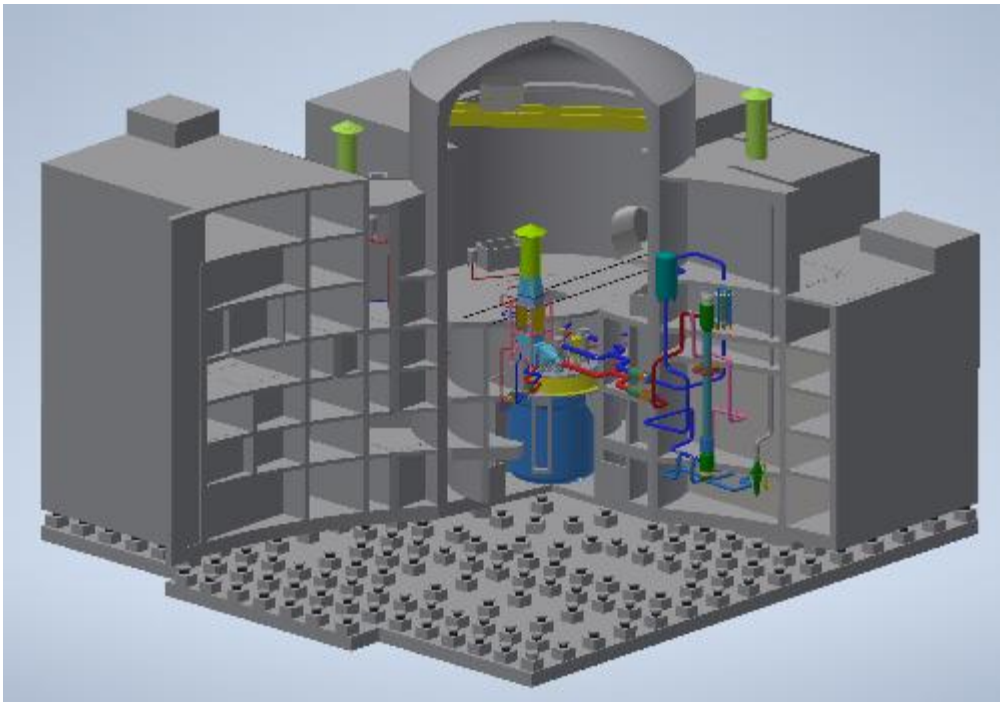
The SALUS-100 instrumentation and control architecture is divided into safety and non-safety systems. The principal safety I&C systems are the Plant Protection System, the Engineered Safety Features-Component Control System, and the qualified indication and alarm system. The Plant Protection System comprises the Reactor Protection System and the Engineered Safety Feature Actuation System. The Reactor Protection System continuously monitors reactor power, the power-to-primary-flow ratio, core inlet temperature, and central and individual fuel-assembly outlet temperatures. When a monitored variable reaches its prescribed setpoint, it automatically initiates reactor trip and provides status indication, alarms, self-diagnostics, bypass control, and periodic testing functions.

The Engineered Safety Feature Actuation System consists of four physically separated and electrically isolated channels. It generates initiation signals for decay-heat removal and containment isolation, while the component control system performs the sequencing and operation of the associated equipment. These protection systems are designed to remain functional under a single failure, maintain independence from non-safety systems, complete an initiated protective action, and enter a fail-safe condition following equipment or power failure. Environmental, seismic, and electromagnetic qualification are also required.

The non-safety I&C systems include plant control and monitoring systems, the Diverse Protection System, the Diverse Indication System, and the Automatic Seismic Trip System. The Diverse Protection System provides electrically isolated reactor-trip and decay-heat-removal actuation paths to address common-cause failures and anticipated transients without scram. Instrumentation and monitoring systems process measurements of temperature, pressure,

flow, sodium level, neutron flux, radiation, equipment condition, and sodium leakage. The Main Control Room integrates these data through operator consoles, safety displays, alarms, and soft controls to support normal operation, accident monitoring, and safe shutdown.

8. Plant Layout Arrangement



9. Testing Conducted for Design Verification and Validation

No separate verification testing was conducted specifically for SALUS-100 at the documented stage of development. Instead, its nuclear steam supply system design was established using design methods, validation programs, and experimental data previously developed for the PGSFR. These data included sodium heat-transfer tests for the intermediate, decay-heat, and sodium-to-air heat exchangers; flow-distribution and hydraulic-characteristic tests; and performance and coastdown tests of the primary sodium pump. The results had been used to verify component-design methods and thermal-hydraulic and safety-analysis codes. Accordingly, SALUS-100 design verification relied primarily on the established PGSFR experimental database and validated analytical capabilities rather than on a dedicated SALUS-100 test campaign.

10.Design and Licensing Status

SALUS-100 was developed to the conceptual design stage, covering the reactor core, nuclear steam supply system, major mechanical components, safety systems, and supporting safety analyses. Its design and safety documentation was subsequently assessed through the IAEA Technical Safety Review process, which was completed. The review addressed the SALUS-100 safety approach and its consideration of relevant IAEA safety requirements. Completion of the IAEA Technical Safety Review does not constitute national regulatory approval or a construction or operating licence. SALUS-100 should therefore be described as a conceptual reactor design with a completed IAEA Technical Safety Review, but without a national nuclear licence.

11.Fuel Cycle Approach

SALUS-100 adopts a 20-year once-through core management scheme using U-10Zr metallic fuel enriched below 20 wt.%. The core operates without refueling or fuel shuffling, after which all fuel assemblies are replaced. Discharged fuel is initially cooled inside the reactor vessel and then transferred to external storage. Long-term spent-fuel management, final disposal methods, and related policies have not yet been determined at the current conceptual design stage.

12.Waste Management and Disposal Plan

SALUS-100 minimizes radioactive waste generation and provides for the treatment, monitoring, storage, and controlled release of operational waste. The relevant systems were based on the PGSFR BOP design, while arrangements for long-term storage and final disposal have not yet been determined.

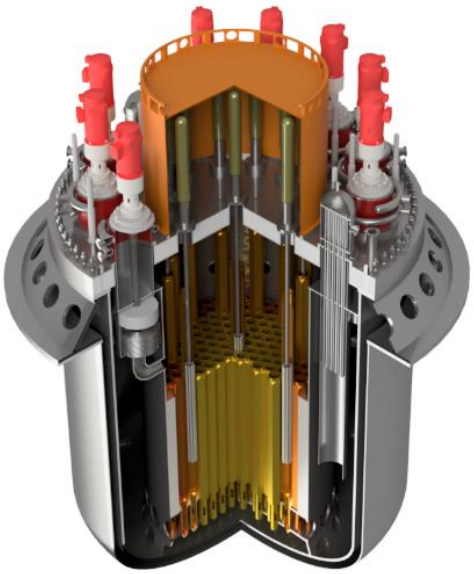
13.Development Milestones

Year	Milestone	Status
2024	Conceptual design stage	Ongoing
2029	Basic design stage (tentative)	Planned
2035	Receive standard design approval	Planned

Blykalla, Sweden
SEALER-55



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*Blykalla, Sweden*
Reactor type	*LMFR*
Coolant/moderator	*Lead / N/A*
Design status	*Conceptual design*
Thermal/electrical capacity, MW(t)/MW(e)	*140 MWt / 55 MWe*
NSSS operating pressure (primary/secondary), MPa	*Non-pressurised*
Core inlet/outlet coolant temperature (°C)	*420 / 550*
Primary circulation	*Forced*
Fuel type/assembly array	*UN/Hex-can*
Number of fuel assemblies in the core	*84*
Fuel enrichment (%)	*12*
Core discharge burnup (GWd/ton)	*60*
Refuelling cycle (months)	*300*
Fuel cycle requirements/approach	*TBD*
Reactivity control mechanism	*B4C, WB2*
Approach to safety systems	*Passive*
Seismic design (SSE)	*None*
Design life (years)	*28*
Plant footprint (m^2)	*Provide upper limit*
RV height/diameter (m)	*5.5 / 4.8*
RV weight (metric ton)	*20*

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

☒ Electricity
- ☒ Baseload
- ☐ Load-following

☒ District Heating / Steam

☒ Hydrogen
- ☐HTSE - electrolysis
- ☐PEM - polymer membrane

☐ Desalination

☒ Industrial process heat
- ☒ <200°C
- ☒ 200–550°C
- ☐ >550°C

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 Vessel (RV) 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 and Steam Generator

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.

Secondary System

Rankine steam cycle at 165 bar and T = 350-530°C.

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.

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.

(f) Pressuriser

N/A

(g) Primary pumps

N/A

(h) Distinguishing features

N/A

5. Safety Features

(i) 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.

(j) Decay Heat Removal System

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.

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

N/A

(m) Safety Approach and Configuration to Manage DEC

N/A

(n) 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.

(o) 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.

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 Systems

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 UF6, 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 UO2 pellets or direct disposal of UN fuel rod assemblies will be evaluated.

12.Waste Management and Disposal Plan

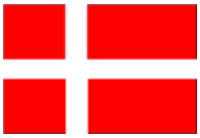
See 11.

13.Development Milestones

Year	Milestone	Status
2018	Concept design of SEALER-UK funded by UK BEIS	Completed
2020	SUNRISE project, including design of a demonstration unit funded by SSF.	Completed
2021	Blykalla and Uniper form joint venture SMR AB	Completed
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.	Completed
2024	The construction of the SEALER-E electrical mockup starts.	Ongoing
2025	SEALER-E is commissioned.	Ongoing
2030	Supporting documentation for the licensing of the SEALER-One nuclear demonstration unit is completed	Planned

MOLTEN SALT SMALL MODULAR REACTORS

Copenhagen Atomics, Denmark
CA Waste Burner



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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
Design status	Detailed design / Equipment manufacturing in progress
Thermal/electrical capacity, MW(t)/MW(e)	100 / N/A
NSSS operating pressure (primary/secondary), MPa	0.05 – 0.25 / 0.1 – 0.25
Core inlet/outlet coolant temperature (°C)	600 / 650 - 700
Primary circulation	Forced circulation
Fuel type/assembly array	7LiF-UF4 or 7LiF-ThF4-(TRU)F3 / none
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	LEU or TRU/RGPu
Core discharge burnup (GWd/ton)	900 – 1000
Refuelling cycle (months)	N/A
Fuel cycle requirements/approach	LEU or transuranic initiated / conversion to Th-U cycle
Reactivity control mechanism	Heavy water level adjustment
Approach to safety systems	Passive
Seismic design (SSE)	N/A
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 (m^2)	50 000 per 2.5GW(th) plant
RV height/diameter (m)	12 / 2.55
RV weight (metric ton)	40

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

☐ Electricity	☐ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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) Nuclear Steam Supply System

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 7LiF-UF4 or 7LiF-ThF4-(TRU)F3 kickstarter fuel salt, 7LiF-ThF4 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

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.

(d) Reactor Vessel (RV) and Internals

(e) Reactor Coolant System and Steam Generator

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

(i) 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.

(j) Decay Heat Removal 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.

(k) Emergency Core Cooling System

(l) Safety Approach and Configuration to Manage DBC

(m) Safety Approach and Configuration to Manage DEC

(n) 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.

(o) 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 Systems

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 m2 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 m2 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-UF4 fuel salt with 27% mol low uranium or 7LiF-ThF4-(TRU)F3 fuel salt with or 3% mol (TRU)F3 or RGPuF3 kickstarter fuel salt and 7LiF-ThF4 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 233U 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.

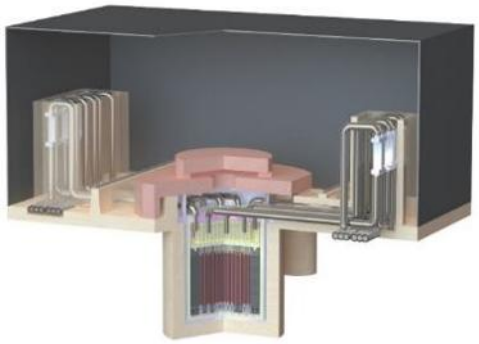
13.Development Milestones

Year	Milestone	Status
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.	Completed
2015	First simulations of neutron economy and online fission product removal.	Completed
2016	First static molten salt test.	Completed
2017	First pressure driven circulation molten salt loop.	Completed
2018	First pumped circulation molten salt loop.	Completed
2019	Award two Danish research grants and start of collaboration with Alfa Laval.	Completed
2020	First public funding round and first fertile salt test.	Completed
2022	Completion of non-fission prototype of a 1MW(t) demonstration reactor.	Completed
2023	Moved to 11,000 m2 production facility.	Completed
2024	Started collaboration to build and license first test reactor.	Completed
2026	Planned first 1 MW(t) test reactor.	Planned
2028	Planned first 100 MW(t) commercial reactor.	Planned

MoltexFLEX, United Kingdom
FLEX Reactor



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	MoltexFLEX, United Kingdom
Reactor type	Molten Salt Reactor
Coolant/moderator	Molten eutectic AlF3/NaF coolant salt / commercial grade graphite moderator
Design status	Pre-conceptual design
Thermal/electrical capacity, MW(t)/MW(e)	Single unit: 60 MWth/ 24 MWe, deployable in arrays
NSSS operating pressure (primary/secondary), MPa	Primary coolant unpressurised (~0.1 MPa) Note secondary loop is GridReserve®
Core inlet/outlet coolant temperature (°C)	725 / 795
Primary circulation	Natural convection loop
Fuel type/assembly array	Molten salt fuel within vented fuel tubes hexagonal array
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	5
Core discharge burnup (GWd/ton)	30-35
Refuelling cycle (months)	36-60
Fuel cycle requirements/approach	3 batch refuelling cycle with 3-5 year cycles, zoned refuelling pattern
Reactivity control mechanism	Strong fuel temperature coefficient, liquid neutron absorber thermometer
Approach to safety systems	Hazard elimination, inherent/passive safety
Seismic design (SSE)	EUR standard - peak ground acceleration of 0.3g horizontally and vertically
Design life (years)	60
Plant footprint (m^2)	1 unit ~450, 20 units (0.5 GWe) ~ 10,000
RV height/diameter (m)	7.5 / 6 (low pressure)
RV weight (metric ton)	Empty ~100 / Filled ~500

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 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

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☒HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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) Nuclear Steam Supply System

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.

Fuel Characteristics

The fuel is 5% enriched uranium within a mixture of UF3, UF4 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.

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 135Cs, a long-lived fission product, to contaminate the coolant salt.

(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 (RV) 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) Reactor Coolant System and Steam Generator

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) Pressuriser

(g) Primary pumps

(h) Distinguishing features

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

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) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) 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 Systems

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



Figure 1. Artist's impression of an array of 32 modules configured in line with turbine hall, GridReserve® storage facility and sub-station

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.

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.

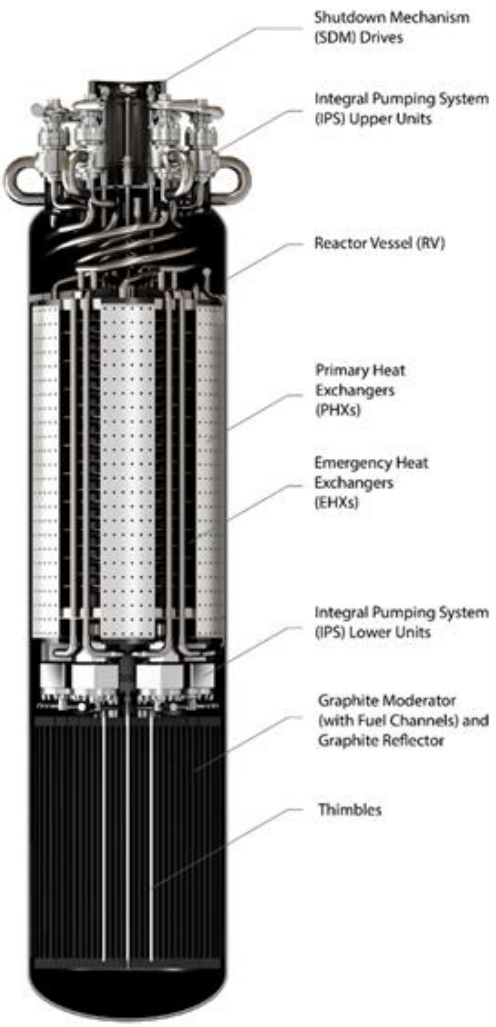
13. Development Milestones

Year	Milestone	Status
2020 – Q2	Initial concept defined	**Completed**
2021 – Q1	Initial Safety Assessment Report Completed	**Completed**
2023 – Q2	Initial concept design matured, progress made towards concept design	**Ongoing**
2024 – Q3	Independent reviews conducted by 3x UK nuclear experts	**Planned**
2025 – Q4	TRL3 Fully Achieved	**Planned**
2027 – Q3	TRL4 Achieved	**Planned**
T* +2 years	Concept design achieved (* T = date when significant funding can be raised)	**Planned**

Terrestrial Energy Inc., Canada/USA IMSR400



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Terrestrial Energy, Canada/USA
Reactor type	Molten Salt Reactor
Coolant/moderator	Molten salt / graphite
Design status	Detailed Design
Thermal/electrical capacity, MW(t)/MW(e)	442 / 195 per operating Core-unit, 884 / 390 dual Core-unit reference configuration
NSSS operating pressure (primary/secondary), MPa	0.4 / 0.45
Core inlet/outlet coolant temperature (°C)	610 / 700
Primary circulation	Forced
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
Fuel cycle requirements/approach	N/A
Reactivity control mechanism	Negative reactivity coefficient
Approach to safety systems	Passive
Seismic design (SSE)	0.3g
Design life (years)	56
Plant footprint (m^2)	35,200
RV height/diameter (m)	18/4.1
RV weight (metric ton)	385

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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

The IMSR plant is designed to serve electric power grids and industrial heat and power application as represented in Figure 2.

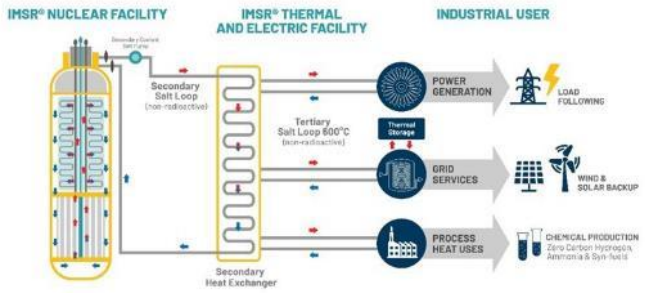


Figure 1. Schematic of the IMSR Core-unit and its Applications

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 Design 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 Vessel (RV) 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.

(h) Distinguishing features

Core-unit replaced completely as a single unit every 7 years; Fuel salt is reused from one unit to the next.

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.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) 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.

(e) 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.

(f) 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.

(g) 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



Figure 2. Dual IMSR Nuclear Facility Representation

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.

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 UF4 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.

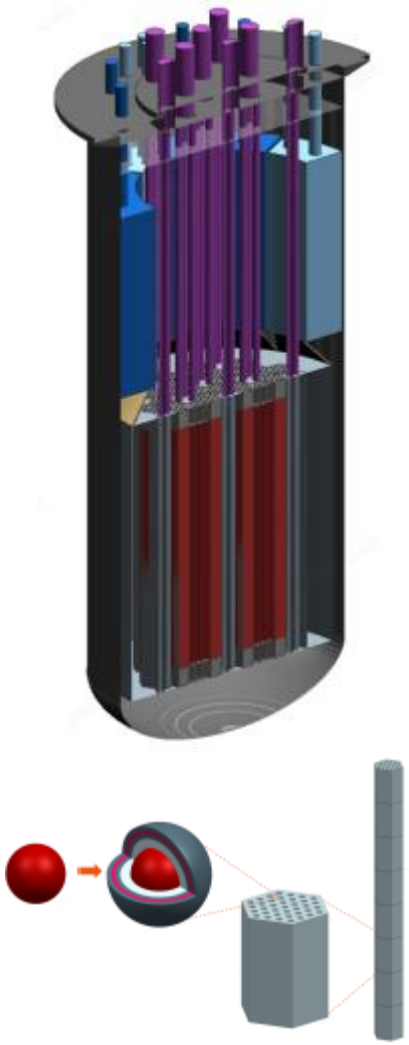
13. Development Milestones

Year	Milestone	Status
2015	Conceptual design completed	**Completed**
2016	Start of basic engineering phase	**Completed**
2017	Completion of CNSC pre-licensing Phase 1 Vendor Design Review	**Completed**
2017	US DOE LPO for up to $890Mn project financing support for first US IMSR Plant	**Completed**
2018	Commenced CNSC pre-licensing Phase 2 Vendor Design Review	**Completed**
2019	Commenced irradiation test program on graphite	**Completed**
2019	US NRC and CNSC conduct joint review of key safety analysis	**Completed**
2020	IMSR Plant short-listed to final three for final evaluations at DNNP	**Completed**
2023	CNSC VDR Phase 2 completed with no fundamental barriers to licensing identified	**Completed**
2025	Planned – Commence licensing at first site in North America	**Planned**
2027	Planned - Commence construction of a first full-scale IMSR	**Planned**
2033	Planned - First IMSR plant in-service	**Planned**

Jereh SMR Energy Holding Ltd, U.A.E. JFHR



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Jereh SMR Energy Holding Ltd., U.A.E.
Reactor type	MSR
Coolant/moderator	FLiBe / graphite
Design status	Conceptual design
Thermal/electrical capacity, MW(t)/MW(e)	442 / 195 per operating Core-unit, 884 / 390 dual Core-unit reference configuration
NSSS operating pressure (primary/secondary), MPa	125 / 62
Core inlet/outlet coolant temperature (°C)	0.1 / 0.15
Primary circulation	650 / 700
Fuel type/assembly array	Forced circulation
Number of fuel assemblies in the core	TRISO in prismatic blocks
Fuel enrichment (%)	42
Core discharge burnup (GWd/ton)	19.75
Refuelling cycle (months)	< 90
Fuel cycle requirements/approach	≥ 36
Reactivity control mechanism	Once-through and take-back
Approach to safety systems	Control rods (B4C)
Seismic design (SSE)	Fully passive
Design life (years)	0.3g
Plant footprint (m^2)	60
RV height/diameter (m)	8,000
RV weight (metric ton)	10.35 / 4.5

1. Introduction

JFHR is a fluoride-salt-cooled high-temperature reactor using TRISO fuel in prismatic form. The reactor module adopts an integrated modular arrangement, where the Primary Heat Exchangers (PHXs), the Passive Residual Heat Removal (PRHR) exchangers, and the primary pumps are all integrated into the reactor vessel. A supercritical carbon dioxide (sCO2) Brayton cycle is preferentially adopted, with air cooling serving as the cold-end heat transfer method. To maximize on redundancy and passive safety, JFHR employs a four-train configuration for the PHXs, supplemented by an additional four-train configuration of smaller-capacity PRHR exchangers operating independently and ensuring reliable heat transport under accident conditions.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☒PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☒ >550°C

3. Design Philosophy

JHFR leverages the low operation pressure of molten Flibe coolant without generating steam or hydrogen, the high efficiency of sCO2 power conversion cycle, the inherent safety features of TRISO fuel in prismatic form and PRHR system.

4. Main Design Features

(a) Nuclear Steam Supply System

The power generation technology leverages the sCO2 power conversion system to achieve thermodynamic and volumetric efficiency. By leveraging sCO2 technology, the power conversion system is made much smaller than the traditional nuclear power conversion cycle. Combining with the high outlet temperature of Flibe, the power conversion system can eventually provide a high efficiency of electricity production.

(b) Reactor Core

The reactor core contains 42 hexagonal assemblies, 13 reactivity control assemblies and replaceable graphite reflectors. The effective core's diameter and height are 3.3 m and 4 m respectively.

(c) Reactivity Control

The reactivity control system consists of 13 reactivity control assemblies which contain three control rods each. The reactivity control during normal operation is achieved by several sets of active control rod inserted in reactivity control assemblies. For shutdown scenario, reactor maintains sub-criticality by passively inserted shutdown rod.

(d) Reactor Vessel (RV) and Internals

The JFHR reactor vessel is designed at near-atmospheric pressure level. The vessel houses the prismatic TRISO fuel columns, PHXs, PRHR exchangers, and primary pumps in an integrated modular layout, which minimizes the risk of the large-diameter penetrations for nozzles. Due to the negligible internal pressure, the containment structure can be simplified. The key advantages include: (1) significantly reduced vessel wall thickness and weight, lowering manufacturing cost; (2) reduction of stress corrosion cracking concerns; (3) inherent leakage resistance - small leaks do not escalate. The internals utilize graphite block stacking, allowing easy on-site assembly and replacement.

(e) Reactor Coolant System and Steam Generator

The reactor coolant system has an intermediate-loop configuration, with the primary and intermediate loops utilizing Flibe and FliNaK as the coolant medium respectively. The tertiary loop employs air as the ultimate heat sink, a design choice inspired by arid-region deployment considerations. The PHXs are mounted at the top plenum of the reactor vessel. During normal operation, three of the four PHXs are sufficient to transport the heat from the core to the intermediate loops.

(f) Pressuriser

(g) Primary pumps

JFHR has four main pumps in the primary circuit. The technical specifications of pumps on irradiation shielding and anti-corrosion is less stringent compared to the liquid-fueled MSR.

(h) Distinguishing features

5. Safety Features

(a) Engineered Safety System Approach and Configuration

JFHR is equipped with four independent PRHR loops. Any three of these loops is capable of removing 0.75% of nominal fission power, with the fourth loop serving as redundancy. The four PRHR exchangers are placed at the top plenum of the integrated reactor vessel, allowing residual heat to be transported from the core to the intermediate loop via natural circulation, and then safely transferred to the ultimate heat sink through four passive air-cooled towers. This is suitable in remote areas and water-scarce regions.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

JFHR has identified a complete list of Postulated Initiating Events (PIEs), which will go through Probabilistic Risk Assessment (PRA) where they are categorized by frequency, and adopts defense-in-depth configuration, inherent safety as well as passive safety for all DBCs. The safety configuration is built upon the inherent resilience of TRISO fuel and the additional retention capabilities of Flibe coolant, providing multiple layers of defense without relying on high-pressure containment or active, electrically powered systems. The integrated reactor vessel design, minimizes large-diameter piping and reduces potential leakage paths. This configuration ensures that safety limits remain within acceptance criteria without active safety-grade systems, significantly simplifying licensing and enhancing overall plant robustness.

(e) Safety Approach and Configuration to Manage DEC

The DECs, including Station Black-Out (SBO) and multiple-failure events, are addressed through a defense-in-depth strategy characterized by additional passive safety features beyond the DBCs. For DEC-A: the reactor is protected by gravity-driven safety rod insertion upon SBO, requiring no operator action. The four-train dedicated PRHR system, independent of the PHXs, is actuated by air natural circulation of the tertiary loops, continuously removing decay heat. With the reactor vessel operating at near-atmospheric pressure level, this eliminates high-pressure-driven failure modes. The TRISO fuel particles tolerate temperatures up to 1600℃, providing substantial margin against fuel failure even under prolonged DEC transients. Although the likelihood of DEC-B occurrence is extremely low due to inherent TRISO robustness, there are safety provisions such as core catcher to ensure coolable debris retention within the reactor vessel. All DEC management features are passive, diversified, ensuring that no active safety-grade equipment is relied upon.

(f) Containment System

JFHR's containment system is fundamentally different from traditional nuclear plants. Instead of a massive, high-pressure concrete and steel dome, it adopts the functional containment concept. Reactor vessel, molten salt, prismatic graphite matrix and TRISO particles serve as multiple barriers in JFHR defensive-in-depth design. The robustness of TRISO makes it a perfect candidate for fission product retention, ensuring the health and safety of both workers and the public. Additionally, a coolant chemical control system serves as a monitor of contamination in molten salt coolant, in order to protect the integrity of fuel assemblies.

(g) Spent Fuel Cooling Safety Approach / System

JFHR is equipped with a spent fuel storage system functionally designed for cooling and storage. This system comprises a cooling pool and a storage vault. After being discharged from the core, the spent fuel is loaded into sealed shielded storage canisters and immersed in the cooling pool temporarily until their decay heat decreases to safe levels. The canisters are then transferred to the storage vault, which features radiation shielding and forced air cooling.

6. Plant Safety and Operational Performances

JFHR leverages the inherent safety features of TRISO-coated particle fuel and molten salt coolant to attain robust plant safety and operational performance. JFHR eliminates the need for a large, high pressure containment structure and therefore avoids the risk of high pressure vessel rupture. The PRHR through natural circulation maintains core cooling at all conditions. Furthermore the use of LEU and a once-through fuel cycle enhances proliferation resistance. The high volumetric heat capacity of the graphite moderator and molten salt coolant leads to predictable thermal transients. This reduces the impact of operational changes on the reactor core and contributes to high reliability and a capacity factor greater than 90%.

7. Instrumentation and Control Systems

The I&C systems of JFHR share fundamental similarities with conventional nuclear power plants, monitoring temperature, pressure, flow rate, coolant level, and neutron flux. The non-safety-related Plant Control System (PCS) manages normal operation and Anticipated Operational Occurrences (AOOs). Under all DBCs and DECs, the safety-related Reactor Protection System (RPS) provides reliable safety functions by automatically or manually tripping the reactor and activating the PRHR system. All I&C sensors must tolerate elevated temperatures up to 700℃, intense gamma and neutron radiation, and salt corrosion, ensuring long-term stability and calibration integrity.

8. Plant Layout Arrangement

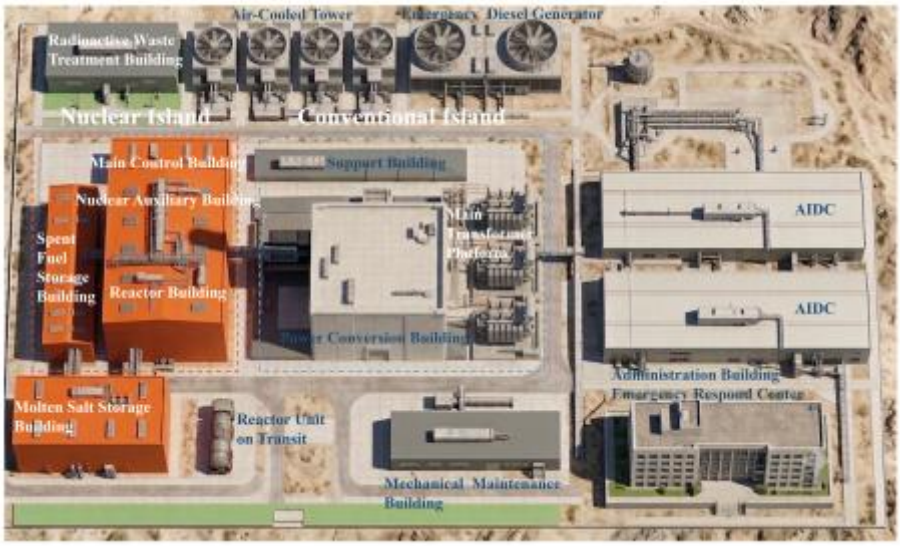


Figure 1. Plant Layout Rendering

The major facilities of the JFHR are Nuclear Island (NI), Conventional Island (CI) and Balance of Plant (BOP). The NI comprises the reactor building, spent fuel storage building, nuclear auxiliary building, main control building, and electrical building. The CI includes the power conversion building, main transformer platform and other supporting structures. The BOP consists of the radioactive waste treatment building, mechanical maintenance building, administration building, and emergency response center.

9. Testing Conducted for Design Verification and Validation

JFHR R&D plan follows a phased, iterative approach integrating separate effects tests (SETs) and integrated effects tests (IETs). SETs will focus on validating fuel-salt compatibility, conducting long-term corrosion tests, as well as undertaking mechanical performance tests. During SETs, a chemistry control and graphite impurities filter system for the Flibe in the primary loop will be established. For IETs, a non-nuclear engineering testing rig will be constructed to demonstrate integrated operations at high temperature.

10.Design and Licensing Status

JFHR design is developed in accordance with IAEA Safety Standards. The design addresses all safety classifications, including DBCs and DECs, with passive safety features such as gravity-driven shutdown and natural circulation decay heat removal. The licensing approach follows a graded, stepwise path consistent with international practice for advanced reactor deployment.

11.Fuel Cycle Approach

JFHR adopts a once-through fuel cycle. The fuel form and manufacturing process stay the same, enabling direct use of the same production line set up for JFHR. Spent fuel is stored on-site in compact storage canisters. The used fuel will finally be transported for disposal in a centralized repository. This approach simplifies fuel handling and reduces proliferation risk.

12.Waste Management and Disposal Plan

In JFHR, gaseous radioactive wastes are processed through high-efficiency filters before being discharged. Liquid radioactive wastes are collected in dedicated tanks, treated, cemented for solidification, and subsequently sealed for storage. Solid radioactive wastes are compressed and packed, loaded into solid waste collection containers after treatment. All radioactive wastes are classified according to their post-treatment radioactivity levels and disposed of in compliance with end-user country's policies.

13.Development Milestones

Year	Milestone	Status
2025	Technology concept	**Completed**
2025	Pre-conceptual design phase and technology validation	**Completed**
2026	Conceptual design	**Ongoing**
2028	Basic design	**Planned**
2029	Detailed design	**Planned**
Early 2030	First criticality	**Planned**

Kairos Power, United States of America
KP-FHR



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Kairos Power, LLC, United States of America
Reactor type	MSR
Coolant/moderator	Li2BeF4 (Flibe) / graphite
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	376 / 150 MWe (2 x 75 MWe)
NSSS operating pressure (primary/secondary), MPa	< 0.2
Core inlet/outlet coolant temperature (°C)	550 / 650
Primary circulation	Forced circulation
Fuel type/assembly array	TRISO particles in graphite pebble matrix / pebble bed
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	19.75
Core discharge burnup (GWd/ton)	N/A
Refuelling cycle (months)	Online refueling
Fuel cycle requirements/approach	Once-through Uranium
Reactivity control mechanism	Control elements, boron
Approach to safety systems	Passive
Seismic design (SSE)	Target: contiguous USA
Design life (years)	20 (vessel), 80 (plant)
Plant footprint (m^2)	N/A
RV height/diameter (m)	7.2 / 3.9
RV weight (metric ton)	N/A

1. Introduction

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

☒ Electricity
- ☐ Baseload
- ☐ Load-following

☐ District Heating / Steam

☐ Hydrogen
- ☐HTSE - electrolysis
- ☐PEM - polymer membrane

☐ Desalination

☐ Industrial process heat
- ☐ <200°C
- ☐ 200–550°C
- ☐ >550°C

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.

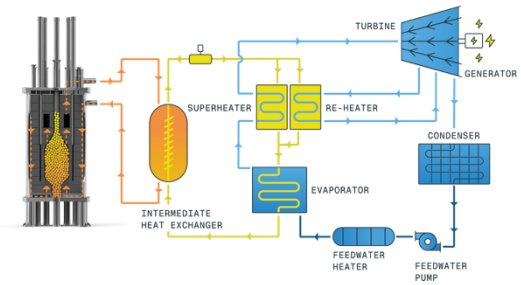


Figure 1. KP-FHR Power conversion system diagram

3. Design Philosophy

The fundamental design concept is the combination of Tri-structural Isotropic (TRISO) particle fuel coupled with molten fluoride salt coolant (2LiF:BeF2, Flibe). This combination results in a high-temperature, low-pressure reactor with multi-layered, 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) Nuclear Steam Supply System

Power Conversion

The power conversion system will leverage significant power generation industry experience with steam generators to transfer heat from the primary salt system to an intermediate heat transport fluid and then to steam. This steam will be used in a conventional steam Rankine power cycle to generate power in a commercially available steam turbine generator.

(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 B4C 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.

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.

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.

(d) Reactor Vessel (RV) 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) Reactor Coolant System and Steam Generator

Chemical Control System

A Chemistry Control System (CCS) monitors the salt coolant to detect impurities, ensuring it stays within an established operating window.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

Longer than 72-hour coping time for core cooling without AC or DC power, or operator action

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) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) 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 Systems

The Kairos Power Instrumentation and Control Systems take 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. Plant Layout Arrangement

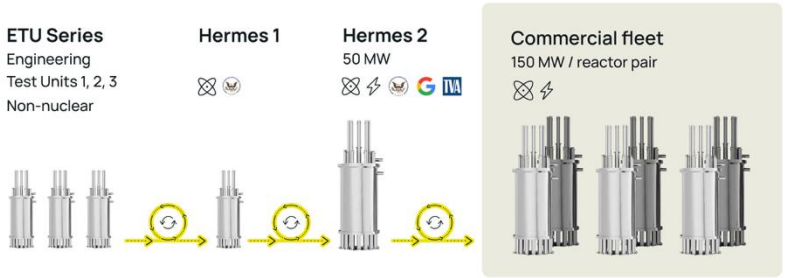


Figure 2. Kairos Power's Rapid Iterative Approach to Hardware Development

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 completed 2,400 hours of pumped salt operations in February 2024 and was decommissioned the same year. ETU 2 will begin operation in 2026.

Hermes 1 – a non-power reactor that will demonstrate Kairos Power's ability to produce affordable nuclear heat. Hermes is the first Gen IV advanced reactor with a U.S. Nuclear Regulatory Commission (NRC) construction permit. Nuclear safety-related construction began May 2025

Hermes 2 – a 50 MW commercial-scale reactor that will incorporate lessons learned from Hermes 1 to demonstrate power generation and supply clean electricity to the grid. Hermes 2 is the first electricity-producing advanced reactor with an NRC construction permit. Excavation and site work will begin in 2026.

Commercial Fleet – Kairos Power's modular commercial power plant is a scalable energy solution that can be configured for a range of needs and grow with demand.

- Minimum configuration (KP-150): A single power block with two reactors and a shared power-conversion system generates up to 150 MWe for data centers and industrial users
- Standard configuration (KP-450): Three power blocks (six reactors) generating a combined 450 MWe supports base load generation and large-scale infrastructure
- Larger configurations: Add more power blocks as energy needs evolve to achieve outputs of 600 MWe or more

9. Testing Conducted for Design Verification and Validation



Figure 3. Engineering Test Unit 1



Figure 3. Salt Lab located in Alameda, CA



Figure 5. RAPID Lab located in Alameda, CA



Figure 6. In-house Manufacturing Shop located in KP-Southwest





Figure 7. Engineering Test Unit 2 in Albuquerque, NMETU 2 Skid being lifted into place in Albuquerque, NM



Figure 8. Aerial view of Oak Ridge reactor demonstration campus from April 2026



Figure 9. Aerial view of Manufacturing Development Campus from March 2026

10. Design and Licensing Status

Kairos Power received its construction permit for the Hermes 1 low-power demonstration reactor from the NRC in December 2023. Hermes is a reduced-scale demonstration reactor that will prove Kairos Power's capability to deliver low-cost nuclear heat. In November 2024, the NRC issued construction permits to Kairos Power for the Hermes 2 demonstration plant, which builds on the lessons from Hermes 1.

11. 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.

12. Waste Management and Disposal Plan

The used TRISO pebble fuel and a small inventory of 'greater than Class C' waste are 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.

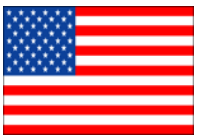
13. Development Milestones

Year	Milestone	Status
2018	Pre-Conceptual Design Completed Commissioned R-Lab Initiation of Pre-Application Review with NRC	**Completed**
2020	Acquisition of KP-Southwest Facility in Albuquerque, NM Commissioned S-Lab Formed strategic partnership with Materion Corporation to produce Flibe Kairos Power is selected for the U.S. DOE's Advanced Reactor Demonstration Program (ARDP) Risk Reduction Award	**Completed**

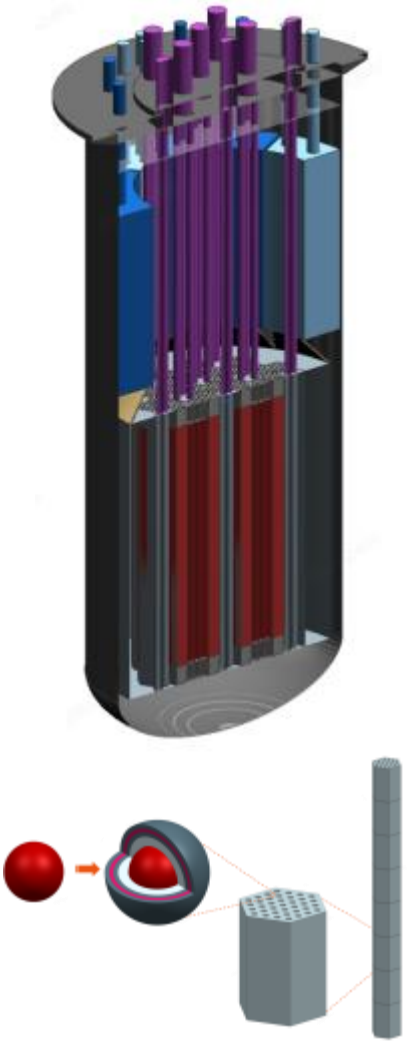
Year	Milestone	Status
2021	Cooperative Development Agreement with Tennessee Valley Authority to deploy Hermes low-power demonstration reactor Testing Facility construction completed in Albuquerque, NM Hermes Construction Permit Application accepted for review by U.S. Nuclear Regulatory Commission Construction begins on the first Engineering Test Unit (ETU 1)	**Completed**
2022	Kairos Power establishes the KP-OMADA Advanced Nuclear Alliance with leading North American utilities and generation companies Kairos Power commissions Molten Salt Purification Plant to produce Flibe coolant for the ETU series 30,000th pebble manufactured for ETU 1. Kairos Power loads 12 metric tons of Flibe into ETU 1 and commences molten salt operations	**Completed**
2023	NRC issues Construction Permit for Hermes low-power demonstration reactor	**Completed**
2024	ETU 1 completes 2,400 hours of pumped salt operations and enters decommissioning Kairos Power and U.S. Department of Energy (DOE) sign a milestone-based Technology Investment Agreement to implement up to $303 million in risk reduction funding under the Advanced Reactor Demonstration Program (ARDP) to support the design, construction, and commissioning of the Hermes low-power demonstration reactor Construction begins on the second Engineering Test Unit (ETU 2) Kairos Power dedicates Manufacturing Development Campus in Albuquerque, New Mexico, and breaks ground on Salt Production Facility to produce molten salt coolant for its advanced reactors Site excavation begins for the Hermes demonstration reactor in Oak Ridge, Tennessee Kairos Power and Google sign a Master Plant Development Agreement, creating a path to deploy a fleet of advanced nuclear power projects totalling 500 MW by 2035 NRC approves construction permits for Hermes 2 Demonstration Plant	**Completed**
2025	Kairos Power completes fabrication and installation of first internally produced reactor vessel for ETU 2 Kairos Power launches Operator Training Simulator at University of Tennessee, Knoxville Kairos Power begins nuclear safety-related construction of Hermes low-power demonstration reactor Kairos Power installs reactor vessel for third Engineering Test Unit (ETU 3) in Oak Ridge, Tennessee Kairos Power and the Tennessee Valley Authority sign the first binding power purchase agreement for a U.S. advanced reactor for Kairos Power's Hermes 2 Plant to deliver up to 50 MW of clean energy to the TVA grid that powers Google data centers in Tennessee and Alabama Kairos Power and BWXT announce collaboration on commercial TRISO fuel manufacturing Kairos Power completes installation of precast concrete shielding demonstration structure at Oak Ridge campus	**Completed**

Year	Milestone	Status
2026	U.S. Department of Energy and Kairos Power finalized contract to receive High-Assay Low-Enriched Uranium (HALEU) sourced from DOE material for the startup and operation of the Hermes low-power demonstration reactor Oak Ridge National Laboratory and Kairos Power enter into a $27 million strategic partnership to accelerate deployment of next generation of advanced nuclear reactors Kairos Power completes ETU 2 vessel assembly with successful top head closure Kairos Power breaks ground on Hermes 2 demonstration plant	**Ongoing**

Filbe Energy, U.A.E., United States of America Lithium Fluoride Thorium Reactor



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Flibe Energy, Inc. (FEI) United States of America
Reactor type	MSR
Coolant/moderator	LiF-BeF2-UF4 (fuel salt LiF-ThF4 (blanket salt)) LiF-BeF2 (coolant salt) Graphite moderator
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	60 / 25 (microreactor) 600 / 250 (SMR)
NSSS operating pressure (primary/secondary), MPa	Salts: <0.5 MPa (low) sCO2: ~20 MPa (moderate)
Core inlet/outlet coolant temperature (°C)	500 / 650
Primary circulation	Forced circulation
Fuel type/assembly array	Liquid core: Dissolved fuel salt/graphite channels
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	0% enrichment; uses thorium for refuelling after U-233 (or alternative) startup
Core discharge burnup (GWd/ton)	N/A; ~99% fuel utilization (continuous processing)
Refuelling cycle (months)	Continuous online refuelling
Fuel cycle requirements/approach	Closed thorium fuel cycle; fissile required only at startup; no actinides enter waste stream
Reactivity control mechanism	Negative temperature coefficient; passive and active control rods may be utilized
Approach to safety systems	Passive
Seismic design (SSE)	Site-dependent (TBD); minimal-harm scenarios expected
Design life (years)	20+ years microreactor Up to 80 years for SMR
Plant footprint (m^2)	~20e3 m2 microreactor ~40e3 m2 SMR
RV height/diameter (m)	~0.5m/0.4m microreactor ~5m/4m SMR
RV weight (metric ton)	~7 MT microreactor ~70 MT SMR

1. Introduction

The Lithium Fluoride Thorium Reactor (LFTR) design by Flibe Energy is a graphite-moderated, thermal-spectrum molten salt breeder reactor using liquid fluoride salts containing both fissile and fertile materials. Thermal power generated from nuclear fission drives electrical generation in a closed-cycle supercritical carbon dioxide (sCO_2) gas turbine power

conversion system. The primary objective is to produce electricity at low cost by efficiently consuming thorium, thereby avoiding known bottlenecks in enriched uranium supply chains (e.g., conversion, enrichment).

Fluoride salts, when heated to operating temperatures (500–700°C), form a stable ionic medium for nuclear fission. Their high volumetric heat capacity and low vapor pressure enable high-temperature operation at essentially ambient pressure, eliminating many concerns with high-pressure reactors.

The high outlet temperature makes the LFTR well-suited for coupling to an sCO_2 Brayton cycle, which can achieve thermal-to-electric efficiencies of ~45%, and a significant improvement/reduction in site footprint. The design features a two-region core (fuel and blanket, productive shielding) and a closed thorium fuel cycle. Thorium-232 in the blanket is converted to uranium-233 via neutron capture and subsequent beta decay. Chemical processing and offgas management allow for refuelling without shutdown and removal of neutron poisons (e.g., xenon). This approach enables near-complete utilization of thorium resources and very low fuel costs. To illustrate this, the table below compares the expected fuel cycle cumulative demand for 1 GWe of reactor(s):

Reactor	Cumulative Demand and Generation (MT)	Year 1 (incl. startup)	10 years	20 years	40 years	80 years
LFTR	U-233 (or equiv.)	1.0	1	1	1	1
	Thorium	217	226	236	256	296
	High-level waste (FPs)	1.0	10	20	40	80
LWR	U-235	1.5	15	30	60	120
	Natural Uranium	1,281	2,112	4,223	8,446	16,892
	Enriched Uranium	150	247	493	986	1,973
	High-level waste (SNF)	24.7	247	493	986	1,973

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☒HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☒ >550°C

The LFTR concept is flexible, maintainable, and will produce power at market-competitive prices. It is being developed first to provide inexpensive, safe, and reliable baseload power. The design can also complement applications for district heating and industrial process heat, among other applications to be pursued in later stages of commercialization.

3. Design Philosophy

The LFTR aims to produce electricity at the lowest cost possible by leveraging inherent safety to lower build costs and maximize fuel utilization and efficiency. Key guiding principles include: 1) inherent safety: no meltdown risk or plume risk, and near-atmospheric operating pressure; 2) simplicity: stable, self-regulating design with strong negative temperature feedback; 3) fuel efficiency: near-complete consumption of thorium in a thermal spectrum, which has the potential to generate practically unlimited electricity from existing thorium resources; 4) waste minimization: significantly reduced waste volume and radiotoxicity compared with other reactors, through careful separations and management is actinides, fission products, and other isotopes/materials. LFTR's small footprint and dry air cooling capability also enables flexible siting options.

4. Main Design Features

(a) Nuclear Steam Supply System

Power Conversion System

There is no steam generation in the reference design for LFTR. Rather, a coolant salt intermediate loop transfers energy from the fuel and blanket salts to the sCO2 working fluid of the power conversion system. This enables tritium management/control, a smaller footprint, higher efficiency, and lower costs. For the microreactor, this is demonstrated technology, and practical scalability is expected for a LFTR SMR coupling.

(b) Reactor Core

The reactor vessel contains two integrated plena: a central active core region with fuel salt flowing through graphite moderator channels, and an outer blanket region with thorium-bearing salt. The vessel also houses passive and active reactivity control systems. Replaceable graphite moderator structures are incorporated to manage neutron damage over the plant lifetime. Core salts drain into passively cooled drain tanks to establish a subcritical configuration for safe shutdown of any duration.

(c) Reactivity Control

Reactivity is managed primarily by the strong negative temperature coefficient. Passive control rods could be utilized, if desired. Conventional active control rods would provide finer operational control and shutdown capability. The design ensures the reactor cannot be started unless control rods are fully withdrawn. No additional reactivity is required to be managed, due to online refueling and removal of neutron poisons. Hard moderation by graphite eliminates risk of a reactivity insertion. For extended safe shutdown, the fuel salt can be drained from the core to an unmoderated and passively-cooled drain tank (subcritical configuration) until restart is desired and the fuel is pumped back to the graphite channels in the reactor vessel. The ability to easily separate the moderator from the fuel is an important feature in LFTRs.

(d) Reactor Vessel (RV) and Internals

The reactor vessel is constructed from a Hastelloy N alloy (or similar) optimized for high-temperature fluoride salt service. Internals include replaceable graphite moderator blocks that separate and direct the fuel and blanket salt flows. The salts are impervious to radiation damage accumulation due to natural recombination.

(e) Reactor Coolant System and Steam Generator

An intermediate coolant-salt loop provides thermal coupling and pressure isolation between the low-pressure primary system and the small-footprint sCO_2 power conversion system. No traditional steam generator is used.

(f) Pressuriser

Not applicable to this low-pressure system. In the sCO2 PCS, compressors will raise the pressure for operation between about 7.4-20 MPa.

(g) Primary pumps

Forced circulation is provided by pumps for the fuel salt, the blanket salt, and the coolant salt. In the sCO2 PCS, turbomachinery includes compressors and turbines. All are designed for the appropriate service lives.

(h) Distinguishing features

Online chemical processing for continuous fuel salt cleanup, fissile/fertile material management, safeguards tracking, potential for isotope harvesting, and watse management (only short-lived waste streams for near-surface storage/disposal solutions). Passive shutdown system through draining. High-efficiency and small footprint sCO2 PCS rather than a large turbine hall. Deployable in various applications. No long-lived waste streams accumulate, and no geologic storage of waste is expected to be required. Low-cost, dual vessle configaution enables near 100% capacity factor and risk mitigation for any opertaional issues on the working reactor side, by simply transfering the fuel to the standby unit.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Natural and passive safety are utilized in place of more costly engineered safety systems. Key safety functions rely on inherent physics and simple enginered features and design decisions

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

Design Basis Conditions (e.g., loss of flow or overheating) are mitigated by the strong negative temperature coefficient and automatic activation of fuel drain to a subcritical, passively cooled configuration.

(e) Safety Approach and Configuration to Manage DEC

Design Extension Conditions are addressed through the chemical stability and high boiling point of fluoride salts, combined with the passive drain system that prevents core damage and maintains subcriticality.

(f) Containment System

The below-grade reactor cavity/silo provides the primary containment boundary for all reactor systems.

(g) Spent Fuel Cooling Safety Approach / System

Fission products are continuously or periodically removed via chemical processing. Any stored hot salt is maintained in passively cooled drain tanks and cells/vaults. The short-lived high-level waste streams are expected to be stored in near-surface locations on-site or in appropriate repositories. No actinides or long-lived isotopes are expected in waste streams that would require geologic storage.

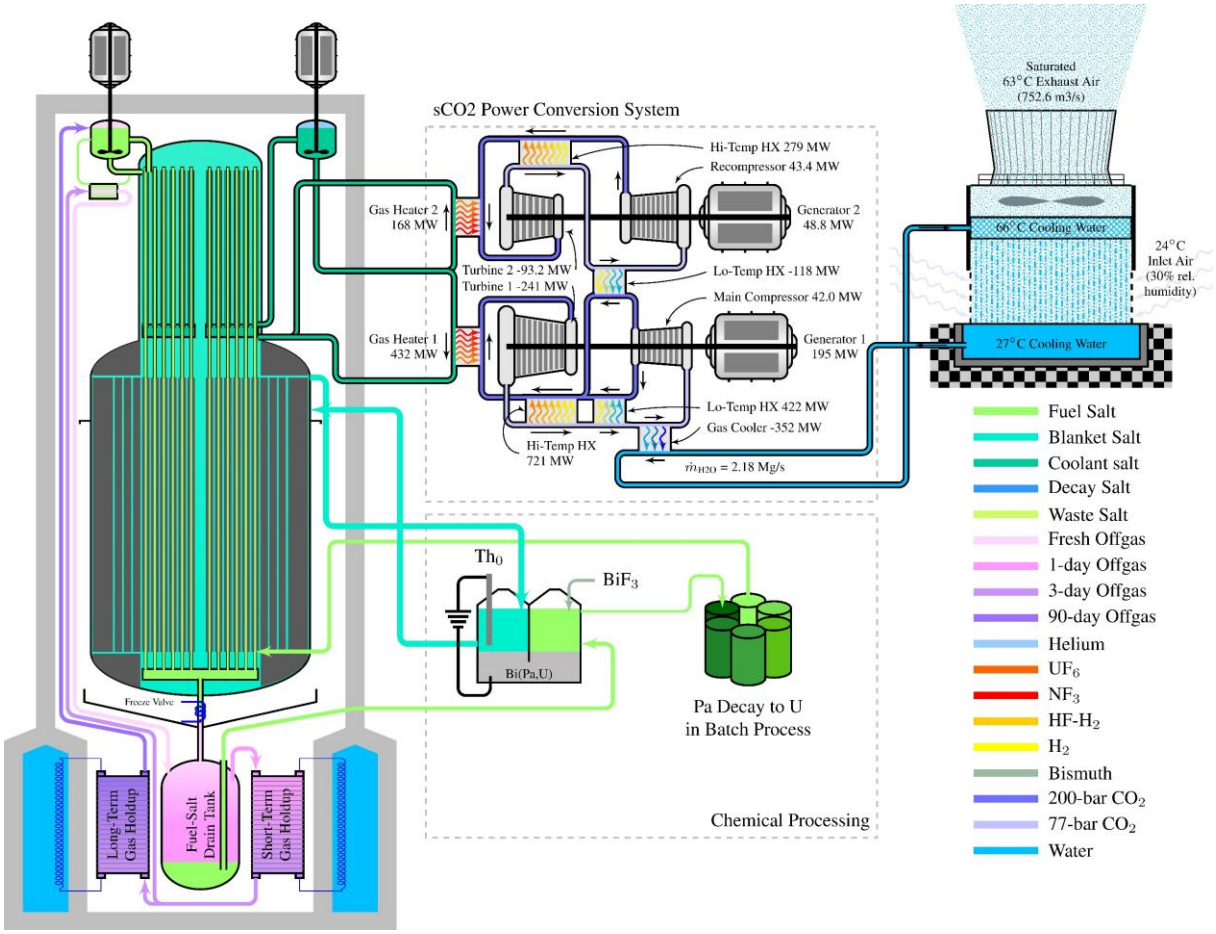
6. Plant Safety and Operational Performances

The LFTR delivers high operational reliability through passive safety systems, low-pressure operation, and continuous online processing. The design supports baseload operation with inherent load-following capability enabled by strong negative temperature feedback. Operational perfromance risks are mitigated with a dual vessel configaution, allowing inspection and repair while continuing power opertions.

7. Instrumentation and Control Systems

The I&C architecture leverages the inherent stability of the molten-salt design to simplify reactor protection and control systems, with dedicated monitoring for salt chemistry and fission gas management. Testing, inventory and fuel inspesction is easily accomplished through bulk sampling of the homogenized fluid.

8. Plant Layout Arrangement



The primary circuit is housed in a below-grade reactor cavity or silo. The modular approach supports factory fabrication of major components to minimize on-site construction time and cost. A simplified layout is below:

9. Testing Conducted for Design Verification and Validation

The design is primarily informed by historical data from the Molten Salt Reactor Experiment (MSRE) and associated testing at ORNL from the 1950-1970s. Modern non-nuclear testing of salt properties, components, and chemical processing systems is ongoing. No nuclear prototype testing has been performed to date, but deviation from historic demonstrations is minimal, and LFTR is ready for prototype demonstration.

10. Design and Licensing Status

Briefings, drop-in meetings, surveys/forms responses, and public presentations have been provided to NRC.

11. Fuel Cycle Approach

LFTR is a closed fuel cycle thermal-spectrum isobreeder (BR~1.0). It only needs a fissile material "charge" at startup – new fuel is then perpetually bred from a low-cost, unenriched thorium feed. Continuous refueling is expected without disrupting operations.

12. Waste Management and Disposal Plan

On-site chemical separations ensure long-lived isotopes (e.g., actinides) do not enter the waste streams, allowing for near-surface storage/disposal of low-value materials, rather than requiring geologic storage.

The fuel and blanket salts don't accumulate damage and with continuous removal of fission products and other materials, can be cycled perpetually through the core to achieve near-total burnup. The expected ~99% fuel (thorium feed) utilization is a 200x improvement over today's reactors that consume about ~0.5% of the natural uranium resource, and ~5% of the enriched uranium made into solid fuels and placed into burner (non-breeder) reactors. At the end-of-life of a LFTR site, the cleaned salts can be transferred and reused in subsequent reactor generations – a long-term asset, not a spent fuel liability.

For fleet-level comparisons, startup and 10 years of operation of 1 GWe of LFTR(s) would require about 226 MT of thorium (a 'waste' product in current mining activities) and 1 MT of U-233 (or equiv. fissile) at startup, and would produce about 10 MT of high-level fission product waste to be stored for 10-300 years. This compares to a traditional LWR that would require mining and milling hundreds of thousands of tons of rock to gather about 2,112 MT of natural uranium, which is then converted and enriched to produce about 247 MT of enriched uranium, which is then fabricated into solid fuel, loaded into the reactor, and converted into a long-lived spent nuclear fuel (~300,000 year liability).

LFTR's highly efficient waste management plan and fuel cycle approach are key distinguishing features for the technology.

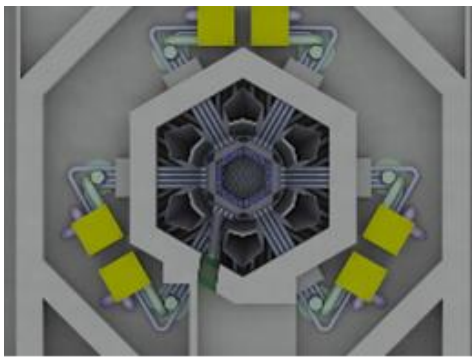
13. Development Milestones

Year	Milestone	Status
2011	Flibe Energy, Inc. (FEI) founded, the world's first private MSR company	**Completed**
2015	EPRI-funded LFTR technical assessment report published (collaboration with Vanderbilt and Southern Company)	**Completed**
2018	DOE award for advanced fluorination development work	**Completed**
2019	Gained control of a portion of the Yellow Creek Nuclear site	**Completed**
2019-2023	DOE awards on safeguards by design, offgas management, critical materials and supply chain assessment, and physical security by design.	**Completed**
2024	Nuclear utility contract to assess alternative fuel cycles, including startup of LFTRs through use of recycled spent nuclear fuel	**Completed**
2026	DOE selection for SNF recycling work, which could fuel startup of large fleets of LFTRs	**Completed**
Early 2030s	University test reactor, a prototype and demonstration project	**Planned**

Moltex Energy, UK and Canada, Stable Salt Reactor-Wasteburner



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MAJOR 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
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	750 / 300 continuous as baseload, 750 / 900 as 8-hour peaking plant
NSSS operating pressure (primary/secondary), MPa	Atmospheric
Core inlet/outlet coolant temperature (°C)	575 / 625
Primary circulation	Forced circulation
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)	N/A
Fuel cycle requirements/approach	N/A
Reactivity control mechanism	Negative temperature reactivity coef, Boron carbide shutdown assemblies
Approach to safety systems	Eliminate hazard, passive engineering
Seismic design (SSE)	Yes
Design life (years)	60
Plant footprint (m^2)	22500
RV height/diameter (m)	14 / 7
RV weight (metric ton)	50

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☒ >550°C

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) Nuclear Steam Supply System

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.
Reactor Vessel (RV) and Internals

The reactor vessel is constructed from a Hastelloy N alloy (or similar) optimized for high-temperature fluoride salt service. Internals include replaceable graphite moderator blocks that separate and direct the fuel and blanket salt flows. The salts are impervious to radiation damage accumulation due to natural recombination.

(d) Reactor Vessel (RV) 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.

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.

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.

(e) Reactor Coolant System and Steam Generator

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) Pressuriser

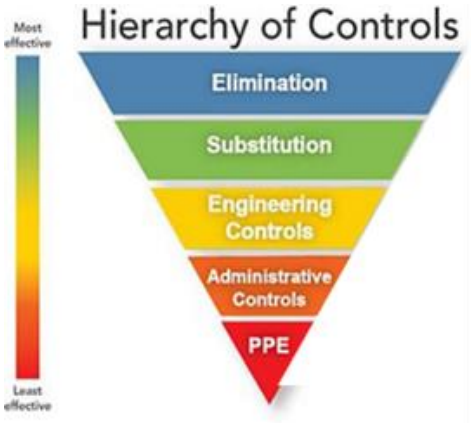
(g) Primary pumps

(h) Distinguishing features

Distinguishing features: Molten salt fuel in conventional fuel assemblies. Thermal energy storage. Very low cost conversion of spent LWR fuel.

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.

(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) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) Spent Fuel Cooling Safety Approach / System

Fission products are continuously or periodically removed via chemical processing. Any stored hot salt is maintained in passively cooled drain tanks and cells/vaults. The short-lived high-level waste streams are expected to be stored in near-surface locations on-site or in appropriate repositories. No actinides or long-lived isotopes are expected in waste streams that would require geologic storage.

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

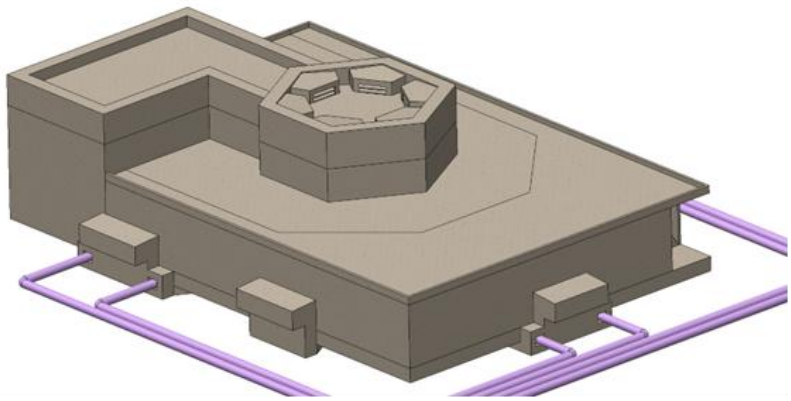


Figure 1. SSR-W Civil Structure

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.

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

Year	Milestone	Status
2014	UK patent granted for use of unpumped molten salt fuel in any reactor Independent capital cost estimate complete.	Completed
2015	Pre- conceptual design complete and key claims validated by the UK's National Nuclear Laboratories.	Completed
2017	Conceptual design completed and CNSC VDR commenced.	Completed
2018	Master patent on static fuelled molten salt reactors granted in major geographies. Several other patents progressing through PCT process. Moltex successful in UK Government Advanced Modular Reactor Competition. Moltex signs agreement with NB Power to advance the development of the first commercial demonstration SSR-W in Canada.	Completed
2020	Moltex signs ONWARDs agreement with Ontario Power Generation.	Completed
2021	Moltex receives funding from the Government of Canada's Strategic Innovation Fund. Moltex completes CNSC VDR Phase 1.	Completed
2022	Moltex announces strategic partnership with SNC-Lavalin Group	Completed
2023	North Shore Mi'kmaq tribal council and its seven First Nation member communities make financial investment in Moltex Successful completion of WATSS experiments using simulated fuel	Completed
2024	WATSS patents registered SSR-W integrated salt test loop built, and waste recycling demonstrated with advanced modelling techniques	Completed

Stellaria, France, Stellarium



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Stellaria, France
Reactor type	Molten Salt Fast Reactor
Coolant/moderator	Molten Salt/no moderator
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	540 / 250
NSSS operating pressure (primary/secondary), MPa	0.13 / 0.4 (hydrostatic) /0.4 (hydrostatic) / 22
Core inlet/outlet coolant temperature (^{o}C)	550 / 750
Primary circulation	Natural circulation
Fuel type/assembly array	Molten Uranium, Plutonium Chloride (optional minor actinides)
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	< 8% PuCl3 or 16.5% HALEU
Core discharge burnup (GWd/ton)	100
Refuelling cycle (months)	240
Fuel cycle requirements/approach	take-back
Reactivity control mechanism	Control Rods
Approach to safety systems	Hybrid
Seismic design (SSE)	0.45 g
Design life (years)	60
Plant footprint (m^2)	45 000
RV height/diameter (m)	7 / 2.3
RV weight (metric ton)	30

1. Introduction

The Stellarium is a 540 MWth /250 MWe molten chloride salt fast breeder reactor developed by Stellaria, a spin-off of the Commissariat à l'Energie Atomique et aux Energies Alternatives (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 achieve 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 continuous breeding and burning of these materials.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☒ >550°C

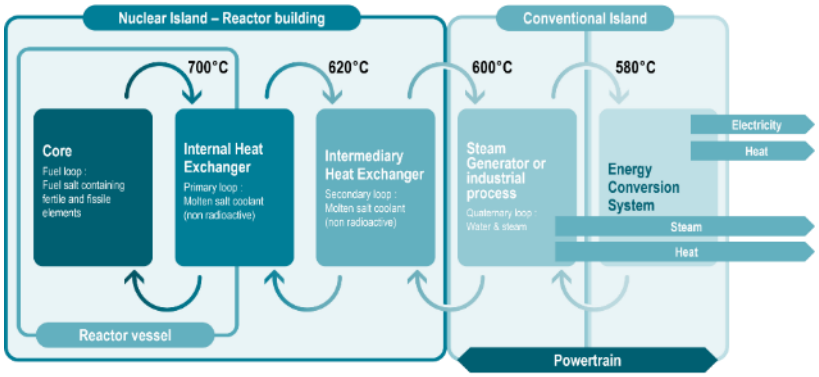
The Stellarium reactor targets energy-intensive industries and utilities aiming to utilize their plutonium inventories and reduce waste. Built in pairs for maximum economic efficiency, a Stellarium plant can generate 220 MW of electricity with supercritical steam turbines (~50% efficiency) or 500 MW 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 extracts the heat form the fuel salt by entering a 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 ~4.5 tons of plutonium and ~23 tons of depleted uranium. The breed and burn process allows for 20 years of autonomy, each fission producing energy and neutrons, the latter being used to fertilize the uranium.

(c) Reactivity Control

The reactivity control is ensured by control rods. The need for antireactivity being modest because the breeding allows for low excess reactivity.

(d) Reactor Vessel (RV) and Internals

The reactor vessel functions at atmospheric 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.

(h) Distinguishing features

Replacement of vessel every 10 years. In-core closed fuel cycle.

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 has 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) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

The Stellarium exhibits favourable 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.

(e) Safety Approach and Configuration to Manage DEC

In more severe transients (such as a main vessel leak followed by a guard vessel leak), most 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.

(f) 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 buried and only the hall is above ground.

(g) 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 designed to fit inside the existing French Autorité de Sûreté Nucléaire et de Radioprotection (ASNR, French nuclear regulator) framework. Studies are ongoing at the European level to harmonize parts of the different existing frameworks and Stellaria will include the conclusions to the design. The Stellarium is also designed to be built everywhere in Europe and in most of the countries. That means that stronger design earthquakes have been considered. 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 refuelled 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 developed 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 540 MWth reactors each producing 250 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 buried 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 dedicated lab. A critical mock-up named ALVIN (Advanced Liquid Validation for Innovative Neutronics) will be built by 2030, aiming for criticality by 2031. It will consist of an experimental reactor measuring 50cm by 50cm, with a thermal power of 200kWh. It will be built for safety demonstration purposes only, and will be succeeded by MEGALVIN, a prototype scheduled for 2032 with a reactor-scale output of 10 MWth, MEGALVIN will allow Stellaria to qualify the fuel, the materials and some components of the Stellarium.

10.Design and Licensing Status

The consortium made of Stellaria, Orano and the CEA has been awarded by the French government with a 14-million grant (France 2030 phase 1) and has begun discussion with French ASNR in the preliminary screening framework. Preliminary design phase is over as end of September 2024. Preliminary design for ALVIN and MEGALVIN is ongoing. In December 2025, Stellaria has applied for the authorization of ALVIN to the ASNR. The reactor will be built in the CEA center of Cadarache, France.

11.Fuel Cycle Approach

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 pluto nium 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 or plutonium.

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

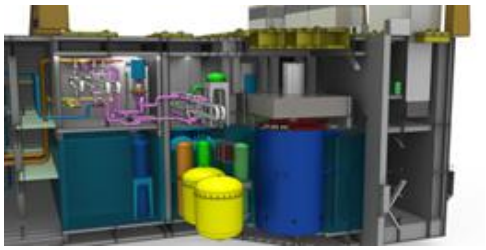
13.Development Milestones

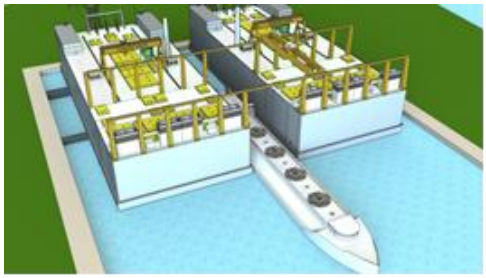
Year	Milestone	Status
2020-2023	Preliminary studies and technological innovation (using previously developed patents).	**Completed**
2023-2024	Pre-conceptual design phase and technology validation	**Completed**
2024-2026	Conceptual Design Phase (and preparation for pre-licensing)	**Completed**
2028-2030	Basic Design Phase	**Ongoing**
2030	First criticality of critical experiment, Alvin	**Planned**
2032	First criticality for the prototype, MegAlvin	**Planned**
2037	Target first criticality for the FOAK plant	**Planned**
2038	Target first criticality for the first NOAK plant	**Planned**

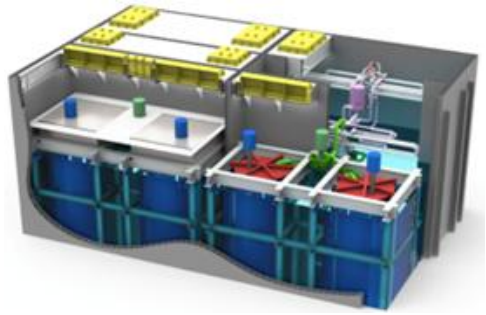
Thorcon International, USA and Indonesia, Thorcon



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Thorcon International, United States of America and Indonesia
Reactor type	MSR (thermal neutron spectrum)
Coolant/moderator	Molten salts / graphite
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	557 / 250 (per module)
NSSS operating pressure (primary/secondary), MPa	1.06 / 1.30 / 0.56 / 25.7
Core inlet/outlet coolant temperature (^{o}C)	560 / 704
Primary circulation	Forced (4 pumps), 1 per loop
Fuel type/assembly array	UF4 / molten salt
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	Startup 2.3 / makeup 4.95
Core discharge burnup (GWd/ton)	145.8
Refuelling cycle (months)	12 months to adding fuel 48 months to Can changeover
Fuel cycle requirements/approach	Fissile LEU05, or LEU19 with thorium conversion
Reactivity control mechanism	Negative temperature coef.
Approach to safety systems	Intrinsic, passive
Seismic design (SSE)	1.0
Design life (years)	80
Plant footprint (m^2)	67x162
RV height/diameter (m)	10.3 / 7.8
RV weight (metric ton)	343

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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.

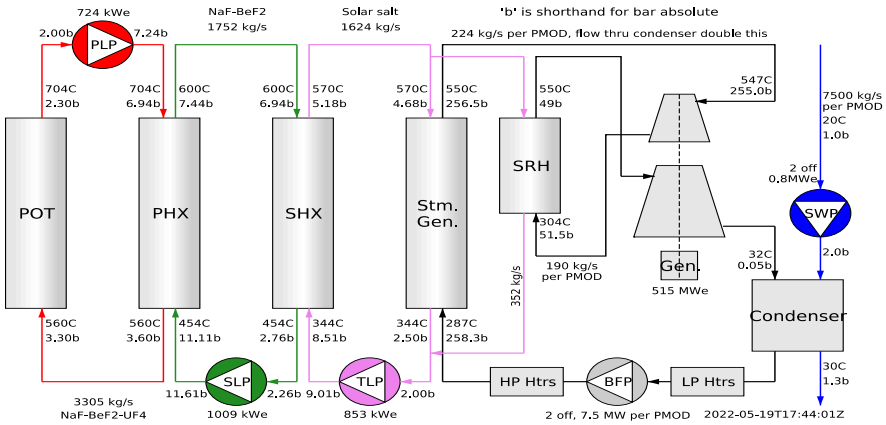
Ready to Go – The Thorcon design needs no new technology. Thorcon is a scale-up of the successful Molten Salt Reactor Experiment (MSRE). A full-scale dual 250 MWe Thorcon prototype can be operating under test in 2026 and then subjected to failure testing before commercial production can begin.

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) Nuclear Steam Supply System

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 30oC 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 704oC. 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).

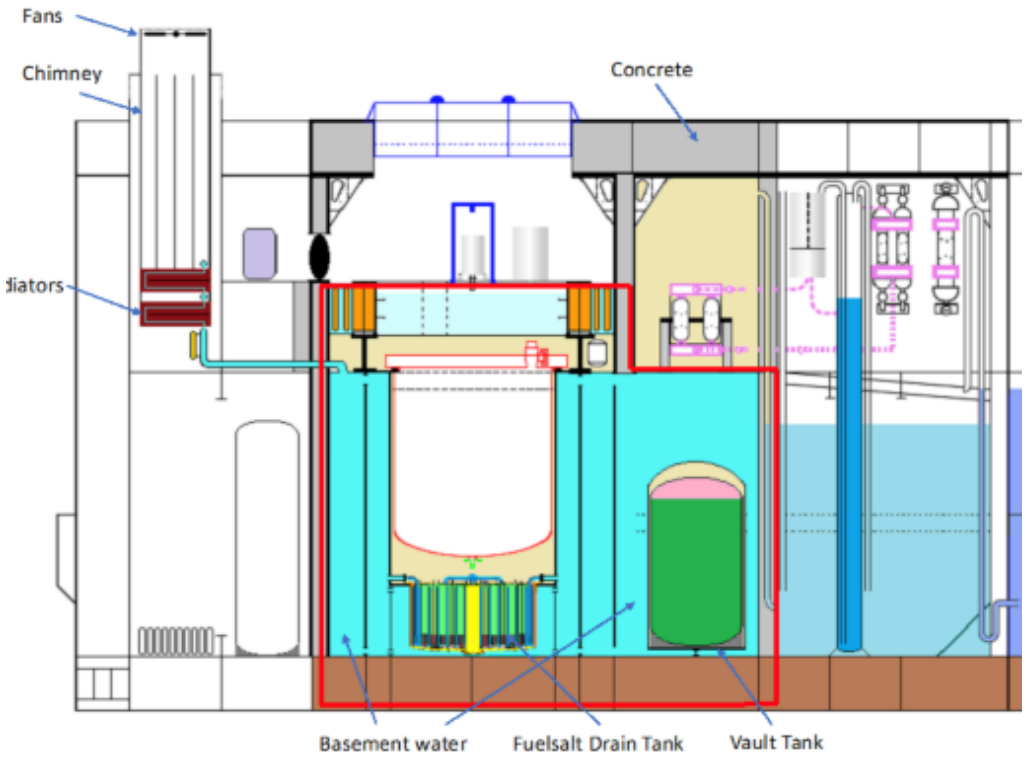
The PHX transfers heat to secondary salt in (green) piping above the Can. The 560oC fuel salt is piped back to the bottom of the Pot, where the graphite moderator slows neutrons, which fission uranium in the fuel salt as it rises through the Pot, heating the salt. Neutrons also convert some fertile 238U 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 BeF_2 is pumped through the PHX in the Can to a Secondary Heat Exchanger (SHX). It transfers heat to a third loop of $NaNO_3/KNO_3$, 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.

Fuel Characteristics

The fuel salt is NaF-BeF2-UF4 72/16/12. As fissile U is consumed fissile Pu239 is generated, but not enough to replace the fuel burned. Makeup fuel must be added continuously.

(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 Vessel (RV) 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) Reactor Coolant System and Steam Generator

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

Full passive safety, short construction time, low cost

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 704oC and the fuel salt boiling temperature of 1430oC 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) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) Spent Fuel Cooling Safety Approach / System

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 Validation

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

Year	Milestone	Status
2022	Initial Conceptual Design	**Completed**
2023-2024	Pre-Application review of Licensing Topical Reports (U.S.A.)	**Completed**
2024-2026	Conceptual Design Phase	**Ongoing**
2027	Preliminary Design Phase	**Planned**
2028	License to Construct application submitted (Canada)	**Planned**

THORIZON B.V., Netherlands, THORIZON



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	THORIZON B.V., Netherlands
Reactor type	MSR
Coolant/moderator	Chloride salt (no moderator)
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	250 / 100
NSSS operating pressure (primary/secondary), MPa	low pressure under all circumstances
Core inlet/outlet coolant temperature (°C)	500 - 800
Primary circulation	Forced circulation
Fuel type/assembly array	Fissile-fertile material bearing molten salt in closed containment modules
Number of fuel assemblies in the core	3-7-19 (current designs, can change)
Fuel enrichment (%)	N/D
Core discharge burnup (GWd/ton)	N/D
Refuelling cycle (months)	5 – 10 years target
Fuel cycle requirements/approach	Take-back
Reactivity control mechanism	Control rods, burnable poison, shutdown rods, and strong negative temperature feedback (pump control)
Approach to safety systems	Passive as much as possible
Seismic design (SSE)	N/S
Design life (years)	Full core replacement strategy allows life extension beyond 60 years
Plant footprint (m^2)	N/A
RV height/diameter (m)	N/A
RV weight (metric ton)	N/A

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 and use LWR-plutonium and commercialize. 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

☒ Electricity	☐ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

THORIZON offers a technology base for a multitude of different molten salt reactor designs. First goal is to maximize efficient LWR-Plutonium use within the shortest route to deployment, to reduce the long-lived nuclear waste burden, and turn waste cost into energy generation income, including depleted Uranium resources. The technology basis combined with the demonstration and commercialisation of first plants, could allow and support further expansion towards high breeding rations for advanced depleted or Thorium use.

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, handling and transport of spent fuel and degraded core materials. Another key aspect of the design approach is to avoid complexity at the plant, eliminate proliferation risk concerns through reprocessing and salt conditioning operations conducted off-site from the reactor, to enhance operability of the plant. Another aspect is to make sure that the design safety and performance can be assessed by basic computational tools, facilitating validation and licensing, as the cartridge introduces forced flow in a simplified geometry, reducing the complexity of the coupled phenomena that have to be represented in the modelling used for the design and for the safety analysis.

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 system, 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 central 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. Control rod and shutdown insertion also renders the reactor subcritical.

(d) Reactor Vessel (RV) and Internals

Thorizon has no reactor pressure vessel, but a radiation shield which also serves as external impact barrier, in which modules are placed. Each module contains its own pump and heat exchanger, and is closed, but connected to a salt venting system, to maintain a low cartridge pressure at all times.

(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. For first systems the cartridge length is limited to lengths that can be transported in currently adopted transport cask dimensions. 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, stability and control of the reactor's primary coolant system.

(h) Distinguishing features

Nuclear safe core material replacement strategy, with external core module series production, continuous improvement by module updates, and fuel cycle flexibility

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 barriers.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) 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, containment for each core module, secondary coolant circuit, the core cavity and shielding structure work together to manage and contain any incidents effectively.

(e) 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, and natural circulation capability of the secondary salt circuit, 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.

(f) 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

(g) 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 second barrier in place 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. Primary instrumentation relates to temperatures. Activity monitoring of secondary and tertiary system, and the core cavity cooling system. Flux measurement in between and outside cartridges.

8. Plant Layout Arrangement

Without cartridges present, the plant has no primary circuit or primary circuit components, and can therefore be considered largely conventional. Primary system components are manufactured in series and fueled 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 fuel salt has been initiated. This program will provide confirmation of speciation assumptions, including volatile and solid element or compound formation. In parallel, salt-material interaction is quantified for all foreseeable salt compositions, both static in laboratory, and by dynamic corrosion testing with a set of Thorizon designed and built corrosion loops. Thorizon has executed an irradiation program at the DICE facility of DIFFER, to determine the impact of irradiation phenomena on chemical interaction at the salt-material interface to support validity of salt-material interaction and corrosion tests without radiation. A gamma irradiation of cold salt is underway, to determine the potential radiolysis and gas release of frozen salt during shutdown or storage scenarios.

Primary system thermal-hydraulic calculations are to be validated with scaled mock-up and test loops, that are also used for component qualification.

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. This program is named Thorizon Pilot, with full sized components ready in 2027.

Thorizon has designed a critical demonstration facility, to confirm neutronic and thermal-hydraulic-neutronic coupled phenomena assumptions, as basis for validation. This demonstrator is named the Thorizon Pioneer.

10. Design and Licensing Status

Interaction with Nuclear Safety Authorities ASNR in France and ANVS in Netherlands has been ongoing since 2024. A location for the Thorizon Pioneer critical demonstrator has been selected at the NRG-Pallas site in Petten, which means that the emphasis of licensing has moved to the ANVS, with ASNR and Belgian FANC involvement. The demonstrator will be operated by NRG-Pallas under the current NRG-Pallas site license. A license to build is targeted for 2028.

A location for the Thorizon One first of a kind powerplant has been found in Zeeland, with EPZ, the current operator of the Borssele nuclear power plant as foreseen operator. License to build is targeted for 2032. 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 the material and salt characterization program, the salt-material interaction program, nuclear tests, the single cartridge non-nuclear demonstration, and the Pioneer critical demonstrator, while the plant build on site is simplified, as the primary circuit is constructed off site, in series, and can commence in early stage, prior to plant license to build.

The current achievements have been executed by a dedicated in-house team that has grown to around 50 people, located in Amsterdam offices and lab space and in Lyon offices, supported by external consultants and partners for specific subjects. Thorizon is closing a new financing round shortly, to support the current schedule to Pilot, Pioneer and First of a

kind realisation. 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 years 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 the material and salt characterisation, material-salt interaction and Pilot programs, and the nuclear test projects and Thorizon Pioneer critical demonstrator.

12.Waste Management and Disposal Plan

By design, Thorizon systems do not enable diversion of fuel salt from the reactor core. The cartridges can be transported in shielded casks of manageable dimensions. The salt is extracted for separation and purification and recycling at an external location, where the cartridge solid materials are cleaned and prepared for final storage.

13.Development Milestones

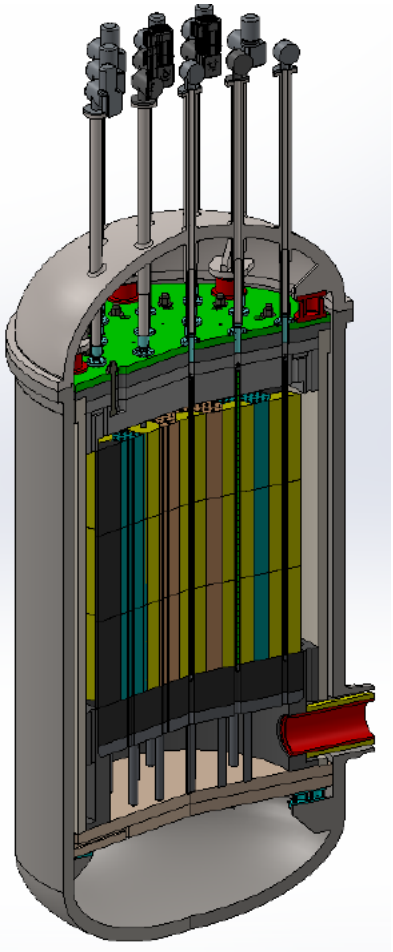
Year	Milestone	Status
2020	Pre-conceptual design phase, technical due diligence successful, patent positively evaluated.	**Completed**
2022	Investment secured, transition from conceptual design to basic design	**Completed**
2024	Licensing pre-screening successful, preliminary licensing evaluation initiated, salt irradiation project initiated, salt experimental program initiated, validation and demonstration program started	**Completed**
2026	Site selected and basic design finalized for the Pioneer demonstrator, pre-licensing for the Pioneer demonstrator initiated, transition to final design	**Ongoing**
2028	License to build Pioneer demonstrator granted	**Planned**
2028-2030	Non-nuclear cartridge component prototype operational (Pilot program), validation program ongoing, irradiation project finalized, material and component qualification finalized, cartridge off-site series production preparation and execution	**Planned**
2030-2031	Pioneer construction complete, commissioning of Pioneer and operating phase	**Planned**
2031	Thorizon One first of a kind detailed design finalized, pre-licensing for the Thorizon One first of a kind completed	**Planned**
2032	License to build Thorizon One granted	**Planned**
2034	Plant construction complete, commissioning of Thorizon One first of a kind, cartridge introduction, license to operate	**Planned**

MICROREACTORS

STL Nuclear (Pty) Ltd, South Africa, Advanced Micro Reactor – AMR



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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
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	10 / 3 for single module plant
NSSS operating pressure (primary/secondary), MPa	4 / 1
Core inlet/outlet coolant temperature (°C)	450 / 750
Primary circulation	Forced circulation
Fuel type/assembly array	TRISO particles/ LBE eutectic/ SiC tubes
Number of fuel assemblies in the core	420
Fuel enrichment (%)	10 to 20
Core discharge burnup (GWd/ton)	80-90
Refuelling cycle (months)	96
Fuel cycle requirements/approach	LEU UO 2
Reactivity control mechanism	Control and shutdown rods in core, and the reflector
Approach to safety systems	Passive / inherent decay heat removal
Seismic design (SSE)	0.3g (generic site) 0.5g under consideration
Design life (years)	40
Plant footprint (m^2)	2 500
RV height/diameter (m)	5.96 / 2.78
RV weight (metric ton)	115

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 (UO2) 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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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) Nuclear Steam Supply System

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

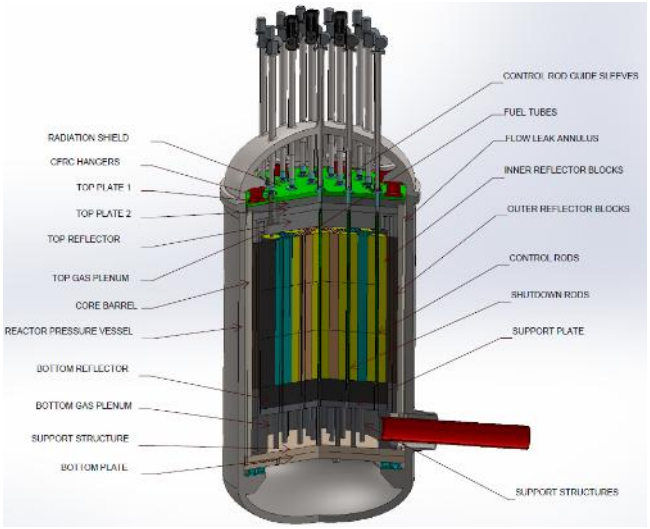


Figure 1. Reactor Core

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 m3 which results in a core power density of ~2 MW/m3. 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.

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.

(c) 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 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. 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".

(d) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

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.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

Additional SiC barrier; Lead Bismuth Eutectic (LBE) heat transfer medium

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.

Nuclear stability - 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.

Thermal stability - 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.

Mechanical stability - 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

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) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) 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.

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 Systems

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 UO2 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)O2 using both LEU and HEU, to a UC2 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

Year	Milestone	Status
2020	Project Started	**Completed**
2021	Pre-conceptual design	**Completed**
2022	To begin with conceptual design	**Ongoing**

Calogena, France, CAL-30



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Calogena S.A, France
Reactor type	PWR (Pool-type)
Coolant/moderator	Demineralized Water
Design status	Basic Design
Thermal/electrical capacity, MW(t)/MW(e)	30-35 MW(t)/0 MW(e)
NSSS operating pressure (primary/secondary), MPa	0.65 MPa/0.65 MPa
Core inlet/outlet coolant temperature (°C)	~110°C
Primary circulation	Natural
Fuel type/assembly array	UO2 17×17
Number of fuel assemblies in the core	32
Fuel enrichment (%)	< 5%
Core discharge burnup (GWd/ton)	40 GWd/ton
Refuelling cycle (months)	24 months (1/8 core)
Fuel cycle requirements/approach	Reprocessing or open fuel cycle
Reactivity control mechanism	Control Rods
Approach to safety systems	Full passive
Seismic design (SSE)	0.2g
Design life (years)	>60 years
Plant footprint (m^2)	3000
RV height/diameter (m)	15m/4m
RV weight (metric ton)	N/A

1. Introduction

The company CALOGENA, a subsidiary of the French industrial group Gorgé, is developing a small modular water reactor for heat-only applications (district heating), with a simple and robust design, operating at low pressure and with high inertia. Calogena has been a recipient of the France 2030 label in 2023 (5.2 M euros) and in 2026 (48 M euros). The company has also completed a fundraising round in 2026, with new investors including SNEF (French industrial group), Bpifrance (institutional) and Banque des Territoires - CDC (institutional).

This project is to be rolled out in series with the first construction launched before the end of the 2020s, unit currently in preparation at the CEA research centre in Cadarache in the south of France.

It aims to provide a sustainable, safe, competitive, and rapidly deployable response to the challenge of decarbonisation of low-temperature heat (<110°C), which represents one third of final energy demand in Europe. It can replace installations that use fossil fuels, and support the massive development of district heating networks, which are essential to achieving the "net zero" goal by 2050.

The project benefits from the extensive feedback from light-water power reactors and pool-type research reactors, and offers a simple, safe, sovereign, carbon-free, and local SMR solution, with a capacity for rapid industrial deployment.

The engineering and industrial development has been thought out to secure the robustness of the product from the start. To quickly meet decarbonisation needs, Calogena uses only proven technological bricks

2. Target Application

☐ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

Calogena is a 30-35 MWth nuclear water heater, with a simple and robust design, operating at low pressure and with high inertia, which aims to respond as quickly as possible to the challenge of decarbonizing urban heat and of extension of urban heat networks, existing or new ones. It provides heat at the boiler outlet adaptable to the needs of the district heating network, between 70 and 110°C.

3. Design Philosophy

Calogena is designed to be installed in a near-urban environment. To ensure the simplicity and robustness of the nuclear heater, Calogena has been developed with proven technologies based on extensive experience feedback from light water power reactors and in-pool research reactors.

The use of natural convection, the absence of electric pumps within the primary circuit, the large volume of water available in the pool and specific technical provisions make it possible to practically exclude accidental situations leading to core meltdown. The resulting intrinsic safety , together with an Emergency Planning Zone possibly limited to the periphery of the installation, and the very small footprint (1000m2 for the nuclear building) allow Calogena to be installed at a close distance from urban networks.

The Calogena module CAL 30 has a power of 30 to 35 MWth.It is installed in a nuclear building comprising a large volume water-filled pool (1000m3). The reactor vessel is immersed in the pool.

An intermediate circuit transfers the heat produced by the nuclear heater to the heating network.

The Calogena module is developed according to a series logic (a series of modules of identical design and manufacture). The modules are transported to their installation sites by road, rail, or sea.

4. Main Design Features

(a) Nuclear Steam Supply System

The Calogena operation is based on single-phase circulation of demineralized water at a temperature of 110°C operating at a pressure of 0.5 MPa (vs. a reference value of 0.65MPa), without steam production.

(b) Reactor Core

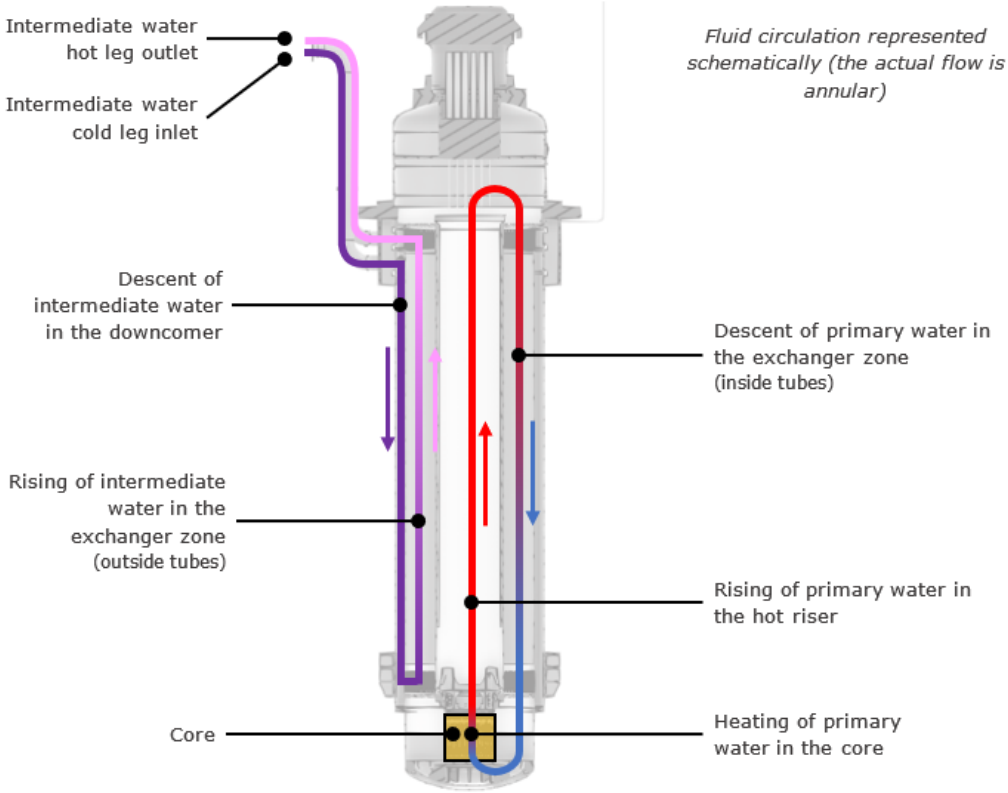
The reactor core consists of 32 fuel assemblies. Each fuel assembly contains 264 fuel rods consisting of UO2 pellets and zirconium alloy cladding, organised in 17x17 assemblies, standard PWR pitch. The technology and fabrication process are similar to those of PWR assemblies operated in France, down to the length of the fuel rods (fissile column ~1 m) and assemblies. The enrichment in uranium-235 of new fuel rods is less than 5%.

(c) Reactivity Control

The core operates in clear water (demineralized water), without the use of a soluble poison (same water conditions as in the pool).

The fuel assemblies accommodate a cluster of control rods. Apart from the height, the clusters are of a design similar to those for PWRs and use a conventional neutron absorber.

(d) Reactor Vessel (RV) and Internals



The reactor vessel, fully submerged in the pool shaft integrates:

- the reactor core at the bottom;
- the riser that carries hot water produced by the core in an upward direction; it is one of the driving elements of the thermosyphon, which allows primary coolant circulation;
- the downcomer integrating a heat exchanger allows the heated water to be cooled while flowing downwards through the heat exchanger tubes back to the inlet of the core. This is the second driving element of the thermosyphon;
- the lower primary plenum at the core inlet, and an upper primary plenum at the top of the hot riser;
- the pressuriser with control rod drive mechanisms, installed from the top;
- the downcomer of intermediate circuit water supplying the intermediate water to the shell side of the heat exchanger through which it flows upwards in a counter-current flow arrangement, ;
- connections with the primary safety valves and with the reactor coolant circuit water purification circuit (outside vessel);
- links with the legs of the intermediate circuit (heat transfer circuit).

(e) Reactor Coolant System and Steam Generator

The cooling of the core and the control rod clusters is provided by the thermosyphon of the nuclear heater by natural convection of the water. The main heat exchanger (single phase, no steam) is integrated in the reactor vessel.

(f) Pressuriser

The primary circuit is pressurised by a self-pressuriser. Thanks to the natural convection operation, no electric pump is necessary to ensure the circulation of water in the primary circuit.

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

The Calogena safety features are described, analysed and assessed in the Safety Options File submitted to the French regulator (ASNR) in October 2024 as well as the pre-assessment by the Finnish Regulator (STUK) completed in 2025.

(a) Engineered Safety System Approach and Configuration

The safety demonstration of the Calogena module is based on the study of a set of envelope design conditions of differing levels of severity that it may be confronted with.

The associated studies must make it possible to verify compliance with the General Safety Objectives defined for the installation. The seven General Safety Objectives have been established for the Calogena module from those identified by WENRA in the 2021 report relating to SMRs.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

A first set of situations studied is composed of situations resulting from the failure of equipment, or, if applicable, from human error, having an impact on a fundamental safety function. The failure is of the "single initiating event" (EIU) type. The accident or incident scenarios resulting from the selected EIUs are called DBC, Design Basis Condition. A classification of the operating conditions according to their estimated frequency makes it possible to frame and simplify the analysis.

The associated studies are conducted with conservative assumptions at the installation level in order to reinforce the determinism of the approach. The study rules include the application of a single complementary failure, independent of the initiator. This is generally of the active type, but it can be extended to a passive failure by considering, for example, a 24-hour leak.

(e) Safety Approach and Configuration to Manage DEC

For defence-in-depth, the study of the reference operating conditions is complemented by the study of a second set of situations corresponding to more complex sequences, which make it possible, in particular, to cover cumulative failures (multiple failures).

The extended analysis is intended to verify the robustness of the safety demonstration, and as such:

- The prevention of degradation of the core in the event of more complex situations than those considered in the design basis conditions;
- Similarly, the consistency between the estimated frequencies of accidental situations, including the impact of accumulations or common failure modes, and their potential consequences.

The study may lead, if necessary, to the identification of additional design provisions aimed at preventing transients or mitigating consequences.

Each accident scenario selected under the extended analysis is called DEC (Design Extension Condition). The associated studies are conducted by considering realistic assumptions and study rules.

(f) Containment System

The containment system is based on the design of the reactor and the implementation of a classified nuclear ventilation. The containment of radionuclides is ensured by an APC shell constituting the nuclear building, by the water of the pool capable of retaining radionuclides from accidental deterioration of the nuclear materials contained in the core and by the ventilation in operation in the reactor hall.

The active extraction network of the reactor hall (normal line change) is connected to Last Filtration Stage station and to the stack. An emergency ventilation with HEPA filters supplements if the situation requires it.

(g) Spent Fuel Cooling Safety Approach / System

The large volume water-filled pool includes an underwater rack that can accommodate a set of irradiated fuel assemblies in decay cooling. After sufficient decay, the spent fuel assemblies are introduced under water into a licensed nuclear transport container and removed from the facility by lorry for further treatment or storage.

6. Plant Safety and Operational Performances

The Calogena site has a local supervision station that allows personnel present on site or on duty to monitor the automated operation of the reactor as well as to check the physical security provisions of the site. The installation is also connected to a centralized supervision station set up for continuous 24-hour operation. The centralized station monitors several Calogena modules in operation on several separate sites.

In terms of performance, Calogena ensures load following and rapid adaptation to the demand of the district heating network in a range from 10 to 100% of the nominal power. The cost of carbon-free heat provided by Calogena is competitive with currently used production sources.

7. Instrumentation and Control Systems

The instrumentation that will be installed on the facility is that usually used in large nuclear facilities in accordance with national and international regulations on nuclear safety and security (instrumentation for monitoring the reactor operation, accelerometers for earthquake detection, site surveillance cameras, etc.).

As previously mentioned, the heating plant is supervised from a local supervision station, and remotely from a centralized supervision station operating 24/7.

8. Plant Layout Arrangement

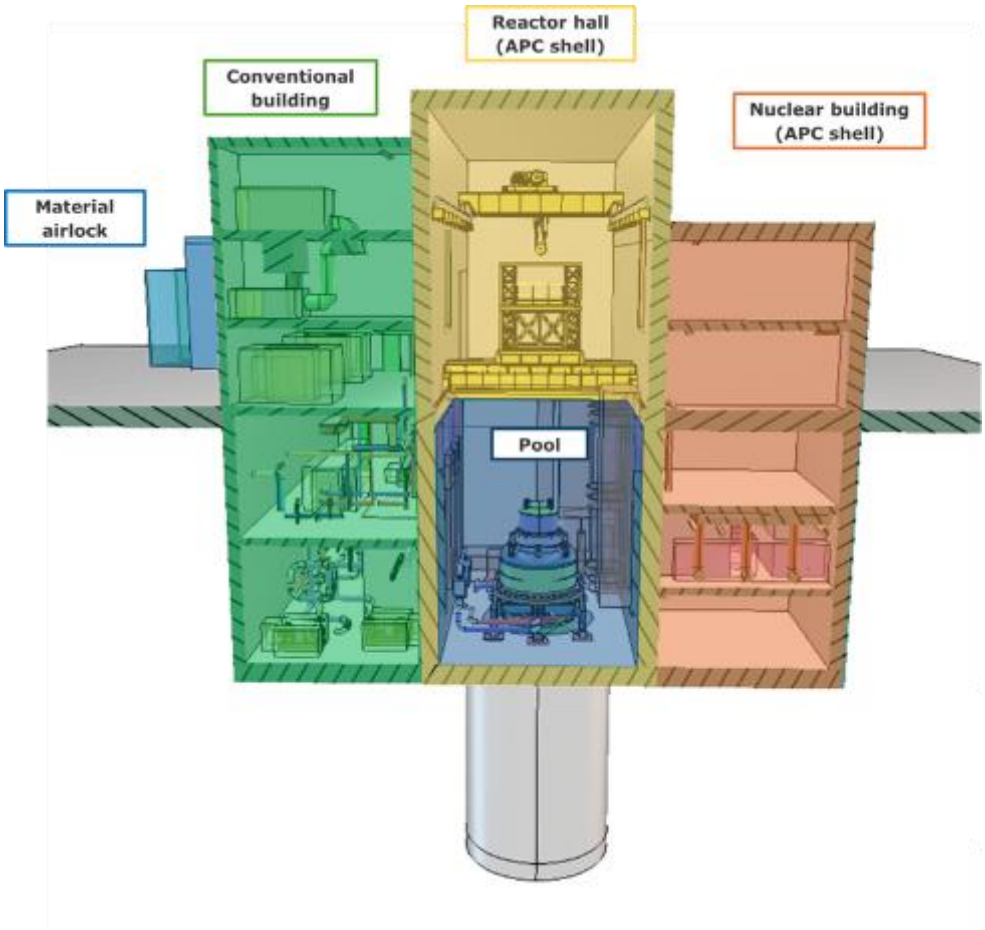


Figure 1. Cross-section view of the CAL-30 layout

The nuclear building is semi-buried for protection against external hazards and malicious acts.

9. Testing Conducted for Design Verification and Validation

The design is largely based on mature and proven technologies.

In this regard, no large-scale technological tests or experimental facilities are required to validate the design. Certain calculations have been performed for verification. Critical equipment will undergo selected qualification checks.

10. Design and Licensing Status

In October 2024, Calogena formally began the pre-instruction phase of pre-licensing with France's Nuclear Safety Authority (ASNR), submitting the initial materials for its Safety Option File (DOS). Furthermore, in 2025, Finland's Radiation and Nuclear Safety Authority (STUK) conducted a thorough pre-assessment of the CAL-30 reactor concept. This assessment was successfully completed in December 2025, with STUK providing official feedback intended to support the ongoing basic design phase.

11. Fuel Cycle Approach

The fuel cycle followed by Calogena is in all respects identical to the industrial PWR fuel cycle currently in place, both fo r fresh fuel and spent fuel. No specific facilities are needed.

The reference plant refuelling cycle of CAL30 is by 4 fuel assemblies every 2 years.

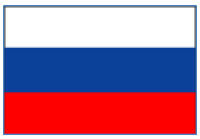
12. Waste Management and Disposal Plan

Each country where a Calogena SMR is installed will make its own choice between reprocessing spent fuel and geological disposal of highly radioactive ultimate waste as practiced in France or interim storage first and later direct storage of spent fuel assemblies.

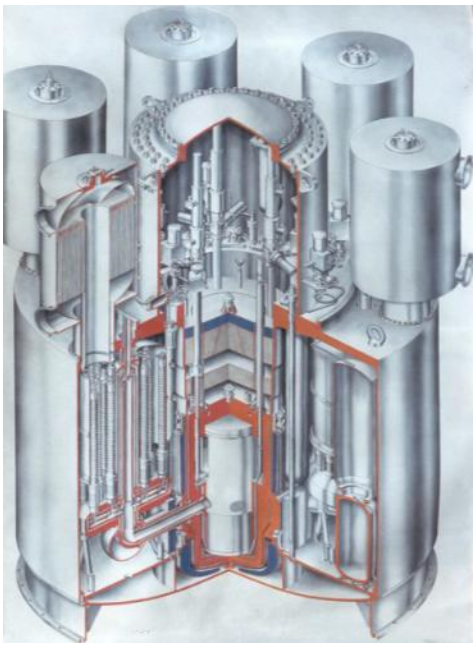
13. Development Milestones

Year	Milestone	Status
2023-2025	Conceptual Design Phase. Early Discussion with Safety authority. Submission of Safety and Security Option Files.	**Completed**
2025-2028	Basic Design Phase. Preparation of Preliminary Safety Analysis Report. Request for Authorization to Build	**Ongoing**
2027-2029	Detailed Design Phase, Licensing, Authorization to Build	**Planned**
2029-2032	1st Plant Construction – Commissioning	**Planned**

NRC “Kurchatov Institute”, Russian Federation, ELENA



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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
Design status	Conceptual design
Thermal/electrical capacity, MW(t)/MW(e)	3.3 / 0.068
NSSS operating pressure (primary/secondary), MPa	19.6 / 0.36
Core inlet/outlet coolant temperature (°C)	311 / 328
Primary circulation	Natural circulation
Fuel type/assembly array	UO2 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
Fuel cycle requirements/approach	Initial factory load for the entire lifetime
Reactivity control mechanism	Control rods and absorber rods
Approach to safety systems	Passive
Seismic design (SSE)	VIII (MSK-64)
Design life (years)	25
Plant footprint (m^2)	N/A
RV height/diameter (m)	3.7 / 1.25
RV weight (metric ton)	N/A

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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.

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.

(b) Reactor Core

Pellet type uranium dioxide fuel is used with the average 235U 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 235U 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 Vessel (RV) 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 and Steam Generator

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) Pressuriser

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.

(g) Primary pumps

(h) Distinguishing features

25 years without refuelling, passive reactivity control and unattended operation

5. Safety Features

(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) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) Spent Fuel Cooling Safety Approach / System

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

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.

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.

Balance of Plant

- Turbine Generator Building: A TEG is used as a heat exchanger between circuits I and II; it is based on semiconductor thermo-elements enabling the generation of 68 kW of power in the reactor nominal operating mode simultaneously with heat transfer to circuit II. This power is used for plant auxiliary needs; it could also be supplied to a small town without district power supply, partially replacing a diesel power plant. The TEG consists of eight identical thermoelectric units (TEU). Each of them includes 36 thermoelectric modules equipped with thermoelectric packs of bismuth tellurides with electronic and hole conduction.
- Electric Power Systems: The ELENA-NTEP CSS has three independent power supply 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. Testing Conducted for Design Verification and Validation

10. Design and Licensing Status

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

11. 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.

12. 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.

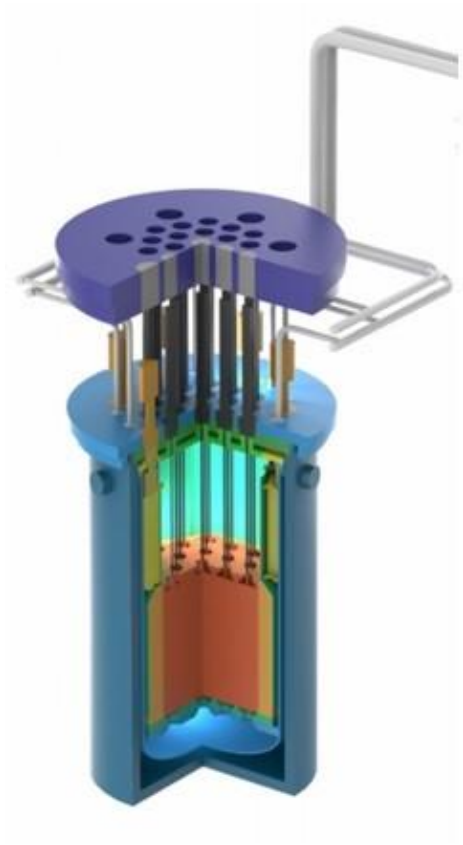
13. Development Milestones

Not determined

Centrum výzkumu Řež s.r.o., Czech Republic, Energy Well



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Centrum výzkumu Řež Czech Republic
Reactor type	Fluoride high temperature reactor
Coolant/moderator	Molten salt FLiBe / -
Design status	Conceptual design
Thermal/electrical capacity, MW(t)/MW(e)	650 / 700
NSSS operating pressure (primary/secondary), MPa	Atmospheric
Core inlet/outlet coolant temperature (°C)	650 / 700
Primary circulation	Forced circulation
Fuel type/assembly array	TRISO
Number of fuel assemblies in the core	25
Fuel enrichment (%)	15
Core discharge burnup (GWd/ton)	70
Refuelling cycle (months)	84
Fuel cycle requirements/approach	Once-through
Reactivity control mechanism	Control rods
Approach to safety systems	Active/passive
Seismic design (SSE)	-
Design life (years)	TBD
Plant footprint (m^2)	< 4,000
RV height/diameter (m)	7 / 3.5
RV weight (metric ton)	< 100

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, amon g others: 25 MW(t) maximum power; transportable; long fuel cycle; fuel enrichment < 20%; and thermal efficiency > 40%.

2. Target Application

☒ Electricity	☒ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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) Nuclear Steam Supply System

Power Conversion

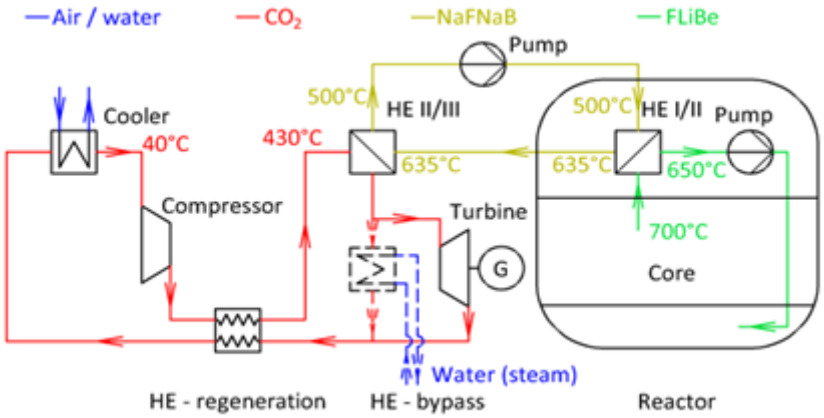


Figure 1. Process Flow Diagram of Energy Well reactor with simple heat recovery system

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.

(b) Reactor Core

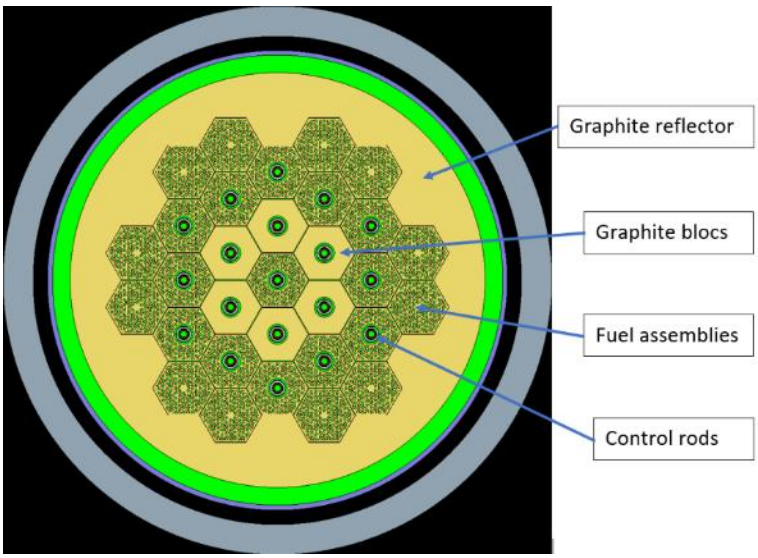


Figure 2. Core Arrangement

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.

(c) Reactivity Control

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) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

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.

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 ‘NaBF4’ salt is used for the secondary circuit with 384°C solidification temperature. Alternatively, LiF-BeF2 with a solidification temperature of 455°C and FLiNaK with a solidification temperature of 454°C, could be foreseen.

Steam Generator

The function of tertiary circuit is for the conversion of heat to electric energy with the sCO2 as working fluid. The sCO2 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 sCO2 circuits lie in their technical ‘immaturity’. The change of media properties significantly complicates the design of components. A range of the experimental facilities with a power between 100 kW and 10 MW are being built around the world in order to verify this technology.

There are dozens of possible sCO2 cycles layouts. Based on preliminary studies, the recompression Brayton cycle with heat regeneration was selected for the Energy Well system as reasonable compromise between the complexity and efficiency of the cycle. Relatively high thermodynamic efficiency of the cycle of 42.71% was reached for the nominal operational conditions.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

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

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) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) Spent Fuel Cooling Safety Approach / System

The spent fuel cooling system is still to be defined.

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 Systems

The I&C system is currently under development and being tested on the experimental loops currently operating in the Centrum výzkumu Řež s.r.o.

8. Plant Layout Arrangement

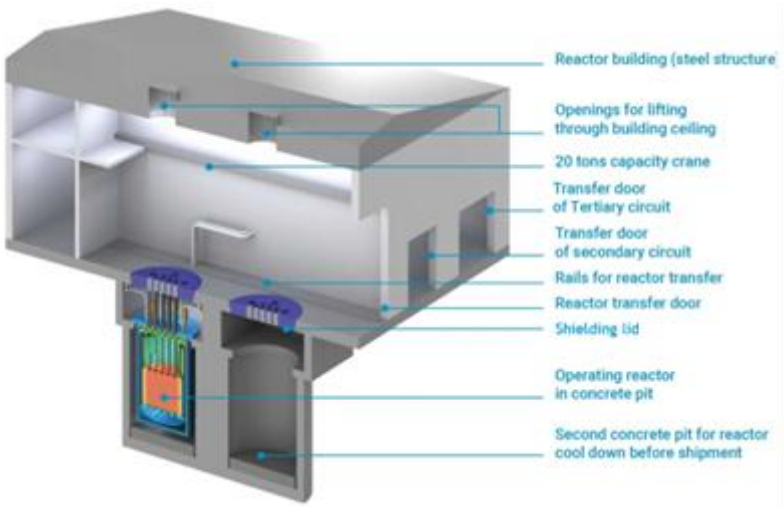


Figure 3. 3D View of Energy Well Reactor Building

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 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.

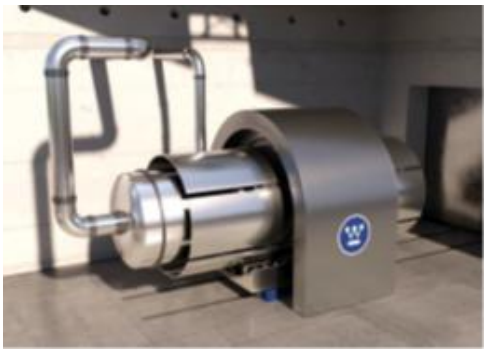
13.Development Milestones

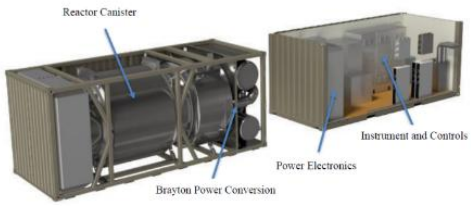
Year	Milestone	Status
2010-2017	Basic research in the neutronics, thermohydraulics and material compatibility in regard to the FLiBe salt	**Completed**
2017-2022	Pre-conceptual design phase and technology validation	**Completed**
2027-2032	Basic design	**Planned**
2030-2035	Experimental verification using an integral test facility	**Planned**
2035-2042	Finalization for manufacturing	**Planned**

Westinghouse Electric Company LLC, United States of America, Westinghouse eVinci™ Microreactor



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MAJOR 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
Design status	Detailed Design
Thermal/electrical capacity, MW(t)/MW(e)	15 / 5
NSSS operating pressure (primary/secondary), MPa	0.12 (primary)
Core inlet/outlet coolant temperature (°C)	N/A
Primary circulation	Natural
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
Fuel cycle requirements/approach	N/A
Reactivity control mechanism	Control drums (Primary) and rods (Defense-in-Depth)
Approach to safety systems	Inherent and passive safety for shutdown and heat removal
Seismic design (SSE)	1g pga
Design life (years)	8
Plant footprint (m^2)	< 2 000
RV height/diameter (m)	N/A
RV weight (metric ton)	N/A

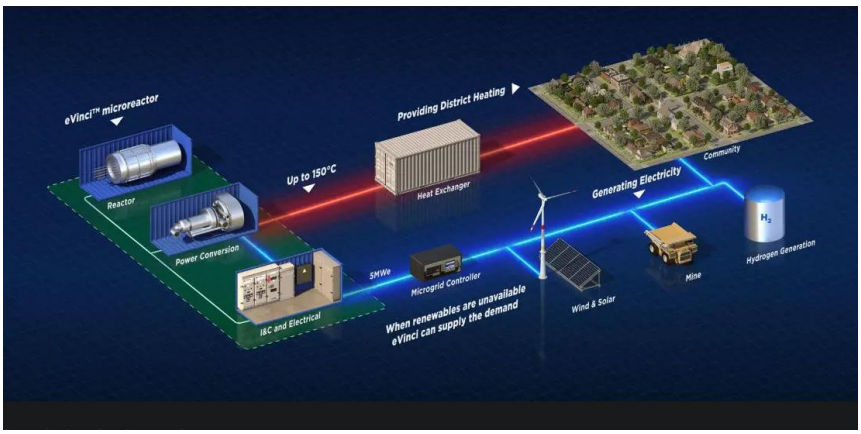
1. Introduction

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

☒ Electricity	☒ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

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

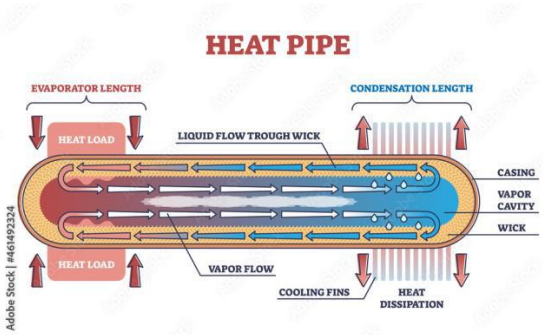


Figure 1. Circulation of working fluid in a heat pipe

(source: bajeczneobrazy.pl)

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.

(b) Reactor Core

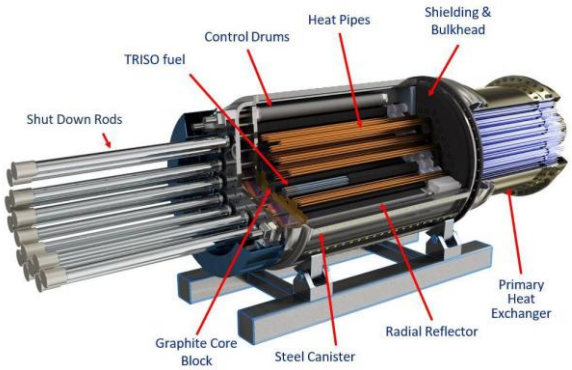


Figure 2. Sectional cut of the eVinci microreactor displaying the key technological components

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.

(d) Reactor Vessel (RV) and Internals

(e) Reactor Coolant System and Steam Generator

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.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

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) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) 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

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

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.Waste Management and Disposal Plan

The waste management and disposal plan are under development.

13.Development Milestones

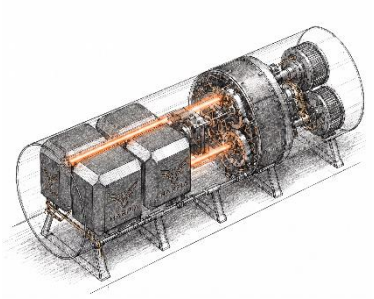
Year	Milestone	Status
2021	Electric demonstration	**Completed**
2027	First nuclear demonstration	**Planned**
2029	Commercial deployment	**Planned**

Terminus R&D in Energy S/A, Brazil
HARPIA™ Nuclear Microreactor



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HARPIA



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Terminus R&D in Energy S/A, Brazil
Reactor type	Heat Pipe Micro-Modular Reactor
Coolant/moderator	Heat pipes / BeO and Graphite
Design status	Basic Design
Thermal/electrical capacity, MW(t)/MW(e)	~ 15 / 5
NSSS operating pressure (primary/secondary), MPa	N/A
Core inlet/outlet coolant temperature (°C)	N/A
Primary circulation	Passive heat transfer via heat pipes, conduction, and radiation
Fuel type/assembly array	HALEU Fuel Rods Uranium Compound
Number of fuel assemblies in the core	TBD
Fuel enrichment (%)	Up to 19.75
Core discharge burnup (GWd/ton)	N/A
Refuelling cycle (months)	>120 months (>10 years)
Fuel cycle requirements/approach	Once-through, factory sealed core
Reactivity control mechanism	Control drums (Primary) and control/safety rods
Approach to safety systems	Inherent and passive safety for shutdown and heat removal
Seismic design (SSE)	TBD
Design life (years)	60
Plant footprint (m^2)	< 4,000
RV height/diameter (m)	N/A
RV weight (metric ton)	N/A

1. Introduction

The Brazilian Nuclear Microreactor (HARPIA™ – Heat-pipe Advanced Reactor for Power and Industrial Applications), currently under development by Terminus R&D in Energy, a company of the Diamante Group, is an advanced nuclear energy system designed to provide reliable, carbon-free, and economically competitive electricity through a compact, transportable, and inherently safe architecture. Developed to support Brazil's energy security, technological sovereignty, and decarbonization goals, the HARPIA™ combines advanced nuclear technologies with modular manufacturing, factory fabrication, and rapid deployment capabilities.

The reactor incorporates a Safe-by-Design philosophy and a Defence-in-Depth safety approach, utilizing High-Assay Low-Enriched Uranium (HALEU) fuel to enable operating cycles exceeding ten years while meeting international non-proliferation requirements. Designed to generate approximately 5 MWe, the system provides continuous baseload power

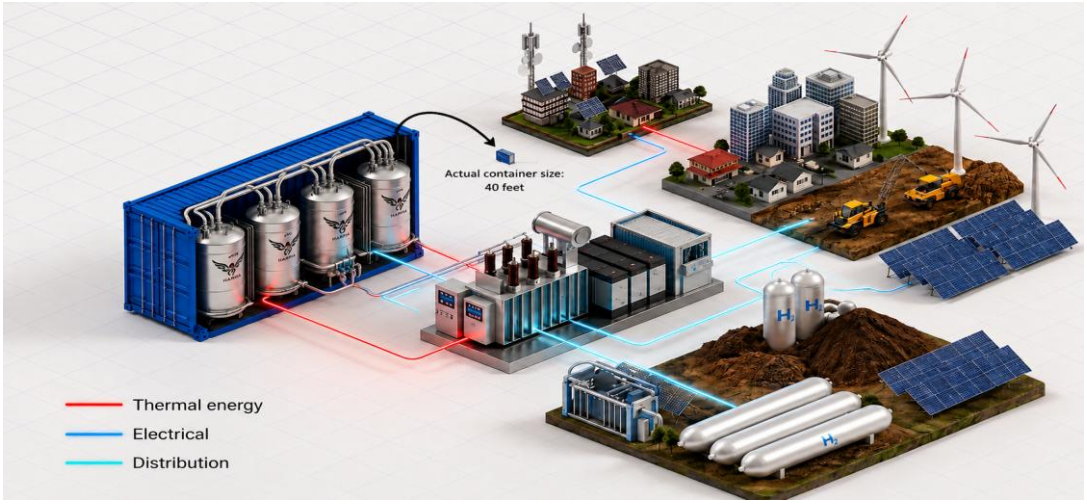
with high availability. Its compact footprint, compatibility with standard 40-foot container transportation, and Plug-and-Play deployment concept enable efficient logistics, installation, and commissioning in less than 30 days after site preparation.

Following an Economics-by-Design approach, the HARPIA™ integrates cost optimization through long fuel autonomy, modular construction, remote operation capabilities, and simplified plant systems. The reactor requires no water for its primary operational and safety functions and is expected to achieve a service life exceeding 60 years. The project is intended to be designed, manufactured, licensed, and operated in Brazil, strengthening the national nuclear industry while providing a resilient and sustainable energy solution.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☒HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☒ >550°C

The Brazilian Nuclear Microreactor (HARPIA™) is designed to provide reliable, sustainable, and carbon-free energy for a broad range of applications requiring resilient and distributed power generation. With an electrical output of approximately 5 MWe, the reactor is particularly well suited for remote communities, mining and resource extraction operations, industrial facilities, isolated microgrids, and other locations where conventional energy solutions are technically challenging or economically unattractive.



Its capacity aligns closely with the Brazilian demographic profile, as nearly 68% of the country's municipalities have fewer than 20,000 inhabitants, creating significant opportunities for distributed energy generation capable of meeting local demand without major investments in transmission infrastructure.

The HARPIA™ is designed to support both baseload and load-following operation, providing continuous electricity with high availability for mission-critical infrastructure such as hospitals, data centres, telecommunications facilities, ports, airports, and other strategic assets requiring long-term energy security and independence.

The microreactor can also serve as the backbone of hybrid off grid microgrids, integrating with renewable energy sources such as solar and wind power to improve grid stability and reliability in remote regions, isolated industrial complexes, and communities beyond the reach of centralized transmission networks.

In addition to electricity generation, the HARPIA™ is capable of supplying high-temperature process heat for industrial applications, including chemical, metallurgical, mining, and cement production processes, as well as supporting advanced energy applications such as hydrogen production through high-temperature electrolysis. Its design also enables deployment in support of electric mobility and logistics infrastructure, including high-power charging hubs for transportation fleets operating in regions with limited grid capacity.

The reactor is particularly attractive for locations where conventional thermal power plants are impractical, as its operation and safety systems do not require water for cooling. Combined with advanced passive safety features, long fuel autonomy, rapid deployment, and a compact footprint, the HARPIA™ provides a flexible solution for distributed power generation, industrial decarbonization, critical infrastructure support, and energy resilience.

Beyond remote and industrial applications, the HARPIA™ can contribute to regional economic development and energy equity. This is especially relevant in Northeastern Brazil, a region that currently exports electricity to the national grid while

still exhibiting substantial latent energy demand. By enabling reliable access to affordable, carbon-free energy, the microreactor can support industrialization, economic growth, and improvements in quality of life while strengthening Brazil's leadership in advanced nuclear technologies and supporting national decarbonization goals.

3. Design Philosophy

The Brazilian Nuclear Microreactor (HARPIA™) is founded on a design philosophy that integrates safety, simplicity, deployability, and economic competitiveness from the earliest stages of development. Inspired by the successful application of heat-pipe reactor technology and guided by an Economics-by-Design approach, the reactor minimizes system complexity by eliminating the need for bulk primary coolant systems, pumps, and extensive auxiliary equipment. Heat is removed from the core through passive heat-transfer mechanisms, reducing the number of moving parts and enhancing overall plant reliability.

The design incorporates Safe-by-Design and Defence-in-Depth principles, leveraging the inherent safety characteristics of the fuel, moderator, and passive heat removal systems to provide strong self-regulation capabilities and robust protection under both normal and off-normal conditions. In parallel, the HARPIA™ emphasizes modularity, factory-based manufacturing, transportability, rapid deployment, and remote operation, enabling a compact and resilient energy solution capable of supplying continuous carbon-free electricity to remote communities, industrial facilities, critical infrastructure, and isolated microgrids. This philosophy ultimately seeks to maximize safety, affordability, operational simplicity, and national technological sovereignty while establishing a scalable and sustainable platform for advanced nuclear energy deployment in Brazil.

4. Main Design Features

(a) Nuclear Steam Supply System

The Brazilian Nuclear Microreactor (HARPIA™) employs a fully passive heat transport system based on heat pipe technology to remove thermal energy directly from the reactor core and transfer it to the power conversion system. Heat pipes operate through natural evaporation and condensation processes within sealed units, eliminating the need for pumps, large coolant inventories, and complex primary cooling circuits. The same passive system also provides residual heat removal following reactor shutdown, ensuring continued core cooling under both normal and off-normal conditions. This approach significantly enhances reliability, reduces maintenance requirements, and supports autonomous operation while reinforcing the reactor's Safe-by-Design philosophy.

Power Conversion System

The Power Conversion System (PCS) converts the thermal energy produced by the reactor into electricity through a highly integrated and efficient energy conversion process. The PCS is designed to operate in conjunction with the passive heat transport system, enabling continuous and reliable baseload power generation with minimal operational complexity. Its modular configuration supports factory assembly, rapid installation, and simplified maintenance while contributing to the overall compact footprint of the plant. Together with remote monitoring and operational capabilities, the PCS enhances system reliability and enables deployment in remote, isolated, and infrastructure-constrained locations.

The integration of passive heat transport, long-life HALEU fuel, inherent safety characteristics, modular construction, and simplified operation enables the Brazilian Nuclear Microreactor to provide a resilient, carbon-free, and economically competitive energy solution. These design features support rapid deployment, long-term autonomous operation, and broad applicability across remote communities, industrial facilities, critical infrastructure, and distributed energy markets.

(b) Reactor Core

The reactor core is designed around the use of High-Assay Low-Enriched Uranium (HALEU) fuel, enabling operating cycles exceeding ten years without refuelling. The compact core configuration is optimized to achieve high energy density, efficient fuel utilization, and stable long-term operation while delivering approximately 5 MWe of electrical power. Inherent reactivity feedback mechanisms provide strong self-regulation characteristics, contributing to safe and predictable reactor behaviour throughout the operating cycle. A compact core design also supports factory fabrication, transportation, and deployment within a modular microreactor architecture.

(c) Reactivity Control

Reactivity control and reactor shutdown are accomplished through dedicated control and shutdown systems designed in accordance with Safe-by-Design and Defence-in-Depth principles. These systems provide precise control of reactor power

during operation and ensure safe shutdown capability under all operating conditions. Combined with the reactor's inherent negative reactivity feedback characteristics, the control system minimizes reliance on operator actions and active safety systems while maintaining robust protection against abnormal operational events.

(d) Reactor Vessel (RV) and Internals

(e) Reactor Coolant System and Steam Generator

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The Brazilian Nuclear Microreactor (HARPIA™) adopts a Safe-by-Design philosophy in which safety is integrated into the reactor architecture from the earliest stages of development. The design combines inherent safety characteristics, passive heat removal systems, and a Defence-in-Depth approach to minimize reliance on active safety systems and operator actions. Reactor resilience is achieved through a compact, modular configuration incorporating passive heat pipe technology for both normal heat transport and residual heat removal. The reactor is designed to operate autonomously with advanced monitoring and control systems, enabling reliable load-following operation and seamless integration with isolated grids, microgrids, and distributed energy systems. These characteristics allow the HARPIA™ to provide highly reliable electricity while maintaining robust safety margins under both normal and off-normal operating conditions.

(b) Decay Heat Removal System

A key safety feature of the HARPIA™ is its passive decay heat removal capability. Following reactor shutdown, residual heat continues to be removed through the same heat pipe system used during normal operation, eliminating dependence on pumps, external power supplies, or operator intervention. The passive heat transfer mechanisms ensure continuous cooling of the reactor core and contribute to maintaining fuel and structural temperatures within safe limits under both anticipated operational occurrences and postulated accident conditions. This capability significantly enhances overall plant safety and resilience.

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) Containment System

The reactor incorporates multiple physical barriers to ensure the confinement of nuclear fuel and radioactive materials throughout its operational lifetime. The reactor core, fuel, heat transport system, and associated safety structures are housed within a robust containment and shielding configuration specifically designed to withstand internal and external hazards. The modular reactor unit is intended to be installed within a protected structure that provides additional physical protection against environmental events, external impacts, unauthorized access, and other potential threats. This layered protection strategy preserves the integrity of the reactor system and prevents the release of radioactive materials under credible accident scenarios.

(g) Spent Fuel Cooling Safety Approach / System

The HARPIA™ utilizes High-Assay Low-Enriched Uranium (HALEU) fuel designed to support operating cycles exceeding ten years without refuelling. Fuel handling activities are minimized through the long-life core concept, reducing operational complexity and transportation requirements. At the end of the operating cycle, spent fuel management will be conducted in accordance with Brazilian regulatory requirements and international nuclear safety standards. Spent fuel will be transferred to licensed storage facilities using certified transport and storage systems until a long-term national disposal solution becomes available. This approach supports safe, secure, and environmentally responsible management of nuclear materials throughout the reactor lifecycle.

6. Plant Safety and Operational Performances

The Brazilian Nuclear Microreactor (HARPIA™) has been conceived with a strong emphasis on inherent and passive safety, aiming to achieve exceptionally low probabilities of core damage and radiological release. The reactor employs an unpressurized primary system with high thermal inertia, eliminating the risks associated with pressurized coolant systems entirely.

Decay heat removal is accomplished through fully passive systems based on natural circulation and heat conduction mechanisms, ensuring safe reactor cooling without reliance on external power supplies or operator intervention. In station blackout scenarios, passive heat removal systems provide extended grace periods and maintain fuel and structural temperatures within acceptable limits.

The use of Uranium Compound fuel provides an additional robust barrier against fission product release, retaining radionuclides even under severe accident conditions. Multiple physical barriers, combined with passive reactivity feedback mechanisms and a below-grade reactor installation, contribute to a high level of defence-in-depth and overall plant resilience.

7. Instrumentation and Control Systems

The HARPIA™ Instrumentation and Control (I&C) architecture is being developed in accordance with modern nuclear safety standards, incorporating redundancy, diversity, and cyber-secure digital technologies.

The reactor protection system continuously monitors key plant parameters, including neutron flux, coolant temperatures, coolant flow rates, reactor power, and structural integrity indicators. Independent shutdown channels provide reliable reactor trip capability under all operational and postulated accident conditions.

The plant is designed for highly autonomous operation, supported by advanced monitoring, diagnostics, and predictive maintenance systems. Remote monitoring capabilities allow centralized supervision while maintaining local safety functions fully independent and operational.

Radiation monitoring systems continuously survey plant areas and environmental release pathways to ensure compliance with regulatory limits and operational safety requirements..

8. Plant Layout Arrangement



The HARPIA™ plant adopts a compact and modular configuration designed to simplify transportation, installation, operation, and decommissioning. The reactor module, primary system, biological shielding, and major safety systems are integrated into a factory-fabricated unit that is transported to the site largely assembled.

The reactor is installed within a reinforced containment structure that provides protection against external hazards and extreme weather events.

Balance-of-plant systems, power conversion equipment, auxiliary systems, and the control room are arranged in separate structures to optimize safety, maintainability, and operational efficiency. The modular architecture minimizes on-site construction activities and enables rapid deployment in remote and isolated locations.

The plant footprint is significantly smaller than conventional nuclear power stations, making it suitable for deployment in mining sites, isolated communities, industrial facilities, and remote microgrids.

9. Testing Conducted for Design Verification and Validation



The HARPIA™ development program includes a comprehensive verification and validation campaign covering reactor physics, thermal-hydraulics, fuel performance, structural integrity, and safety system performance.

Advanced computational models are being employed to evaluate steady-state operation, transient behaviour, and accident scenarios. These analyses are supported by experimental programs involving scaled thermal-hydraulic facilities, materials testing, and component qualification activities.

The HARPIA™ Verification and Validation (V&V) programme is structured through dedicated Technology Development Units (TDUs) for progressive engineering risk mitigation:

WP1. Critical Assembly: Construction and operation of a zero-power Critical Assembly to provide experimental validation of core neutronics, reactivity coefficients, control rod worth, shutdown margins, power distributions, and benchmarking of computational methods and nuclear data libraries used in the reactor design process.

WP2. Separate Effects Testing — Heat Pipes: Laboratory-scale experimental facilities dedicated to the thermodynamic characterization of heat pipes, including determination of capillary limits, startup behaviour, thermal transport capacity, operational stability, and verification of working fluid performance across the full range of design temperatures.

WP3. Integrated Heat Transfer Testing: Experimental facilities dedicated to the validation of coupled thermal-mechanics/thermal-hydraulic performance involving the reactor core simulator, heat pipes, intermediate heat exchangers, and power conversion interfaces. These tests are intended to demonstrate integrated heat removal capability, thermal stability, load-following behaviour, and overall system efficiency under representative operating conditions.

WP4. Protection, Control, and Remote Supervision Systems: Development and validation of advanced instrumentation and control systems capable of supporting autonomous operation, remote monitoring, cybersecurity requirements, fault diagnosis, predictive maintenance, and integration with isolated microgrids and distributed energy systems.

WP5. Engineering Design: Development and maturation of the engineering design for the HARPIA™ microreactor, covering both the pilot demonstration unit and future commercial plants. Activities include the conceptual, basic, and detailed design of the reactor core, heat transport and power conversion systems, instrumentation and control, safety systems, balance-of-plant, and microgrid integration. The work package also encompasses design optimization through analyses and testing feedback, incorporation of licensing requirements, and preparation of the technical documentation required to support technology qualification, prototype construction, and commercial deployment, with emphasis on safety, reliability, modularity, manufacturability, transportability, and economic competitiveness.

WP6. Advanced Materials Development and Qualification: Dedicated research and qualification programs for key reactor materials, including nuclear-grade graphite, beryllium oxide (BeO), boron carbide (B4C), heat pipe materials and working fluids, as well as advanced nuclear fuels. Activities include fabrication process development, thermo-physical characterization, irradiation testing, and long-term performance assessments.

WP7. Sustainability and Lifecycle Assessment Studies: Comprehensive evaluations addressing environmental performance, resource utilization, fuel cycle optimization, waste generation, decommissioning strategies, economic competitiveness, and overall sustainability of microreactor deployment throughout the reactor lifecycle.

10.Design and Licensing Status

The HARPIA™ microreactor is currently progressing through conceptual and preliminary engineering development phases under a coordinated national research and development program involving Brazilian research institutions, industrial partners, and government stakeholders.

Licensing activities are being conducted in close interaction with the Brazilian National Agency for Nuclear Safety (ANSN), with the objective of establishing a licensing framework appropriate for advanced microreactor technologies. The development program incorporates regulatory engagement from the early design stages to facilitate efficient licensing and deployment.

Ongoing activities include reactor design optimization, safety analysis development, fuel qualification studies, technology demonstration planning, supply chain assessment, and site selection methodologies for future demonstration projects.

The long-term objective is to establish a fully domestically developed and manufactured microreactor technology capable of supporting Brazil's energy security, technological sovereignty, and decarbonization goals.

11.Fuel Cycle Approach

The HARPIA™ fuel cycle strategy prioritizes operational simplicity, long refuelling intervals, and the elimination of on-site fuel handling activities, making it particularly suitable for remote and distributed energy applications.

The commercial version of the reactor is designed to utilize High-Assay Low-Enriched Uranium (HALEU) fuel, with uranium enrichment below 20%, enabling the high-power density required for a compact core and extended operational lifetimes. The fuel and core design are intended to support continuous operation for periods potentially exceeding ten years without refuelling, depending on the final reactor configuration and mission profile.

The project is founded on the long-term objective of establishing a complete domestic capability for the production and enrichment of HALEU fuel in Brazil, while maintaining the flexibility to procure fuel from international suppliers when strategically advantageous and consistent with applicable regulations and non-proliferation commitments.

The reactor employs a high-performance fuel system designed to maintain structural integrity and retain fission products under both normal operating conditions and postulated accident scenarios. This approach contributes to enhanced safety, operational reliability, and reduced maintenance requirements throughout the reactor lifetime.

At the end of the fuel cycle, the entire nuclear module—including the sealed reactor core and associated shielding structures—can be removed during a scheduled refuelling outage and transported to a centralized, licensed nuclear facility for interim storage, processing, recycling, or final disposal, in accordance with Brazilian regulatory requirements. This modular approach minimizes fuel handling at the deployment site, reduces transportation frequency, and simplifies end-of-life logistics.

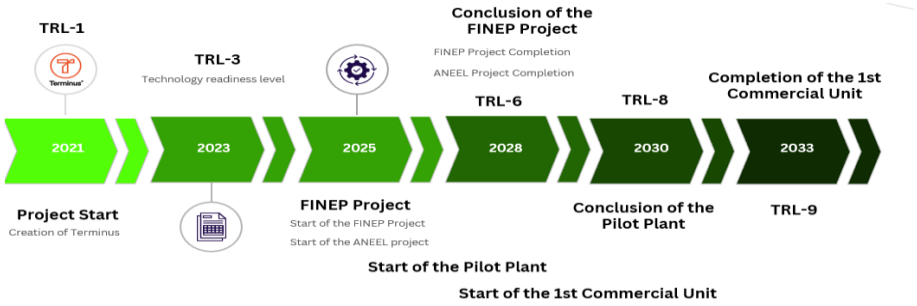
Overall, the HARPIA™ fuel cycle strategy seeks to maximize fuel utilization, minimize waste generation and operational complexity, and provide a practical and economically attractive solution for long-duration deployment in remote locations while supporting the development of a sustainable national nuclear fuel cycle infrastructure.

12.Waste Management and Disposal Plan

The HARPIA™ Waste Management and Disposal Plan is currently under development and will be finalized during the detailed design and licensing phases in coordination with the relevant Brazilian regulatory authorities.

The plan will address the complete lifecycle management of radioactive materials generated during operation, maintenance, refuelling, transportation, storage, decommissioning, and final disposal activities. Its primary objectives are to protect workers, the public, and the environment, while ensuring compliance with Brazilian nuclear safety, radiation protection, transport, and waste management regulations, as well as applicable international standards.

13.Development Milestones

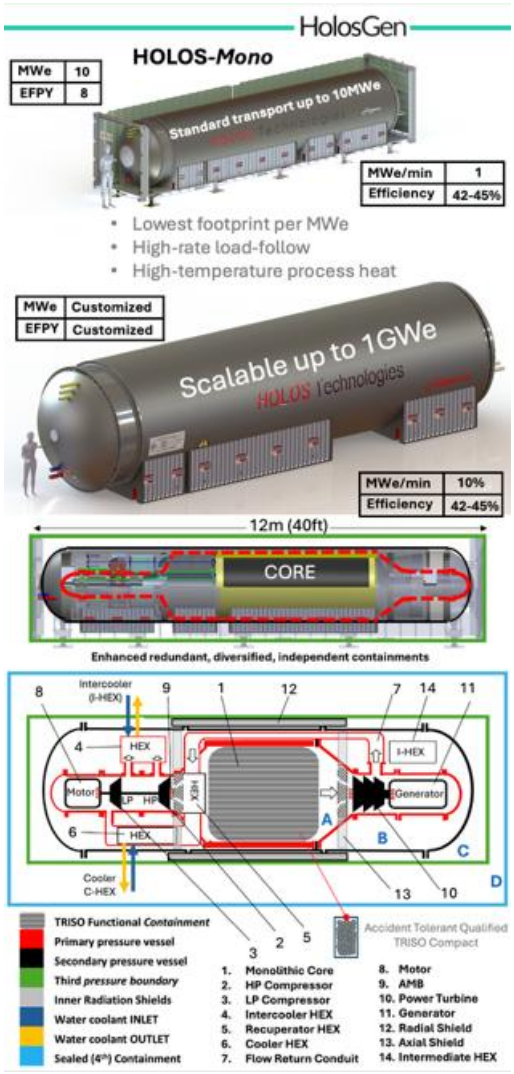


Year	Milestone	Status
2021	Terminus P&D founded; basic principles observed and reported / TRL-1	Completed
2022	Technology concept and application formulated / TRL-2	Completed
2023	Analytical modelling and experimental proof of concept validated / TRL-3	Completed
2025	FINEP and ANEEL R&D programmes; component validation - laboratory environment / TRL-4	Ongoing
2027	Key components validated in a relevant environment (heat pipes, control systems) / TRL-5	Ongoing
2028	Pilot plant construction; system prototype demonstrated - relevant environment / TRL-6	Ongoing
2030	Pilot plant commissioned; system qualified through test and demonstration; FINEP and ANEEL programmes concluded / TRL-7 & TRL-8	Planned
2033	First commercial unit completed and proven through operational deployment / TRL-9	Planned

HolosGen, United States of America
HOLOS-Monolithic



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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.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	HolosGen, USA
Reactor type	HTGR
Coolant/moderator	Helium / graphite
Design status	Basic Design
Thermal/electrical capacity, MW(t)/MW(e)	22 / 10
NSSS operating pressure (primary/secondary), MPa	Full Brayton power cycle: Max 7, Min 3.5. Helium loop comprises power conversion system – no traditional Balance of Plant.
Core inlet/outlet coolant temperature (°C)	590 / 855
Primary circulation	N/A
Fuel type/assembly array	Qualified TRISO UCO compacts
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	19.95
Core discharge burnup (GWd/ton)	82
Refuelling cycle (months)	96 at EFPY operating conditions
Fuel cycle requirements/approach	Once-through
Reactivity control mechanism	Redundant control drums and shutdown rods
Approach to safety systems	Passive decay heat removal via natural cooling mechanisms
Seismic design (SSE)	4g (rail transport shocks limit)
Design life (years)	40 to 60
Plant footprint (m^2)	5000 (stationary 10MWe underground version) 7000 (30MWe & ISFSI)
RV height/diameter (m)	9.5/2.3 (individual SPM)
RV weight (metric ton)	60 operational

1. Introduction

The HOLOS-Monolithic 10MWe (“HOLOS-Mono-10”) design is a scalable reactor featuring high-efficiency electricity and variable temperature process heat supply to closely match customer and application-specific power profile requirements.

The HOLOS-Mono-10 main configuration leverages the innovative integral power conversion system developed for the HOLOS-Quad design, seamlessly interfaced with commercial power conversion components, thus reducing technological risk, capital cost and deployment time. While the HOLOS-Quad features multiple independent scaled-down Sealed Power Modules (SPMs) enabling rapid transport of individual irradiated SPMs for reassembly and operation at a new deployment location, the HOLOS-Mono-10 design utilizes a single SPM housing a traditional monolithic core. This traditional core configuration requires a longer "core cooling period" to enable transport after power operations. The HOLOS-Mono-10 design advances and optimizes the High Temperature Gas Reactor (HTGR) typology utilizing qualified Accident Tolerant Fuel (ATF), TRistructural ISOtropic (TRISO). This fuel is formed by ceramic, millimeter-size particles that surround and seal nucler fuel as well as fission product elements. Under severe design basis accident scenarios the TRISO ceramic particles do not melt as they can withstand extreme temperatures well beyond the threshold of traditional nuclear fuels. TRISO fuel does not require containment as high radionuclide retention capability has been demonstrated, qualifying it as "functional containment". Although with very low probability, undetected fuel damage occurred during manufacturing, handling, or installation, radionuclides generated during operation might migrate from such damaged fuel into the working fluid. Just as is the case for commercial water reactors equipped with a Chemical and Volume Control System (CVCS) to monitor, filter and trap radionuclides and clean-up the working fluid during operations, HTGRs may also be equipped with similar CVCS, specialized to handle gaseous working fluids. In the HOLOS-Mono-10 design, the CVCS is integrated altogether with the helium inventory control system and the containment functions are passively enhanced through multiple, low-cost, diversified and independent pressure boundaries surrounding the core to further strenghten the "defense-in-depth" safety principle applied throghout the design. The reactivity control of the HOLOS-Mono-10 core utilizes redundant, diversified and resilient control mechanisms comprising control drums for regulation and shutdown, shutdown rods with both passive and active insertion, with both control drum and shutdown rods mechanisms capable of achieving their safety functions even under several permutations of reactivity control mechanisms malfunctions, thus providing further redundancy and independence of these safety functions. The core is also equipped with burnable neutron absorbers to ensure reactvity remains within safety margins under design basis accident scenarios incuding sudden water flooding when the core is at its highest reactvity. The HOLOS-Mono-10 core design utilizes helium as primary coolant circulating through graphite moderated TRISO fuel with burnable neutron absorbers to optimize core perfromance over extended refueling cycle. For power rating up to 10MWe the design can be transported as a comprehensively integrated reactor system fully contained within an intermodal shipping container. Special transport is required for scaled up configurations of the design with power rating >10MWe. The design can be scaled to 80MWe per individual unit, comparable to the power rating of very-Small Modular Reactors (vSMRs), further scaled up to 320MWe per unit, comparable to the power rating of Small Modular Reactors (SMRs). The design is scalable all the way to 1.28GWe per unit, comparable to the power rsating of Generation III+ Large Modular Reactors (LMRs), while conserving the integrated power conversion features and the smallest plant footprint per MW generated.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☒ <200°C
☒ Load-following	☐PEM - polymer membrane	☒ 200–550°C
☒ District Heating / Steam	☐ Desalination	☒ >550°C

The HOLOS-Mono-10 features high-efficiency load-follow electricity, maximizing fuel utilization and reducing spent fuel production per unit of generated core power. Intermediate heat exchangers with working fluid selected by customers, are thermally coupled to the Brayton heat rejection and intercooler heat exchangers to supply high-temperature process heat to support non-carbon fuel production, steelmaking and other energy intense industrial applications, and/or low-temperature process heat to support, for example, chillers cooling of data center servers. This configuration result in a very high efficiency data center efficiency. The HOLOS-Mono-10 system/sub-system prototype has been demonstrated in an operational environment qualifying it to TRL6 (Technology Readiness Level, 1-9 scale per NASA definition). This achievement retires technological risks associated with new and innovative technologies and represents approximately a decade of successfully completed R&D activities. The average cost of achieving TRL6 validated by independent authoritative parties for these typologies of reactors may exceed $500M. The HOLOS-Mono-10 scalability, customization and capability to support variable electrical loads makes it ideal to satify a vast portfolio of markets. For example, electric power generation for computer servers supporting Artificial Intelligence with cooling of the Data Center's equipment supported by <200°C process heat offered by the design heat rejection. Additonal, applicatoins include, disctrict heating, power production aboard ships, offshore rigs, on-demand high-rate load-follow power compensating for the intermittent power generated by renewable energy sources, industrial process heat applications in the temperature range of 200°C-

550°C, and applications requiring >550°C process heat to decarbonize energy-intensive industrial processes such as steel making and hydrogen generation.

3. Design Philosophy

The philosophy for all HOLOS design configurations integrates “safety at all costs”, paramount “respect for the public and the environment”, and reliance on “mature-validated technologies only” to substantially “reduce technical and economic risks”.

Safety at all costs for all HOLOS designs and configurations translates into permeating the design philosophy with inherent safety measures at the individual Systems, Structures and Components (SSCs) level, regardless of stressors often induced by economic and/or political decisions. Safety is prioritized above all other considerations with enhancement of the defense-in-depth passive safety philosophy. For example, although the fuel adopted is “self-contained” TRISO, the design features two pressure vessels: The reactor pressure vessel, comprising and sealing the core and the power conversion components, further surrounded and sealed by a secondary pressure vessel. The secondary pressure vessel function is to protect the reactor pressure vessel and capture potential working fluid leaks should design basis accident events result in working fluid contamination and leakage from the reactor pressure vessel. Once a certain threshold of working fluid is collected within the volumes formed between the secondary and the reactor pressure vessels, a compressor activates to pump this inventory of working fluid back into the primary working fluid inventory control system for reutilization. This approach further enhances the defense-in-depth principle as self-contained TRISO fuel provides the first barrier to radionuclide migration into the working fluid (e.g., from potentially defective or damaged fuel elements). For traditional HTGRs, potentially contaminated working fluid remains trapped within the reactor pressure vessel and the pressure boundaries represented by the traditional Balance of Plant (BoP). Therefore, the reactor pressure vessel represents the second barrier preventing radionuclides migration potentially transported by the working fluid into the environment. This safety feature is considered adequate and in compliance with safety standards. However, the HOLOS design provides an additional third barrier to address potential working fluid contamination originating from damaged TRISO fuel. This barrier is represented by the secondary pressure vessel as it intercepts, captures and pumps potentially contaminated working fluid back to the reactor pressure vessel. For subterranean power station configurations, the HOLOS designs includes a fourth barrier, represented by a sealed reactor cavity housing the secondary and reactor pressure vessels during operations. The reactor cavity is saturated with helium at slightly above atmospheric pressure. This feature prevents environmental air ingression into the reactor cavity and the possibility for air to be subjected to irradiation in the proximity of the reactor pressure vessels components housing the core. Should air breach the reactor cavity seals, a natural circulation chimney enables ejection of potentially activated air through a stack enabling sufficient time for natural decay to neutralize the contaminated air prior to returning to ground level. Although the design is passively able to dissipate decay heat through natural heat transfer mechanisms, the design is equipped with redundant enhanced passive heat transfer features ensuring decay heat is removed even under substantial damage of the heat cavity, the secondary and reactor pressure vessels, potentially resulting from hostile actions (e.g., shape charge, kinetic events, and design basis accident scenarios involving piercing of multiple engineered barriers). The design provides these enhanced protections and mitigation features with respect to regulatory requirements in response to world events wherein intentional hostile targeting of nuclear power plants represents a new category of “beyond design basis accident” scenarios. The design is further protected from sudden loss of electrical load, does not depend on offsite or onsite electric power to safely startup and shut down, and is equipped with redundant hard-wired monitoring and control systems.

The philosophy addressing HOLOS designs respect for the public and the environment in addition to the Safety at all costs philosophy, factors “cradle-to-grave” Life Cycle Analysis (LCAs) for the SSCs forming the whole HOLOS power station to decarbonize SSCs manufacturing, installation, and Deactivation and Decommissioning (D&D) activities. This philosophy is implemented by the design reduction in the total amount of steel alloys generally employed by the traditional BoP, with further reductions on steel for rebars and substantial volume reductions of concrete, only utilized for prefabricated footers and lining of the power station floor and reactor cavity. Streamlining, simplifying and facilitating D&D activities further reduces the total volume of low-level radiation materials to be transported for decontamination to specialized facilities. This approach accelerates and reduces costs associated with the restoration of the reactor site to original conditions at the end of the station life in full respect of the environment. Additional environmentally beneficial aspects resulting from HOLOS design philosophies include the reduction of spent fuel production and thermal pollution per MWe generated mainly due to the high-power conversion efficiency offered by the design.

The HOLOS design philosophy addressing the reduction of technical risks involves the adoption of SSCs formed by proven technologies, or technologies with a maturity whose Technology Readiness Level (TRL) is minimum 5, verified by independent authoritative parties (e.g., peer-reviewed publicly accessible technical reports). This portion of the HOLOS design philosophy ensures that the technological risk is eliminated and the schedule for commercial deployment is not impacted by generally over optimistic marketing claims.

Finally, the HOLOS designs economic philosophy leverages decades of analyses dedicated to the identification of cost avoidance opportunities obtained through the development of economic models to analyze Generation III, III+ and IV

reactor designs at the individual SSC level. The results of these analyses indicated that for reactors to be competitive and environmentally friendly the power conversion efficiency must be greater than 40% and remain at comparable levels during power maneuvering to maintain high fuel utilization, minimize fuel costs, spent fuel volume, handling and storage costs, and reduce thermal pollution. High net plant efficiency induces spent fuel volume reductions per MW generated, along with associated reduced handling, transport and storage costs. Overall, manufacturing, transportation, capital, operation, maintenance and D&D costs are substantially reduced through the elimination of the BoP and integration of the PCS.

The overarching design philosophy prioritizes safety under all operating, shutdown, transport and storage modes. By design the qualified TRISO fuel cannot experience challenging temperatures under both design basis and beyond design basis accident scenarios. Passive cooling provides heat removal under loss of coolant and total loss of electricity. The HOLOS-Mono-10 core operates at a relatively low power density with a “thermal flywheel” represented by the graphite moderator preventing rapid temperature transients under all normal and off-normal operational scenarios. Under worst case scenarios with total loss of coolant, the fuel maximum temperature temporarely reaches approximately 1350°C, well below TRISO fuel melting temperature. Overall, all HOLOS designs adopt a multi pillar philosophy to simultaneously satisfy very conservative requirements associated with safety and respect for the public and the environment, while offering a competitive electric and process heat generator with reduced or eliminated technical and economic risks.

4. Main Design Features

(a) Nuclear Steam Supply System

Thermal-to-Electric Power Conversion System

HOLOS-Mono-10 integrates the components forming a Brayton cycle. The intercooler and recuperator heat exchangers altogether with the turbomachinery representing the compressor and expander-generator monolithic core, provide a unique design architecture enabling the elimination of the traditional Balance of Plant, thus reducing the working fluid inventory, the overall plant footprint and cost. The compressor turbomachinery is driven by a high-speed electric motor, mechanically decoupling the expander. This configuration enables unique power control features, eliminates the gearbox normally coupling the compressor to the expander turbomachinery along with bearings, lubrication system and control. By enabling independent speed control of the compressor and the expander rotors, the design results in the highest thermal-to-electric conversion efficiency while enabling high efficiency during load-follow power maneuvering to match electric demand at high rate.

(b) Reactor Core

HOLOS-Mono-10 core is helium cooled, graphite moderated and fueled with commercially qualified accident tolerant TRISO fuel. The core optimized for the 22MWt, 10Mwe transportable configuration is further equipped with burnable neutron absrobers optimizing core burnout with approximately 8 years of thermal-power at Effective Full Power Years (EFPY) conditions, while ensuring shutdown margins under required regulatory design basis acciddent conditions, including sudden water flooding event occurring when the core is at its maximum reactivity. Safety and technical performance parameters have been independently verified by US national laboratories and academic institutions.

(c) Reactivity Control

Reactivity is controlled by independent, diversified and redundant reactivity control mechanisms based on independent banks of control drums and shutdown rods with assured negative reactivity insertion under failure of a number of control drum and shutdown rod banks, and under sudden moderation surge due to water flooding of the core.

(d) Reactor Vessel (RV) and Internals

HOLOS-Mono-10 is equipped with a non-traditional, simpler to manufacture reactor pressure vessel, forming a Sealed Power Module (SPM) comprising the full Brayton 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 raminaing majority of the vessel operating at 3.5MPa. In the HOLOS-Monolithc-10 design, the control drums and shutdown rods are also housed within the primary pressure vessel. A secondary pressure vessel, fully surrounding and sealing the reactor pressure vessel, represents an additionl barrier to potential radionuclide migration (e.g., from defective or damaged TRISO fuel undetected during manufacturing, hanldling and/or core loading). During normal operations, helium potentially leaking out of the reactor pressure vessel is captured within the secondary pressure vessel and pumped back to maintain helium inventory at minimum. During accident scenarios, the secondary pressure vessel intercepts and contains potential radionuclides for processing and disposal. For subterranean installations, a third barrier to radionuclide migration is represented by a selaed reactor cavity, comprising the secondary pressure vessel. The sealed reactor cavity is kept sightly pressuried during operations. The secondary pressure vessel and the sealed reactor cavity prevent air ingression and irradiation in the proximity of the core and represent additional low-cost embedded containment systems enhancing safety performance beyond that normally required to obtain a license to commercially operate the reactor.

(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 net power plant efficiency calculations. When water-steam is selected as process heat working fluid, steam can be generated through an intermediate heat exchanger, wherein the process heat working fluid is thermally coupled without mixing with 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) Pressuriser

The pressurizer is a heat exchanger generally not applicable to HTGR design typologies. However, should process heat applications require stabilized high-pressure superheated steam, a pressurizer with functions similar to those adopted for Pressurized Water Reactors (PWRs) can be installed on the secondary side of the HOLOS-Mono-10 intermediate heat exchanger.

(g) Primary pumps

The SPM fully comprised by the Reactor Pressure Vessel of the HOLOS-Mono-10 design, houses 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 as the HOLOS-Mono-10 is equipped with redundant active magnetic bearings (AMBs). Pumps can be considered as part of auxiliaries hydraulically coupled to the Ultimate Heat Sink (UHS) for active cooling of the Brayton intercooler and heat rejection heat exchanger. The water-cooled configuration of the design offers the highest plant net thermal-to-electric efficiency of 45%. For air-cooled configuration the efficiency drops to approximately 42% as plant hotel-loads represented by electric air-cooling fans increases to actively circulate environmental air through radiators.

(h) Distinguishing features

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The design is equipped with the traditional HTGR safety features with functional containment inherently achieved by qualified TRISO fuel particles. In addition to the functional containment characteristic of qualified Accident Tolerant Fuel (ATF), HOLOS-Mono-10 further enhances safety through additional engineered redundant containments capable of storing and processing potentially contaminated working fluid. Decay heat is removed passively with multiple natural heat transfer mechanisms with the maximum fuel temperature excursions remaining within safety margins under combined severe Loss Of Coolant Accident and Station Black Out scenarios.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) 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 multiple heat dissipation pathways to the Ultimate Heat Sink (UHS).

(e) Safety Approach and Configuration to Manage DEC

The design underwent Design Basis Accident (DBA) and Beyond Design Basis Accident (BDBA) simulations validated by physical data obtained through testing of the Sealed Power Module (SPM) hardware. Testing was conducted through scaled down hardware and a reactor simulator equipped with a graphite core. This core was non-nuclear as it was equipped with electrical heaters, replacing TRISO fuel compacts, through graphite-fuel simulated elements operated with helium at relevant full-scale design operating conditions. The results validated the inherent safety performance of the design under postulated DEC as documented by numerous peer-revieed techncial reports and publications developed by the Argonne National Laboratory and academic institutions.

(f) Containment System

Although qualified TRISO fuel does not require the traditional large volume reinforced concrete containment mandatory for light water reactors, the reactor pressure vessel forming the SPM is surrounded by the secondary pressure vessel operated

at a low vacuum to capture potential helium leaks from the reactor pressure vessel. As a certain threshold of helium is collected within the free volume of the second pressure vessel a small compressor is activated to compress back the leaked helium inventory into the reactor pressure vessel. A third pressure boundary is factored in the containment design of the HOLOS-Monolithic-10 depending on the application and final installation. For example, the thrid barrier may be represented by the volume of the shipping container comprising the secondary pressure vessel, or the volume of the reactor cavity for subterranean installations. These additional sealed and controlled volumes are designed to capture potential working fluid leaking from the secondary pressure vessel (e.g., accident scenarios resulting in piercing of both secondary and reactor pressure vessel). When the third pressure boundary is represented by the shipping container, or "reactor compartment" for power generation onboard of ships or off-shore installations, this additional defence-in-depth barrier ensures that potentially leaked helium and potentially activated Argon gas contained in the air transiting in the proximity of the core remains captured until the activated Argon naturally decays (a few hours later), while the potentially contaminated helium is treated prior to opening the reactor compartment or shipping container (e.g., for refueling, maintenace operation or for deactivation and decommissioning activities).

(g) Spent Fuel Cooling Safety Approach / System

Utilizing as reference HOLOS-Mono-10 units each with cores lasting 8 years at EFPY conditions, the whole unit is left within its compartment or cavity for a cooling period prior to lifting the unit altogether with its external radial shields in preparation for transport to a specialized refueling, refurbishing or storing facility. Alternatively, these units can be opened on-site to conduct onsite refueling, refurbishing and maintenance activities. 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 reactor pressure vessel and inserted within transport and storage licensed spent fuel casks for onsite or offsite long term storage.

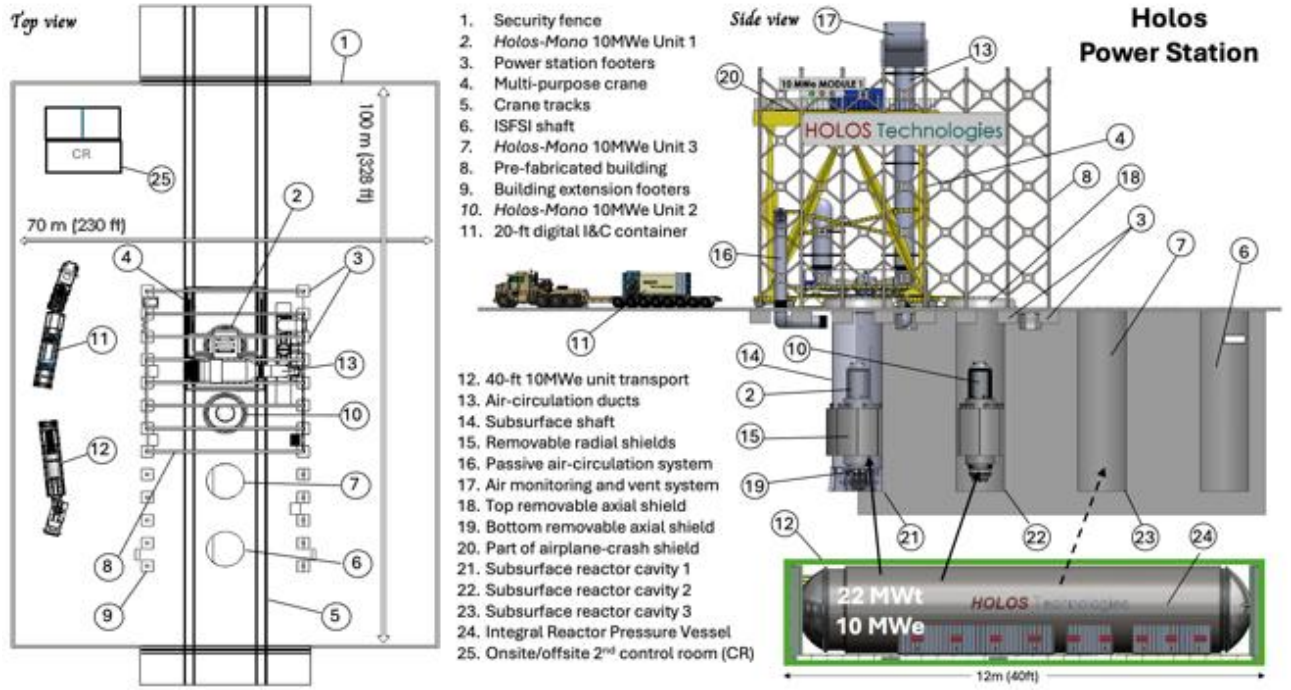
6. Plant Safety and Operational Performances

The underlying defense-in-depth offered by the HOLOS-Mono-10 design is based on its qualified, ATF TRISO. This type of fuel, tested for decades in multiple countries, enables high-temperature operations leading to a high thermal-to-electric conversion efficiency. The SPM is further sealed within the secondary pressure vessel and a sealed intermodal container (for power rating up to 10MWe), or a sealed reactor cavity. These configurations offer multiple pressure boundaries engineered to intercept and trap potentially contaminated working fluid (e.g. originating from defective or damaged TRISO fuel particles), for treatment and neutralization, thus enhancing the already high safety performance associated with HTGR technologies. The unique integral power conversion system characterizing the HOLOS-Mono-10 design further enables a subtsantial reduction in the inventory of helium – there is no inventory of helium normally filling the balance of plant. The integral power conversion system featuring a full Brayton gas thermodynamic cycle with decoupled compressor-expander turbomachinery enables to maintain constant core temperature while conducting aggressive power maneuvering . This unique characteristic of the HOLOS designs enables high operational performance, higher fuel utilization, and proportionally lowered spent fuel production per unit energy generated. These features further enhance safety performance as less frequent refueling and lowered volumes of spent fuel per MW basis, directly reduce risks and costs associated with fuel handling, transport and storage. The increased operational performance efficiency, the elimination of the traditional Balance of Plant, the high-rate load follow capability while maintaining high thermal-toelectric efficiency overall drive lowered capital, operation & maintenance, deactivation & decomissioning costs, supporting higher economic sustainability and competitiveness with non-nuclear energy sources.

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. HOLOS-Mono-10 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). The equipment supporting I&C altogether with the power management modules regulating the Electric Storage System (ESS), occupies the volume equivalent to a dedicated "digital I&C container" electrically coupled to the intermodal container housing the integrated reactor via shielded electro-magnetic interference and electro-magnetic pulse shunted data and power cables.

8. Plant Layout Arrangement



The HOLOS-Mono-10 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 resulting HOLOS Power Station can be equipped with multiple HOLOS-Mono-10 units, with coupled load-follow power output. The power station can be expanded to increase the total station power rating, or it can be equipped with HOLOS-Monolithic-80, 320, 1280MWe while conserving the highly integrated architecture resulting in the smallest footprint per MWe generated.

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 modeling 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. Licensing documentation supporting the Regulatory Engagement Plan (REP) and the submission of a combined construction and operation license (COL) will be completed in 2026.

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-10 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-10 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

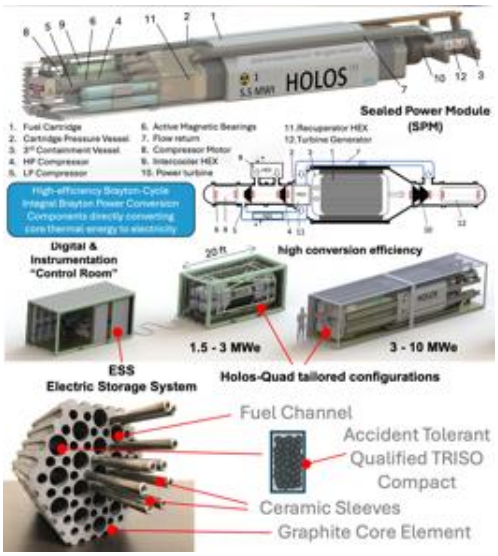
Year	Milestone	Status
2015 – 2017	Conceptual design developed by Filippone & Associates LLC (F&A)	**Completed**
2017 – 2018	HolosGen LLC developed and successfully tested the HOLOS-Test-Assembly-1 (HTA-1) subscale prototype equipped with a surrogate core.	**Completed**
2018 – 2019	HolosGen awarded by the U.S. DOE ARPA-e MEITNER funding program to demonstrate safety and techno-economic performance.	**Completed**
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.	**Completed**
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.	**Completed**
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.	**Completed**
2024 – 2027	Complete digital twin and electrical full-scale twin of the HOLOS-Mono-10 configuration.	**Completed**
2026 – 2028	Complete Regulatory Engagement Plan and COL documentation for submission to the NRC	**Planned**
2027 – 2028	FOAK nuclear testing at U.S. DOE or alternate facility.	**Planned**
2029	Conversion of the FOAK demonstrator unit into first commercial unit.	**Planned**

HolosGen, United States of America
HOLOS-QUAD



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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, HEX: Heat Exchanger, I-HEX: Intermediary-Heat Exchanger, AMB: Active Magnetic Bearings.

MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	HolosGen LLC, United States of America
Reactor type	HTGR
Coolant/moderator	Helium / Graphite
Design status	Manufacturing – codes validated via prototype with power conversion components and DBA safety performance tested. No criticality test.
Thermal/electrical capacity, MW(t)/MW(e)	22 /10
NSSS operating pressure (primary/secondary), MPa	Full Brayton power cycle: Max 7, Min 3.5. Helium loop comprises power conversion system – no traditional Balance of Plant.
Core inlet/outlet coolant temperature (°C)	590 / 855
Primary circulation	Helium
Fuel type/assembly array	Qualified TRISO-UCO Compacts
Number of fuel assemblies in the core	1170kg UCO to supply 22MWt at 8 EFPY
Fuel enrichment (%)	19.95
Core discharge burnup (GWd/ton)	61
Refuelling cycle (months)	96 at EFPY operating conditions
Fuel cycle requirements/approach	Once-through
Reactivity control mechanism	Redundant control drums and shutdown rods
Approach to safety systems	Passive decay heat removal via natural cooling mechanisms
Seismic design (SSE)	4g (rail transport shocks limit)
Design life (years)	40 to 60
Plant footprint (m^2)	82 (reactor comprised in ISO container + external radiation shields)
RV height/diameter (m)	9.5/2.3 (4 coupled SPMs)
RV weight (metric ton)	40 (inclusive of Power Conversion System)

1. Introduction

The HOLOS-Quad microreactor is a transportable High-Temperature Gas Reactor (HTGR), with high-efficiency load-follow power conversion whose Systems, Structures and Components (SSCs) are fully integrated within Sealed Power Modules (SPMs). When multiple SPMs are mechanically coupled to one another, they form a "neutron-coupled" core. The design eliminates the traditional Balance of Plant (BoP), normally formed by networks of tubing and electrical conduits coupling the thermal source represented by the core to the SSCs forming a Rankine or Brayton power cycle for most of Generation III+ and IV designs. In the HOLOS-Quad, the traditional BoP is replaced by the integration of tested and independently validated SSCs forming the SPM. This innovative plant layout approach greatly simplifies the design with enhanced safety and net plant efficiency, while substantially reducing the plant footprint per MWe basis. With the elimination of the BoP and comprehensive integration of the power conversion components, the design substantially reduces capital, operational, maintenance, and long-term spent fuel storage costs. The design Power Conversion System (PCS) features a closed-loop full Brayton helium gas thermodynamic cycle with recuperator and intercooler heat exchanger operating at a net plant efficiency of 42% when the Ultimate Heat Sink (UHS) is represented by environmental air at 30°C, or 45% for the configuration with the UHS represented by water. The net power plant conversion efficiency factors an environmental UHS temperature of 30°C. The design features a load-follow rate of 10% of the reactor nominal power per minute, while maintaining the net plant efficiency >40% for power maneuvering >30% of nominal power. For power rating up to 10MWe the whole power plant is comprised within a single 12-meter intermodal container. The fuel selected for this configuration is TRISO, a qualified, ready for commercialization, Accident Tolerant Fuel (ATF). The core of the 10MWe design configuration is graphite moderated with burnable neutron absorbers, optimized to provide 22MWt for approximately 8 years at Full Effective Power Years (FEPY) conditions, with an anticipated power plant life span of 40 years, extendable to 60 years. Following an extended period of operation, each of the SPMs can be individually transported within a shielded intermodal container, enabling transport of an individual irradiated SPM only 72 hours from last shutdown. Irradiated "Hot" SPMs can be removed and transported away from the first deployment site to a different site and reassembled to continue operations until the reactivity of the coupled core becomes insufficient to sustain chain reactions. When the SPMs become "spent", intermodal containers internally retrofitted with shields can be utilized to transport the SPMs offsite to Independent Spent Fuel Storage Installations (ISFSIs) for the extraction and storage of core components, while refurbishing the SSCs forming the PCS to restart operations with a fresh coupled core. The HOLOS-Quad is scalable to closely match application and customer-specific requirements and can be configured to operate with the SPMs positioned vertically or horizontally, above or below ground or water. The HOLOS-Quad can be configured with fixed SPMs, wherein four independent SPMs, each equipped with its integrated power conversion components, are mechanically coupled to form a fixed pseudo cylindrical core geometry. Reactivity control is achieved through redundant, resilient, diversified and independent control mechanisms, based on traditional redundant control drum regulation, redundant shutdown rod insertion and burnable neutron absorbers.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

In addition to electricity generation, the HOLOS-Quad design features high-temperature and low-temperature process heat capabilities. High-temperature is supplied through intermediary heat exchangers that support, for example, non-carbon fuel production (e.g., hydrogen), steelmaking and other high-temperature industrial applications. The low-temperature process heat is a retrofitted waste heat recovery capability utilized to drive chillers for cooling of data centers or urban heating without electric consumption. The design load-follow capability closely matches electricity and process heat demand. The design scalability and flexibility support a large portfolio of applications including distributable energy source, for example, supplying power to microgrids or bulk power grids, stand-alone electric island, charge electric vehicles at locations without bulk power grid and/or supplement variable on-demand power, while compensating for the intermittency of renewable energy sources. The design can be configured to supply electricity and high-temperature process heat to industrial and offshore or ship power applications; supply dedicated (onsite) power with non-invasive footprint for industrial facilities and/or data centers. The integration of the PCS enables compactness for use as an emergency power generator, rapidly deployable to disaster areas to, for example, desalinate water and provide electricity to disaster coordination agencies. The design is suitable as mobile emergency electric generator to supply emergency power to large commercial reactors undergoing low probability high impact beyond design basis accident events such as those manifested at the Fukushima Daiichi nuclear power station. Scaled-down versions of the design can provide from kW-to-MW power rating

for special applications, including power on demand for space propulsion, deep-space, lunar surface and Mars missions. These configurations can also provide electric power and propulsion to submergible drones. More traditional applications include electric island capabilities to supply carbon-free power to remote communities and terrestrial or offshore mining operations currently relying on fossil fuel burning. Overall, the HOLOS-Quad design flexibility and compactness enable up to 10MWe power generation transportable via traditional vehicles, satisfying the requirements of multiple applications and markets. For power rating >10MWe to SMR power rating (e.g., 80MWe-320MWe), the reactor pressure vessel diameter induces need for special transport, as required for all designs with comparable power rating. However, the HOLOS designs enhanced safety philosophy with the elimination of the BoP are maintained for all scaled configurations, thus delivering the smallest footprint per MWe basis.

3. Design Philosophy

The HOLOS design philosophy integrates “safety at all costs”, “respect for the public and the environment” and reliance on “mature-validated technologies only” to substantially “reduce technical and economic risks”.

Safety at all costs for all the HOLOS designs translates into permeating the SSCs with inherent safety measures at the individual component level, regardless of stressors often induced by economic and/or political decisions. Safety is prioritized above all other considerations with enhancement of the defense-in-depth passive safety philosophy. For example, although the fuel adopted is “self-contained” TRISO, the design features two pressure vessels: The reactor pressure vessel, comprising and sealing the core and the power conversion components, surrounded and sealed by a secondary pressure vessel. The secondary pressure vessel has safety and operational functions: i) It captures working fluid as an additional containment system should series of design basis accident events result in working fluid contamination and leakage from the primary vessel, and ii) Once a certain threshold of working fluid is collected within the free volumes of the secondary pressure vessels, a compressor pumps it back into the primary working fluid inventory control system for reutilization or cleanup, thus performing an operational function. These additional engineered passive barriers enhance the defense-in-depth principle as self-contained fuel provides the first barrier to radionuclide migration into the working fluid (e.g., from potentially defective or damaged fuel elements). For traditional HTGRs, the inventory of working fluid occupying the reactor pressure vessel internals and the traditional Balance of Plant (BoP) is contained by the pressure boundary formed by this equipment, which represents the second barrier preventing radionuclides migration. This safety feature is considered adequate. However, the HOLOS-Quad design provides a third barrier to potential radionuclides leaks as the secondary pressure vessel intercepts potentially contaminated working fluid within the secondary pressure vessel free volume and pumps it back to the reactor pressure vessel or to an inventory cleanup system with functions similar to those of the Chemical Volume Control System (CVCS) in water reactors. For subterranean power station configurations, a fourth barrier preventing migration of potentially contaminated working fluid is further represented by the reactor cavity which houses and seals the secondary and primary pressure vessels during operations. The reactor cavity is saturated with helium at slightly above atmospheric pressure to prevent environmental air ingression into the reactor cavity during operations. This approach prevents air irradiation in the proximity of the pressure vessels components housing the core. Should air ingress the reactor cavity seals, a natural circulation chimney enables ejection of potentially activated air through a stack for natural decay and neutralization of the air prior to flowing to ground level. Although the design is passively able to dissipate decay heat through natural heat transfer mechanisms as demonstrated through prototype testing, the design is further equipped with redundant enhanced passive heat transfer features ensuring decay heat is removed even under partial damage of the heat cavity the primary and secondary pressure vessels resulting from hostile actions (e.g., shape charge, kinetic events, and pressure boundary piercing scenarios). The design provides enhanced protection, and mitigation features exceeding safety margins and regulatory guidance in response to world events demonstrating intentional hostile targeting of nuclear power plants. These events represent “beyond design basis accident” scenarios to be fully integrated with regulatory processes. The design is further protected from sudden loss of electrical load, does not depend on offsite or onsite electric power to safely startup and shut down and is equipped with redundant hardwired monitoring and control systems, bypassing satellite and wireless monitoring and communications.

HOLOS design philosophy addressing respect for the public and the environment factors the enhanced barriers to radionuclides leakage into the environment through passive added pressure barriers and utilizes “cradle-to-grave” Life Cycle Analysis (LCA) for the SSCs forming the HOLOS power station to decarbonize manufacturing, installation, Deactivation and Decommissioning (D&D) activities at the individual SSC level. Through the elimination of the traditional BoP the design reduces the total amount of steel alloys employed. This design characteristic simplifies D&D activities as the total volume of low-level radiation materials to be processed and transported for decontamination to specialized facilities is reduced. Additional environmentally beneficial aspects include decreasing the spent fuel volume and thermal pollution per MWe generated. This feature is achieved by increasing the thermal-to-electric power conversion system efficiency and equipping the design with features enabling maintaining high operational efficiency during power maneuvering.

The philosophy addressing the reduction of technical and economic risks is implemented by the HOLOS-Quad design by utilizing only SSCs with proven technologies already commercial (e.g., active magnetic bearings, high-speed electrical

motors/generators, electronic power modules, aero-derivative turbomachinery and so on), or technologies with a third-party validated maturity measured under the Technology Readiness Level (TRL) scale with TRL5 or higher.

The philosophy adopted to reduce the economic risks leverages decades of cost avoidance opportunities identification obtained through the development of economic models to analyze Generation III, III+ and IV reactor designs and their SSCs. These analyses indicate that for reactors to be competitive with non-nuclear power producing technologies their thermal-to-electric power conversion efficiency must be greater than 40% and remain at comparable levels during power maneuvering. As the conversion efficiency increases the fuel utilization factor also increases, and the reactor economics becomes sustainable and competitive. Overall, high net plant efficiency further induces spent fuel volume reductions per MW generated, as well as reduction in costs associated with spent fuel handling, transport and storage. Manufacturing, transport, capital, operation, maintenance and D&D costs are reduced through the elimination of the BoP and the integration of the PCS. Overall, all HOLOS designs adopt a multi pillar philosophy to simultaneously satisfy very conservative requirements associated with safety, respect for the public and the environment while reducing technical and economic risks to ensure competitiveness and long-term sustainability.

4. Main Design Features

(a) Nuclear Steam Supply System

Power Conversion 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 overall plant footprint, while enabling high-efficiency electricity supply matching highly variable power profiles. Thermal-to-thermal conversion with energy transfer from helium to a client/ application-selected working fluid, the design is equipped with an intermediate heat exchanger to support process heat for industrial applications at variable temperatures.

(b) Reactor Core

In the HOLOS-Quad configuration, each SPM contains ¼ of the core. Once the SPMs are mechanically coupled, they form a pseudo cylindrical core geometry. Each SPM is equipped with a fuel cartridge loaded with accident tolerant qualified TRISO fuel and burnable neutron absorbers embedded within hexagonal graphite elements, ensuring optimized burnout for approximately 8 Effective Full Power Years (EFPY). The core design is safety compliant under postulated accident scenarios including sudden core water flooding with a fresh core (maximum reactivity). The core is designed to enable safe transport when the SPMs are re-deployed and re-assembled to supply power to different sites or when their coupled reactivity is no longer sufficient to sustain chain reactions. The volume and geometry of each SPM enables “hot transport” by surrounding it with adequate shielding, without violating the volumetric constraints represented by shipping containers. Analyses of the HOLOS-Quad SPM shielding and dose rate calculations for the SSCs forming the integral Brayton power conversion system were conducted and published by the U.S. Argonne National Laboratory.

(c) Reactivity Control

The reactivity control for the HOLOS-Quad design is achieved through independent, diversified and redundant reactivity control mechanisms based on redundant banks of control drums and shutdown rods with assured negative reactivity insertion even under failure of control drum and sbutdown rod banks.

(d) Reactor Vessel (RV) and Internals

The HOLOS-Quad design replaces the conventional Reactor Pressure Vessel housing the core and related components with an expanded SPM made with relatively small pressure vessel components, easier to manufacture (compared to forging large components), welded together and containing not only the core but also the full range of power conversion components, thus eliminating the Balance of Plant. These components are dimensionally optimized to comprise the Brayton compressor-motor assembly at one end of the SPM, the core components in the central region, and the turbine-generator assembly at the opposite end. The separate pressure boundary within the core portion of the SPM is represented by Silicon Carbide (SiC) sleeves through which the helium coolant circulates at high-pressure, while the pressure outside of the sleeves is at or below atmospheric. Each SiC sleeve is coupled to flexible joints to eliminate expansion/contraction stresses. The resulting equivalent pressure vessel design minimizes the inventory of helium for each SPM and the reactor system as a whole. The SiC sleeves are under optimization and the reason for the readiness level of this design to be at TRL5. Overall, the design features four identical SPMs, each housing and sealing dedicated power conversion components and fuel elements with less challenging manufacturing requirements when compared to traditional reactor pressure vessels, balance of plant and power conversion components.

(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 efficiency calculations. 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) Pressuriser

The pressurizer is generally not applicable to HTGR design typologies. However, certain process heat applications might require superheated vapors (e.g., steam) at stable pressure. For such applications, a pressurizer can be integrated with the intermediate heat exchanger supporting process heat at variable temperatures.

(g) 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). As for most HTGR configurations, there are no primary pumps as those equipping, for example, pressurized water reactors. However, for the high-efficiency HOLOS-Quad configurations coupled to water as the UHS, realtively small power rating pumps are employed to circulate water to transfer thermal energy with the Brayton intercooler and UHS heat exchangers.

(h) Distinguishing features

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The HOLOS-Quad design is equipped with traditional HTGR safety features with additional, redundant, pressure boundaries further preventing migrations of radionuclides from potentially defective or damaged TRISO fuel elements. As the HOLOS-Quad utilizes sleeves in each SPM, helium is only thermally coupled to the core internals: No physical contact between the coolant and graphite, TRISO fuel elements, burnable neutron absorber pins and the reactivity control components. This design feature eliminates the need for a helium clean up system. More generally, the design features a low power density core and relies on passive cooling under i) environmental extremes, ii) loss of coolant, and iii) loss of electricity scenarios. The fuel temperature remains well below safety margins under normal and off-normal operating conditions and postulated design basis accident scenarios. The safety and technical performance analyses on reactivity control under transients and load-following operations were conducted by the Argonne National Laboratory and validated by comparison with test data obtained by operating the HOLOS Sealed Power Module-Subscale Simulator (SPM-SS) equipped with an electrically heated surrogate core. The performance results of the inherent safety of the HOLOS-Quad thermal-hydraulic design are documented in multiple technical reports as for example: “Preliminary Thermal-Hydraulics and Selected Safety Analysis for HOLOS-Quad Reactor Design,” ANL/NSE-22/1, published by the Argonne National Laboratory in 2022.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) 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.

(e) 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.

(f) Containment System

The HOLOS-Quad design utilizes accident tolerant TRISO fuel, wherein each microsphere forming the fuel element features a "functional containment" that traps radionuclides, thus eliminating the need for sealed containment as represented by traditional containment structures. However, to further enhance safety, in addition to the functional containment, all HOLOS design configurations feature additional engineered pressure boundaries, which remain sealed during power and maintenance operations. These features prevent radionuclides potentially leaking from damaged or defective TRISO microspheres into the environment. These engineered pressure boundaries are also designed to trap the air surrounding the SPMs and passively circulating between the radial shields in the regions near the core during operation. Depending on the HOLOS-Quad design configuration selected (e.g., underground versus above surface), additional pressure barriers further seal the SPMs within the shipping container and the reactor compartment housing the shipping container.

(g) Spent Fuel Cooling Safety Approach / System

Decay heat removal always occurs passively. The HOLOS-Quad design enables "hot" removal of each individual SPM by mechanically separating and transferring the SPM into shipping containers equipped with shielded casks. As each SPM only occupies approximately ¼ of the standard container internal volume, the remaining volume of a shielded container may be filled with removable radiation shielding materials that provide adequate radiation stopping power to enable safe transport. If the SPMs are removed from their Quad "neutron coupled" configuration prior to reaching "spent" conditions (e.g., at the end of the EFPY), each SPM can be repositioned and reassembled into the engineered container configured to couple the 4x SPMs once safely transported to a new deployment location. The reassembled unit can therefore resume load-follow power generation until the SPMs become spent (e.g., unable to sustain fission chain reactions). Overall, the irradiated individual SPMs can be removed and reassembled, or permanently stored, following a minimum of 72 hours from their last shutdown. The ability to rapidly remove irradiated SPMs from a given location to a new location for refueling, refurbishing, relocation, or permanent storage is one of the innovative and unique features of the HOLOS-Quad design.

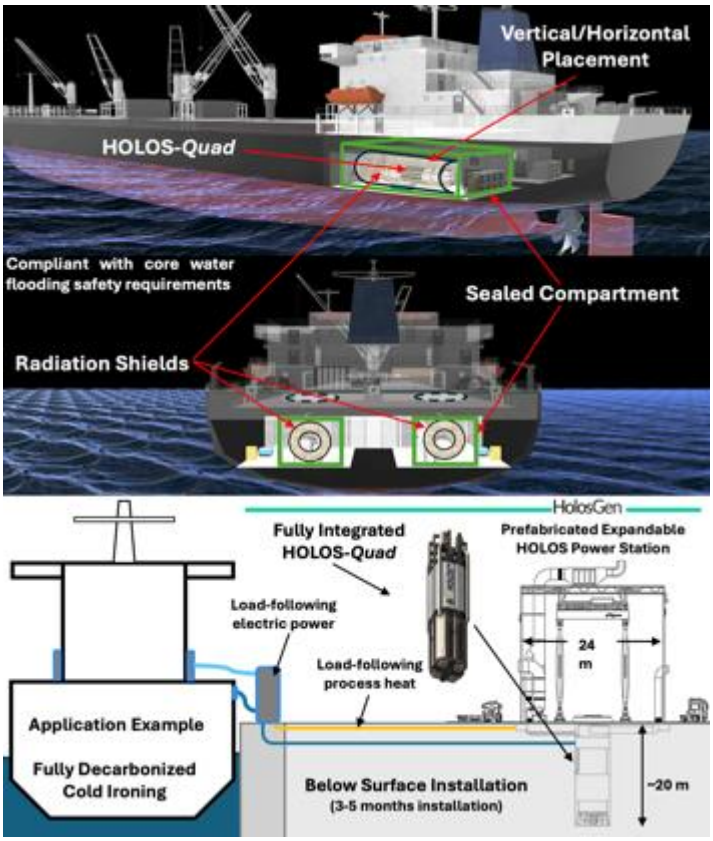
6. Plant Safety and Operational Performances

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 neutron absorbers within the graphite-TRISO matrix to ensure that core reactivity remains below safety margins even under water flooding events when the core is at its maximum reactivity. 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

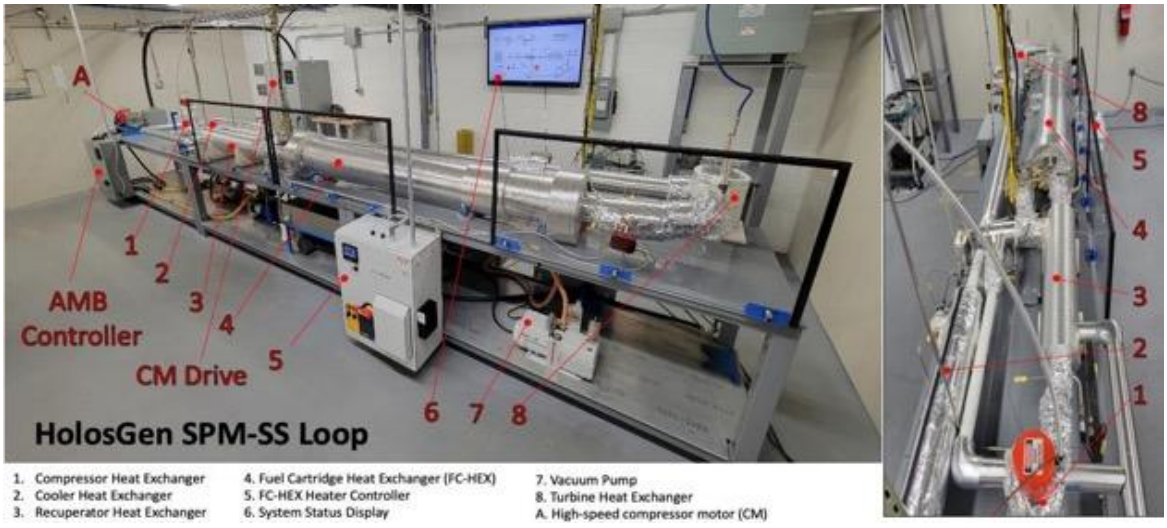
All HOLOS design configurations are equipped with integral power conversion components designed to follow the electrical load automatically and at high rate. The control system regulates the power flows between the thermal-to-electric conversion components, the black-start & ballast battery or flywheels (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.

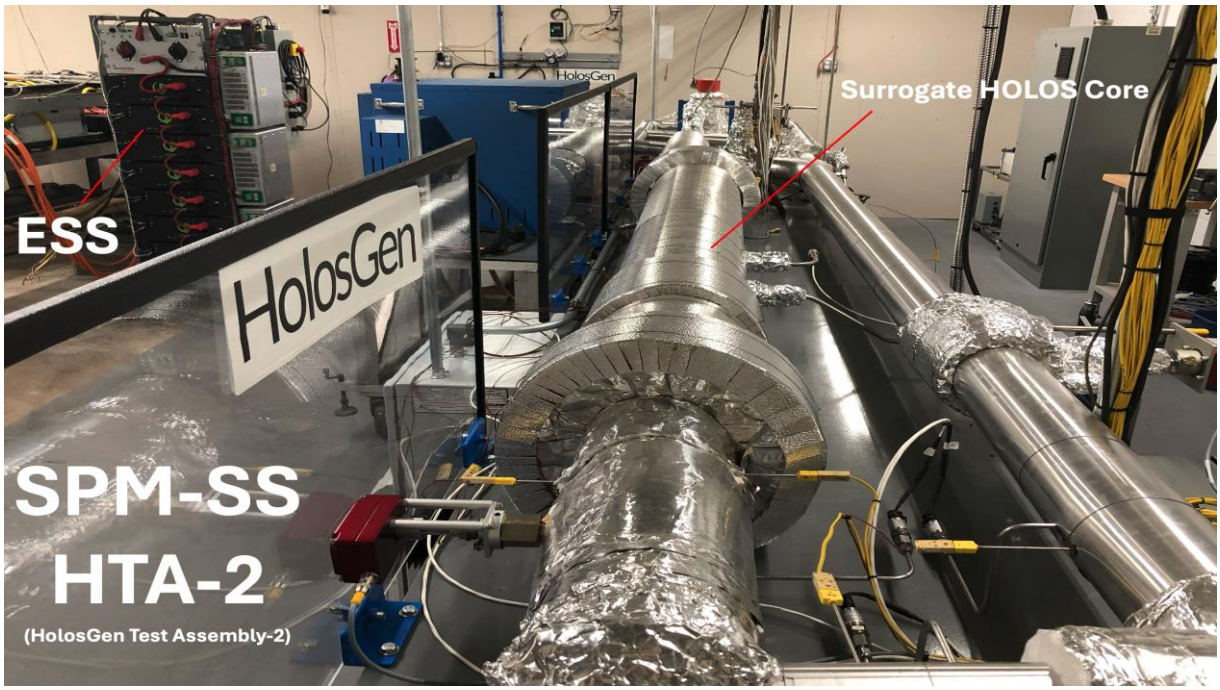
8. Plant Layout Arrangement



The HOLOS-Quad features a highly integrated, transportable nuclear power plant fully comprised within the dimensional constraints of standard shipping containers. For terrestrial above surface applications, the “HOLOS Container” may be positioned horizontally within a prefabricated removable concrete “shield building” with engineered airflow pathways to support passive decay heat removal and decay of potentially irradiated air prior to venting it to atmosphere. For subsurface terrestrial installations, one or multiple Containers may be “verticalized”. (The Active Magnetic Bearings (AMBs) enable horizontal or vertical positioning of the SPMs). In this configuration, the Container may be positioned within a low-cost borehole, equipped with a prefabricated liner with features to align removable radial and axial radiation shields. 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. The prefabricated building design expedites installation and disassembly activities, with installation to be conducted over a period of 3-5 months. This construction approach reduces both capital and decommissioning cost at the end of the licensed life of the resulting “HOLOS Power Station”. The prefabricated building is designed to facilitate expansion to gradually house multiple units with reduced upfront cost and may house an Independent Spent Fuel Storage Installation (ISFSI). As the design eliminates the Balance of Plant and increases thermal-to-electric conversion efficiency, the power plant footprint is substantially reduced per unit of electric energy generated, thus enabling compact plant layouts ideal for applications on commercial ships and as a decarbonized power generator installed through prefabricated removable facilities supporting, for example, “cold ironing” for docked vessels.

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. Licensing documentation supporting the Regulatory Engagement Plan (REP) and the submission of a combined construction and operation license (COL) will be completed in 2026.

11. Fuel Cycle Approach

The 10MWe HOLOS-Quad design is currently equipped with a core optimized to supply approximately 22MWt for approximately 8 years at Equivalent Full Power Years (EFPY) conditions. 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.

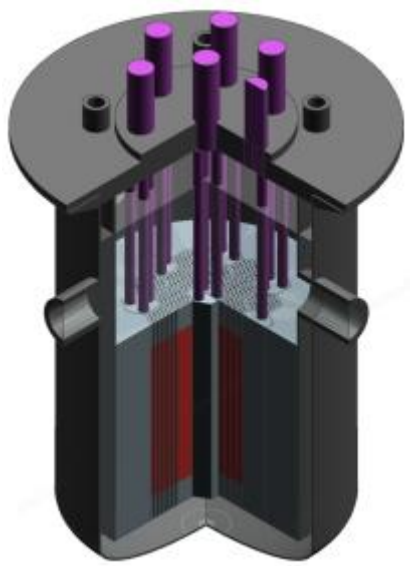
13. Development Milestones

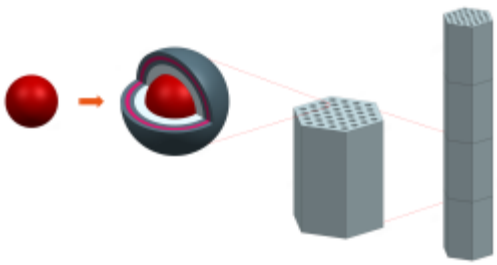
Year	Milestone	Status
2015 – 2017	Conceptual design developed by Filippone & Associates LLC (F&A)	**Completed**
2017 – 2018	HolosGen LLC developed and successfully tested the Holos-Test-Assembly-1 (HTA-1) subscale prototype equipped with a surrogate core.	**Completed**
2018 – 2019	HolosGen awarded by the U.S. DOE ARPA-e MEITNER funding program to demonstrate safety and techno-economic performance.	**Completed**
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. Completed Holos Test Assembly 2 (HTA2) for safety performance quantification.	**Completed**
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.	**Completed**
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.	**Completed**
2024 – 2027	Complete digital twin and electrical full-scale twin of the HOLOS-Quad configuration.	**Completed**
2026 – 2028	Complete Regulatory Engagement Plan and COL documentation for submission to the NRC	**Planned**
2027 – 2028	FOAK nuclear testing at U.S. DOE or alternate facility.	**Planned**
2029	Conversion of the FOAK demonstrator unit into first commercial unit.	**Planned**

Jereh SMR Energy Holding Ltd, U.A.E., JMR



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Jereh SMR Energy Holding Ltd., U.A.E.
Reactor type	MSR
Coolant/moderator	Flibe / graphite
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	10 - 20 / 5 - 10
NSSS operating pressure (primary/secondary), MPa	0.1 / 0.15
Core inlet/outlet coolant temperature (°C)	650 / 700
Primary circulation	Forced
Fuel type/assembly array	TRISO in prismatic blocks
Number of fuel assemblies in the core	12
Fuel enrichment (%)	19.75%
Core discharge burnup (GWd/ton)	< 90
Refuelling cycle (months)	≥ 96
Fuel cycle requirements/approach	Once-through and take-back
Reactivity control mechanism	Control rods (B4C)
Approach to safety systems	Fully passive
Seismic design (SSE)	0.3 g
Design life (years)	60
Plant footprint (m^2)	< 5000
RV height/diameter (m)	6.1 / 3.3
RV weight (metric ton)	54

1. Introduction

JMR is a fluoride-salt-cooled high-temperature microreactor that preserves the same TRISO-in-prismatic-block fuel form and manufacturing pedigree as JFHR, including direct use of the identical fuel production line. To maximize transportability, the reactor module has been volumetrically compacted, prioritizing a footprint compatible with standard heavy-haul road, rail, or maritime shipping.

2. Target Application

☒ Electricity
- ☒ Baseload
- ☐ Load-following

☒ District Heating / Steam

☒ Hydrogen
- ☐HTSE - electrolysis
- ☒PEM - polymer membrane

☒ Desalination

☒ Industrial process heat
- ☐ <200°C
- ☐ 200–550°C
- ☒ >550°C

3. Design Philosophy

As a 10 - 20 MWt microreactor JMR, adopts the same fuel and coolant technology as JFHR, but with its reactor considered as an independent skid module coupled to one or more independent skid modules. This is to achieve more stringent transportability and siting flexibility requirements. JMR also leverages the low operation pressure of molten Flibe coolant without generating steam or hydrogen, the inherent safety features of TRISO fuel in prismatic form and the Passive Residual Heat Removal (PRHR) system.

4. Main Design Features

(a) Nuclear Steam Supply System

For power conversion, a supercritical carbon dioxide (sCO2) Brayton cycle remains thermodynamically and volumetrically favorable. However, for remote off-grid applications where simplicity, modularity, and rapid deployment are paramount, a steam Rankine cycle may be adopted to offer superior system-level reliability and reduced maintenance burdens. The JMR therefore adopts a dual-readiness power conversion strategy: a compact sCO2 Brayton loop for high-efficiency, high-availability installations, and a steam cycle configuration for contingency and low-logistics environments where system mass and field reparability dominate over thermodynamic optimization.ments, and supports autonomous operation while reinforcing the reactor's Safe-by-Design philosophy.

(b) Reactor Core

The reactor core contains 12 hexagonal fuel assemblies, seven reactivity control assemblies and replaceable graphite reflectors. The effective core diameter and height are 1.85 m and 2.5 m respectively.

(c) Reactivity Control

The reactivity control system consists of seven control assemblies with each assembly containing three control rods. The reactivity control during normal operation and shutdown are achieved by active regulation rods and passive shutdown rods respectively.

(d) Reactor Vessel (RV) and Internals

The reactor vessel is designed at near-atmospheric pressure level. The key advantages include: (1) significantly reduced vessel wall thickness and weight, lowering manufacturing cost; (2) reduction of stress corrosion cracking concerns; (3) inherent leakage resistance - small leaks do not escalate; (4) simplified seismic design due to lower structural loads. The internals utilize graphite block stacking, allowing easy on-site assembly and replacement.

(e) Reactor Coolant System and Steam Generator

The reactor coolant system has an intermediate-loop configuration, with the primary and intermediate loops utilizing Flibe and FliNaK as the coolant medium respectively. The primary heat exchangers (PHXs) are in a separate independent module and will be connected to the reactor module. During normal operation, two of the three PHXs are sufficient to transport the heat from the core to the intermediate loops.

(f) Pressuriser

(g) Primary pumps

JMR has three main pumps in the primary circuit. The technical specifications of pumps on irradiation shielding and anti-corrosion is less stringent compared to the liquid-fueled MSR.

(h) Distinguishing features

5. Safety Features

(a) Engineered Safety System Approach and Configuration

JMR is equipped with three independent PRHR loops. Any of the three loops is capable of removing 0.75% of nominal fission power after reactor scram. Heat is then transferred to three independent FliNaK intermediate loops, and ultimately transferred to the ambient air. During any postulated accident scenario, no pumps or external power are needed to ensure core cooling, instead, all heat transport relies on gravity, natural circulation and thermal radiation.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

JMR has identified a complete list of Postulated Initiating Events (PIEs), which will go through Probabilistic Risk Assessment (PRA) where they are categorized by frequency, and adopts defense-in-depth configuration, inherent safety as well as passive safety for all DBCs. The safety configuration is built upon the inherent resilience of TRISO fuel and the additional retention capabilities of Flibe coolant, providing multiple layers of defense without relying on high-pressure containment or active, electrically powered systems.

(e) Safety Approach and Configuration to Manage DEC

JMR manages DEC without active systems. For DEC-A, TRISO fuel tolerates temperatures below 1600℃ even with stuck rods, or station blackout combined with complete RHRS loss. For DEC-B (extreme external events), the chemically inert Flibe (no hydrogen, no exothermic reactions) maintains containment integrity. Specific DEC events (e.g. salt freezing, graphite dust blockage) are prevented by decay heat (maintaining Flibe's temperature above its melting point - 459℃) and the coolant purification system.

(f) Containment System

JMR's containment system is fundamentally different from traditional nuclear plants. Instead of a massive, high-pressure concrete and steel dome, it adopts the functional containment concept. Reactor vessel, molten salt, prismatic graphite matrix and TRISO particles serve as multiple barriers in JMR defensive-in-depth design. The robustness of TRISO makes it a perfect candidate for fission product retention, ensuring the health and safety of both workers and the public. Additionally, a coolant chemical control system serves as a monitor of contamination in molten salt coolant, in order to protect the integrity of fuel assemblies.

(g) Spent Fuel Cooling Safety Approach / System

JMR is equipped with a spent fuel storage system functionally designed for cooling and storage. This system comprises a cooling pool and a storage vault. After being discharged from the core, the spent fuel is loaded into sealed shielded storage canisters and immersed in the cooling pool temporarily until their decay heat decreases to safe levels. The canisters are then transferred to the storage vault, which features radiation shielding and forced air cooling.

6. Plant Safety and Operational Performances

JMR leverages the inherent safety features of TRISO-coated particle fuel and molten salt coolant to attain robust plant safety and operational performance, including operating at near atmospheric pressure and high temperature. This eliminates the need for a large, high pressure containment structure and therefore avoids the risk of high pressure vessel rupture. A passive reactor shutdown system ensures a reliable reactor shutdown without operator action, while PRHR through natural circulation maintains core cooling at all conditions. Furthermore, the use of LEU and a once-through fuel cycle with no on-site reprocessing enhances proliferation resistance. The high volumetric heat capacity of the graphite moderator and molten salt coolant leads to predictable thermal transients. This reduces the impact of operational changes on the reactor core and contributes to high reliability and a capacity factor greater than 90%.

7. Instrumentation and Control Systems

The I&C systems of JMR share fundamental similarities with conventional nuclear power plants, monitoring temperature, pressure, flow rate, coolant level, and neutron flux. The non-safety-related Plant Control System (PCS) manages normal operation and Anticipated Operational Occurrences (AOOs). Under all DBCs and DECs, the safety-related Reactor Protection System (RPS) provides reliable safety functions by automatically or manually tripping the reactor and activating

the PRHR system. All I&C sensors must tolerate elevated temperatures up to 700℃, intense gamma and neutron radiation, and salt corrosion, ensuring long-term stability and calibration integrity.

8. Plant Layout Arrangement

The key parts of JMR plant include an independent reactor skid module coupled to one or more integrated and, independent skid modules. These skid modules will be delivered to the end-users by truck, rail, etc and assembled on site. The highlighted mobility and adaptability of JMR to various sites including harsh, inaccessible environments allows for fast deployment of stable, sustainable and resilient power. The re-deployment to another site is also easier.

9. Testing Conducted for Design Verification and Validation

The consistency of the diameter, materials and properties of fuel assembly between JMR and JFHR provide a possibility that the experimental data of Separative Effect Tests (SETs) are directly validated between these two reactors. However, given the distinct system configuration and operational characteristics of JMR, the Integrated Effects Tests (IETs) must be redesigned.

10.Design and Licensing Status



Figure 1. JMR Plant Site Rendering

JMR design inherits from JFHR, thus significantly reducing the licensing procedure. Early pre-application discussions are ongoing.

11.Fuel Cycle Approach

The JMR adopts a once-through fuel cycle the same as JFHR. The fuel form and manufacturing process stay the same, enabling direct use of the same production line set up for JFHR. Spent fuel is stored on-site in compact storage canisters. The used fuel will finally be transported for disposal in a centralized repository. This approach simplifies fuel handling and reduces proliferation risk.

12.Waste Management and Disposal Plan

In JMR, gaseous radioactive waste is processed through high-efficiency filters before being discharged. Liquid radioactive wastes are collected in dedicated tanks, treated, cemented for solidification, and subsequently sealed for storage. Solid radioactive waste is compressed, packed, and loaded into solid waste collection containers after treatment. All radioactive waste is classified according to their post-treatment radioactivity levels and disposed of in compliance with end-user country's policies.

13.Development Milestones

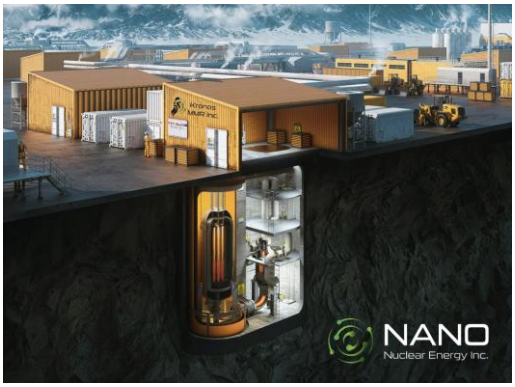
Year	Milestone	Status
2026	JMR conceptual design	Ongoing

Year	Milestone	Status
2028	Basic design	Planned
2029	Detailed design; engagement with regulator for licensing	Planned
2030	FOAK modular skids fabrication	Planned
Early 2030s	Modular skids delivery; skids mounting; first criticality	Planned
By 2030	First concrete poured	Planned

Nano Nuclear Energy, United States of America, KRONOS MMR



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Nano Nuclear Energy, United States of America
Reactor type	HTGR
Coolant/moderator	He, Graphite
Design status	Basic Design
Thermal/electrical capacity, MW(t)/MW(e)	45/15
NSSS operating pressure (primary/secondary), MPa	6/0.3
Core inlet/outlet coolant temperature (°C)	300/600
Primary circulation	Forced
Fuel type/assembly array	TRISO FCM (UCO) / Prismatic
Number of fuel assemblies in the core	150
Fuel enrichment (%)	9.9 – 19.75
Core discharge burnup (GWd/ton)	50-150
Refuelling cycle (months)	18-360
Fuel cycle requirements/approach	N/A
Reactivity control mechanism	Control rod injection, negative temperature coefficient
Approach to safety systems	Hybrid
Seismic design (SSE)	0.3g-0.6g
Design life (years)	40
Plant footprint (m^2)	4700
RV height/diameter (m)	6/4.5
RV weight (metric ton)	60

1. Introduction

KRONOS MMR is a 4th Generation nuclear energy system that is designed to provide carbon-free, high-quality process heat, for co-located industries such as data centers, chemical and industrial operations, and population centers. KRONOS MMR units are High-Temperature Gas-Cooled Reactors, designed as a stationary installation capable of producing power up to 45 MWth (15 MWe) per unit. The modular system is capable of being transported by road and assembled at site, and is designed to form part of a multisystem deployment to allow for scaling and lowering levelized cost of electricity.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☐ Baseload	☒HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☒ >550°C

A KRONOS 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

KRONOS MMR Energy System, featuring integrated thermal energy storage via a Molten Salt intermediate loop. The reactor utilizes TRISO-based Fully Ceramic Microencapsulated (FCM) fuel using LEU+ (9.9% enriched uranium).

The KRONOS MMR Energy System is suitable to be installed on remote, inland sites next to the offtake facility.

KRONOS 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 HTGRs
- Use the technology development work done in the US for NGNP and Advanced Reactors
- Avoid novel technology requiring extensive development

4. Main Design Features

(a) Nuclear Steam Supply System

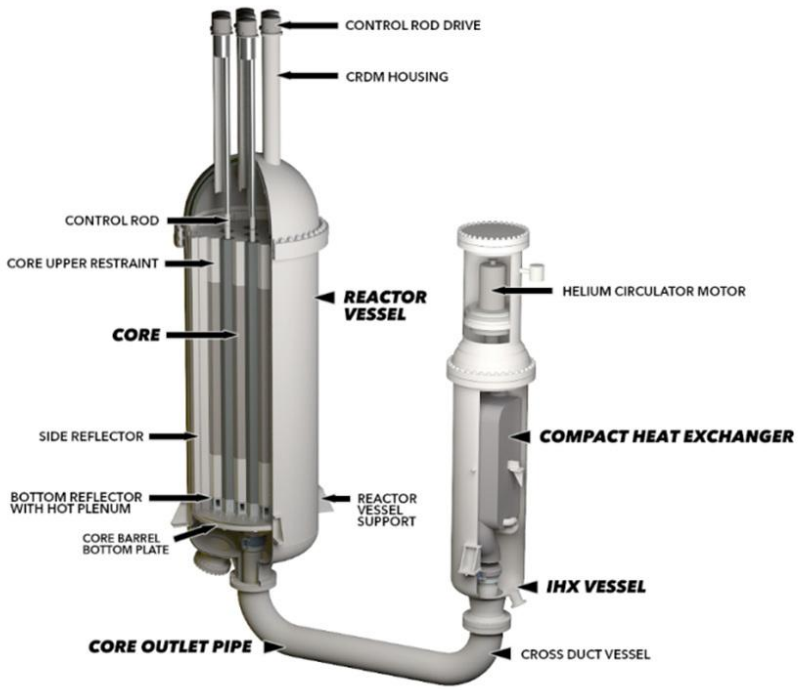


Figure 1. Nuclear Heat Supply System

Primary Circuit

The Nuclear Heat Supply System (NHSS) converts the nuclear potential energy in the fuel to usable process heat. A nuclear fission process generates heat in the reactor core. This heat is transferred via pure helium to an intermediate heat exchanger (IHX), where the heat is transferred to the molten salt loop. The helium pressure is circulated by a fully submerged circulator. After which the helium is recirculated though the reactor core again. The figure 4 shows a conceptual representation of the Nuclear Heat Supply System and identifies some of its major SSCs. It should be noted that the helium is a heat transfer fluid and not a conventional nuclear "coolant", as a loss of helium would have no effect on the fuel integrity.

The NHSS can be configured to operate within a large range of power levels, utilizing varying fuel enrichment levels to match different project objectives. This flexibility is achieved by adjusting the operating pressure of the helium. The design point is fixed prior to construction of a specific project, as some of the customizations are permanent.

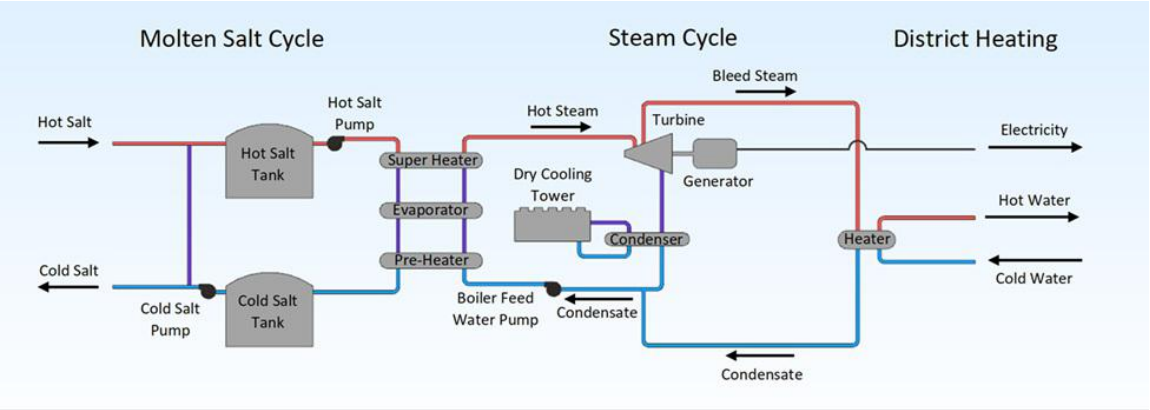


Figure 2. Power Cycles - Simplified Process Flow Diagram

Secondary Side

The plant makes use of a molten salt energy storage system (commercialized through concentrated solar power development) combined with a conventional Rankine steam cycle for power conversion. The inclusion of thermal storage allows the power cycle to provide more energy to the client than the maximum power output of the reactor, for limited periods of time.

The energy storage, steam, and power conversion system removes heat and directs heat energy from the heat transport system. Hot molten salt is received from the intermediate heat exchanger and stored in a hot salt tank. This tank serves as an energy storage system. Molten salt from the hot salt tank is directed to a steam generator. The heat available in the molten salt is removed through the generation of steam. The steam energy is converted to electrical power and low-grade district heat through the turbine generator and district heat exchanger. Steam extracted from the turbine interstage is used to generate low-grade district heat.

(b) Reactor Core

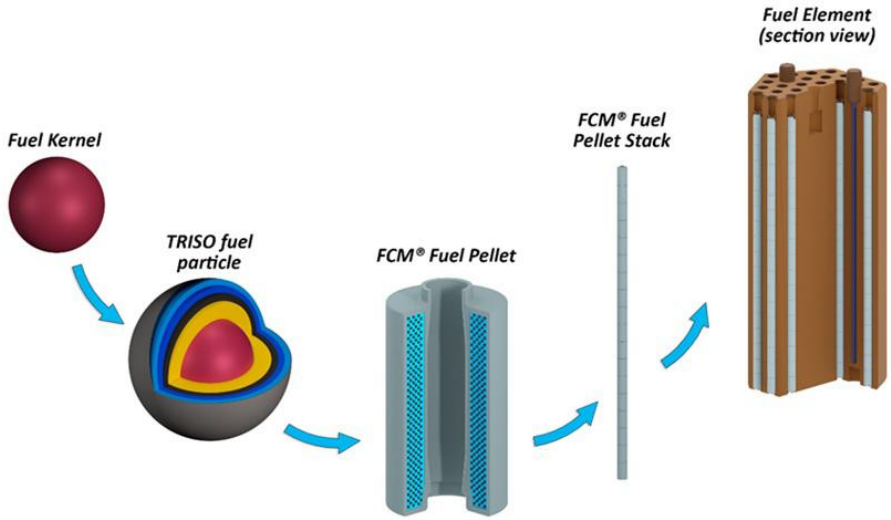


Figure 3. Integration of the fuel element components

The active reactor core consists of hexagonal graphite blocks that contain stacks of the FCM fuel pellets. The KRONOS 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.

The nuclear fuel used in KRONOS MMR has been developed through extensive operating experience with helium cooled reactors and test programs. A primary objective of the fuel design is to provide a high integrity barrier, serving to localize radioactive fission products at the point of production.

KRONOS MMR uses Fully Ceramic Micro-encapsulated (FCM) fuel. FCM fuel provides a new approach to inherent reactor safety. Industry standard TRISO fuel particles, which contain the radioactive byproducts of fission within layered ceramic coatings, are encased within a silicon carbide matrix to form fuel pellets. This is like encasing the fuel in a diamond-like substance. The combination of TRISO fuel particles and the silicon carbide coating provides an extremely rugged and stable fuel with extraordinary high temperature stability.

The refractory SiC pellet matrix offers excellent thermal and chemical stability which effectively precludes melt-down or fission product release due to chemical attack. It also protects the TRISO particles from mechanical damage and adds an additional barrier for long-term retention of radioactive fission products. Since multiple diverse barriers protect against fission product release, the FCM fuel provides functional containment for KRONOS MMR and a containment building with stringent leakage requirements is not needed.

The annular-shaped fuel pellets are stacked in holes drilled in hexagonal graphite blocks to form fuel elements. The annular shape allows the fuel pellets to be cooled on both the inside and outside surfaces, reducing the thermal stress and temperature in the pellet, which decreases the probability of failure of a pellet and reduces diffusion of fission products.

(c) Reactivity Control

For reactor control and protection, KRONOS MMR uses control rods placed within the core area. Each control rod contains boron carbide neutron absorber. To control the reactor, the boron carbide segment is raised or lowered within a control rod channel. To provide a diverse and redundant method for reactor shutdown, the inherent reactivity feedback of the core is used by allowing the core to heat up slightly and shutdown by itself.

Fission product retention in the KRONOS MMR reactor starts at the fuel. The FCM fuel elements consist of Tri-structural Isotropic (TRISO) nuclear fuel particles packed within a silicon carbide matrix. The silicon carbide within the fuel particles and the surrounding matrix serves as a very high temperature barrier to prevent release of radioactive fission products.

(d) Reactor Vessel (RV) and Internals

The Reactor Pressure Vessel (RPV) houses the core. KRONOS MMR avoids major water ingress events because there is no steam generator in the primary circuit. The KRONOS 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

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

The KRONOS MMR Energy System uses standardized modules which will be to the extent possible assembled, commissioned and tested off-site, prior to transport, and then installed at the project site. Piping, cabling, lighting, etc., will be included in the modules with specific interfaces for ease of connection during installation. Minimum work is foreseen for assembling the modules at the project site.

5. Safety Features

The safety features of KRONOS MMR include the safety characteristics common to all HTGRs, as well as features unique to the particular configuration of KRONOS MMR. The general safety characteristics of an HTGR design are strongly affected by the inherent characteristics of the helium heat transfer fluid, core materials and fuel, as described below:

Helium Heat Transfer Fluid – The inert, single-phase gaseous heat transfer fluid has several advantages. A helium system has the following advantages over a water system. Flashing or boiling of the heat transfer fluid is impossible, pressure measurements are certain, no heat transfer fluid level measurements are required, no hydraulic ram, and pump cavitation cannot occur. Furthermore, there are no reactivity effects associated with the helium and no chemical reaction is possible between the heat transfer fluid and fuel or fuel particle coatings.

Graphite Core – The strength of the graphite core and the stability of the ceramic FCM fuel at high temperatures combine to create a wide margin between normal operating temperatures and those highly elevated temperatures necessary to

result in core damage. Furthermore, the high heat capacity and low power density of the core result in very slow, predictable temperature transients.

Coated Fuel Particles – The multiple ceramic coatings of the TRISO particles surrounding the fuel kernels constitute tiny independent fission product retention spheres. Because of the ceramic material, these coatings can maintain their integrity to very high temperatures in excess of 1,600°C.

KRONOS MMR has been designed to take advantage of these inherent characteristics so that no reliance is placed on active or powered systems, or on operator actions to accomplish safety functions. The geometry and size of the reactor core, its power density, and the uninsulated steel reactor vessel have been selected to allow decay heat removal from the core to the ultimate heat sink through the natural processes of radiation, conduction and convection. A passive system removes heat from the reactor during normal operating modes and accident conditions; however, there is sufficient capacitance and thermal conductivity such that the fuel temperature limits are not exceeded during DBAs and BDBAs, even in the absence of water in the RCCS for heat absorption. As a result, KRONOS MMR can accommodate a loss of helium in combination with the loss of all forced circulation, from full power, without fuel temperatures exceeding a level at which significant fuel particle failure would occur.

KRONOS MMR has additional unique safety features which enhance the safety characteristics of the HTGR described above:

FCM Fuel – The FCM fuel encapsulates the TRISO particles in a silicon carbide matrix with a fully dense silicon carbide layer on the surface of the pellet. This outer dense silicon carbide layer adds an additional barrier for the retention of fission products from failed TRISO particles, diffusion through intact TRISO particles, and any free uranium from the TRISO manufacturing process. The FCM fuel is the functional containment of KRONOS MMR and is a fission product barrier that effectively limits the physical transport and release of radionuclides to the environment across the full range of normal operating conditions, Anticipated Operational Occurrences (AOOs) and accident conditions. This functional containment can be relied upon to ensure that the dose at the site boundary as a consequence of postulated accidents meets regulatory limits.

The above-described passive features are intrinsic. They require neither operator action nor active power systems. By combining the inherent characteristics common to all HTGRs with the special features unique to the KRONOS MMR design, the level of safety that is provided by KRONOS MMR is consistent with nuclear regulatory requirements.

The design ensures that the following key safety features are met:

- No requirement for electrical power during Design Basis Accidents (DBAs),
- No requirement for engineered control actions during DBAs,
- No requirement for operator action during DBAs,
- Low personnel radiation dose.

(a) Engineered Safety System Approach and Configuration

The safety concept for KRONOS MMR 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 660 °C, the peak fuel temperature can be kept far below the fuel safety limit of 1600 °C. KRONOS MMR's small size means the reactor has a high surface area to volume ratio. Fission product decay heat is conducted into the graphite radial reflector to the reactor pressure vessel. The heat can then be convected and radiated to the surrounding shielding and containment structure. KRONOS MMRs uses two diverse and redundant methods for insuring shutdown. The control rods arranged in the core, and the negative reactivity coefficient. KRONOS MMR leverages multiple defense-in-depth features including a helium purification system, an ASME-rated primary loop pressure boundary, the Citadel Building, and a site boundary that is also the exclusion area boundary. 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 reactor cavity cooling system. Radioactive fission products will be contained within the FCM fuel. Fast neutron and gamma radiation will be blocked by the external shielding structure.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) Containment System

KRONOS 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.

(g) Spent Fuel Cooling Safety Approach / System

6. Plant Safety and Operational Performances

7. Instrumentation and Control Systems

8. Plant Layout Arrangement



Figure 4. NANO's Kronos MMR

A KRONOS MMR Energy System consists of two plants:

- The Nuclear Plant: consists of the KRONOS MMR nuclear reactor unit(s) and includes all the equipment required to transport the heat produced to the adjacent plant.
- The Adjacent Plant: is a non-nuclear facility which harnesses the heat for client specified applications. These applications could be electricity generation, process heat (for example, steam), hydrogen production, district heating etc.

The design embraces the separation principle, in which the nuclear plant is physically separate from the adjacent plant, which reduces the nuclear footprint of the overall facility. Figure 4 shows an artistic rendering of a KRONOS MMR Energy System. 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.

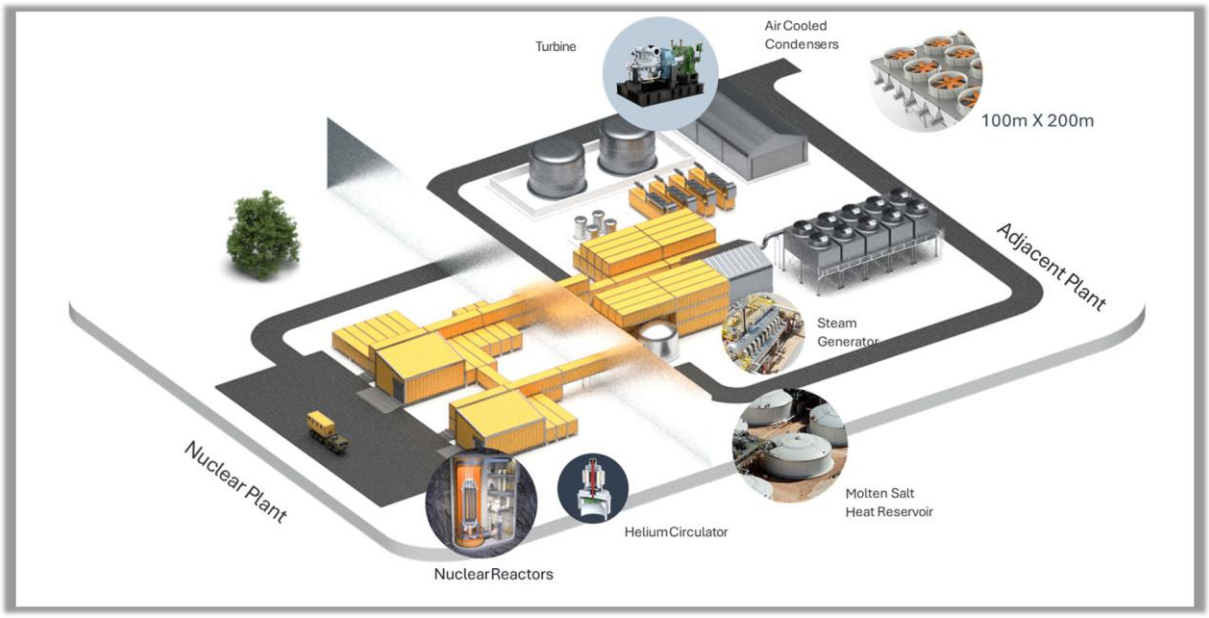


Figure 5. Kronos MMR Generic Plant Separation

Figure 5 shows an illustrative rendering of the separation between nuclear plant and adjacent plant, for a 2-unit layout.

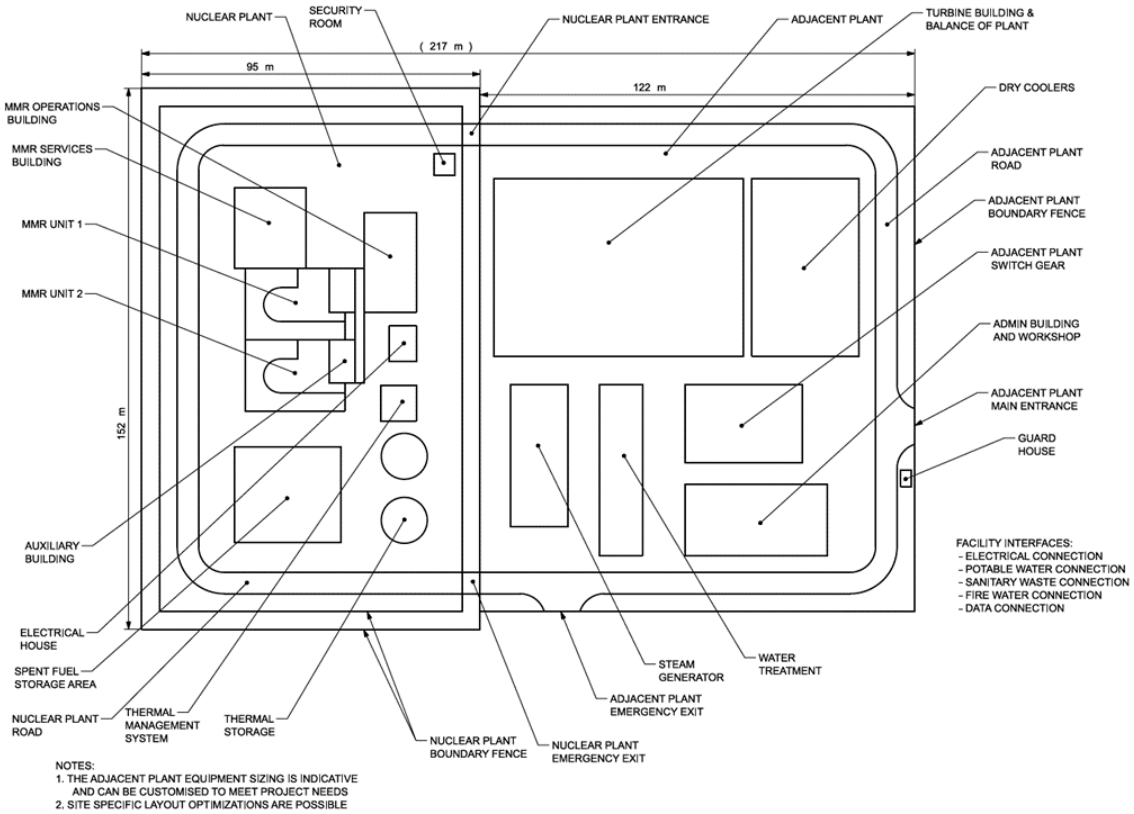


Figure 6. Typical Kronos MMR energy system - plot plan

A typical installation of a KRONOS MMR Energy System may consist of the following buildings and structures:

Citadel Building – The Citadel Building is a vertical, reinforced concrete cylindrical structure set below grade. The citadel houses the entire NHSS and some of its supporting services. The citadel acts as a barrier from external hazards while also providing biological shielding against radiation exposure from the reactor.

Unit Auxiliary Building – Each Citadel Building has a dedicated Auxiliary Building, located at gradelevel, which is comprised of several modules that contain the balance of the supporting equipment for the NHSS.

Maintenance Enclosure – The Maintenance Enclosure resides over top of the Citadel Building and provides shelter over the operating floor. To facilitate refueling activities, the floor at grade includes a set of heavy-duty rails that allow the reactor refueling device to travel linearly along the centerline of the reactor and the spent fuel cask pits. Additionally, the structure provides the ability to confine potential airborne contaminants during the refueling process.

Reactor Services Building – The Reactor Services Building is a steel-clad warehouse-type structure adjoined to the Maintenance Enclosure. Reactor support activities taking place within the building include handling of new and spent fuel, and maintenance of activated or contaminated reactor components. The Reactor Services Building is designed to be shared among multiple reactor units.

Operations Building – The Operations Building is located at grade level and is comprised of an arrangement of modules that house the control room, electronics room, and activities associated with plant and radiation protection, administration and security services. The plant control, data, and instrumentation systems control/display panels and computers are housed in the control room. The Operations Building is designed to be shared among multiple reactor units.

Security Room – The Security Room provides primary access to the MMR Product facilities.

9. Testing Conducted for Design Verification and Validation

10.Design and Licensing Status

NANO Nuclear is collaborating with the University of Illinois at Urbana-Champaign with the aim of demonstrating the KRONOS MMR's high technology readiness level. The first KRONOS MMR will be constructed on the University's campus, for use as a research and training facility. In parallel, NANO Nuclear is continuing the licensing process with the U.S. Nuclear Regulatory Commission. Further demonstrations of KRONOS MMR are expected to take place at Chalk River at the Canadian Nuclear Laboratories. KRONOS MMR is the first small modular reactor to enter the Canadian Nuclear Safety Commission’s formal licensing review.

11.Fuel Cycle Approach

Refueling frequency is mainly determined by reactor power and fuel enrichment. As the output power increases, the duration between fueling cycles decreases. Equally, as the fuel enrichment decreases, the duration between fueling cycles decreases. During refueling operations, some or all of the fuel elements in the reactor core are removed and replaced with fresh fuel element. Other internal components (such as reflector elements, control rod drives and control rods) are replaced as required.

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.

12.Waste Management and Disposal Plan

13.Development Milestones

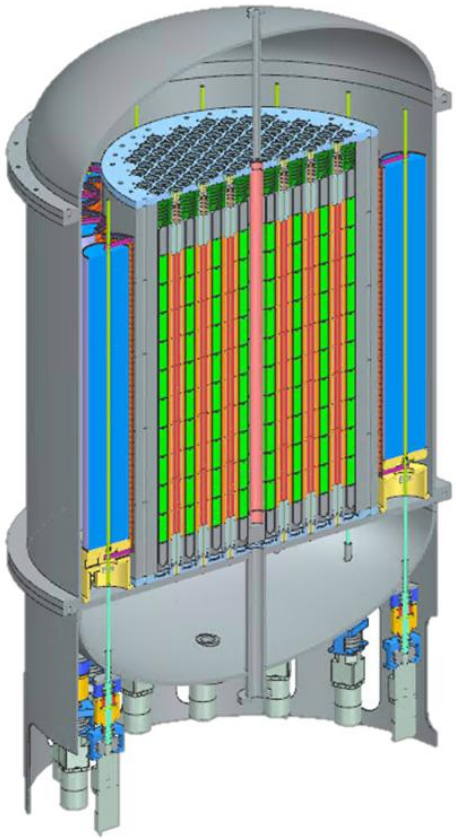
Year	Milestone	Status
2011	Secured FCM fuel and MMR reactor patents	**Completed**
2016	Established R&D and fabrication laboratories	**Completed**

Year	Milestone	Status
2017	Initiated FCM fuel qualification plan; Started Vendor Design Review Phase I in Canada	**Completed**
2018	GFP submits proposal to CNL, supported by OPG & Ultra Safe Nuclear Company	**Completed**
2019	Started Vendor Design Review Phase II in Canada; Started license to prepare site application in Canada	**Completed**
2021	Start NRC engagement in USA	**Completed**
2028	Demonstration project construction	**Ongoing**

Nano Nuclear Energy, United States of America, LOKI MMR



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Nano Nuclear Energy, United States of America
Reactor type	HTGR
Coolant/moderator	He/Metal Hydride
Design status	Conceptual Design
Thermal/electrical capacity, MW(t)/MW(e)	- / 1.5-5
NSSS operating pressure (primary/secondary), MPa	2/20
Core inlet/outlet coolant temperature (°C)	327/727
Primary circulation	Forced
Fuel type/assembly array	TRISO FCM (UCO)/Prismatic
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	9.9
Core discharge burnup (GWd/ton)	2.5
Refuelling cycle (months)	30
Fuel cycle requirements/approach	N/A
Reactivity control mechanism	Control Drums
Approach to safety systems	Hybrid
Seismic design (SSE)	N/A
Design life (years)	N/A
Plant footprint (m^2)	500
RV height/diameter (m)	1.5/1.65
RV weight (metric ton)	2

1. Introduction

NANO Nuclear Energy Inc. is developing the LOKI MMR, a portable nuclear reactor designed for versatility in application and deployment. Designed to provide between 1.5MWe and 5MWe of power, it is tailored to specific applications ranging from remote terrestrial, marine, and space deployments. Its transportability via road, rail, sea, and air ensures adaptability for diverse deployment scenarios and supports scalability through interconnected systems to meet greater energy demands. As a 10-ton class micro reactor, the LOKI MMR architecture builds on NANO's KRONOS MMR and Fully Ceramic Microencapsulated (FCM) fuel technologies to be capable of safely generating electrical and thermal power anywhere.

As part of the development process, NANO is pursuing the deployment of LOKI MMR for demonstration at the National Reactor Innovation Center (NRIC) Demonstration of Microreactor Experiments (DOME) facility at Idaho National Laboratory. Demonstration will allow validation of the technology including licensability, performance, operations, safety, and cost.

2. Target Application

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

3. Design Philosophy

The LOKI MMR contains a small, high-temperature gas-cooled reactor (HTGR) that utilizes Fully Ceramic Microencapsulated (FCM) nuclear fuel, metal hydride neutron moderator, and graphite control drums.

LOKI MMR 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.

The overarching design philosophy maintains a very high degree of conservatism, coupled with a simple reactor plant design that enables reaching LOKI MMR 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 LOKI MMR demonstration system are:

- Achieve 1 MWth power
- Achieve 1000 K reactor outlet temperature

4. Main Design Features

(a) Nuclear Steam Supply System

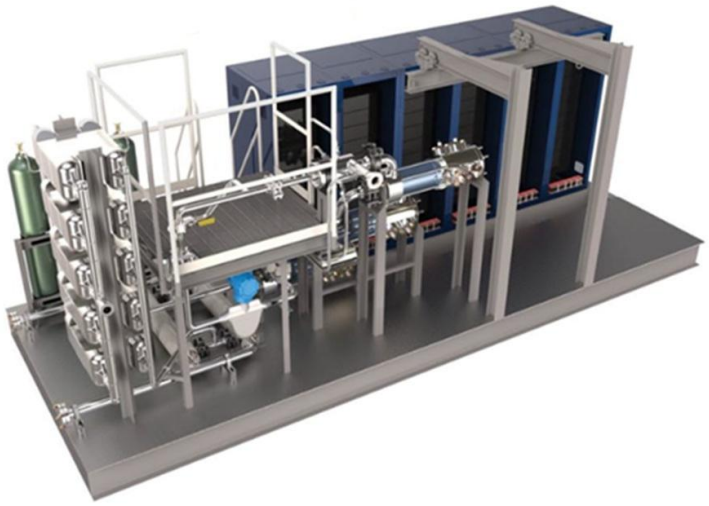


Figure 1. PTT's Advanced Closed-Cycle Merlin 1MW Turbine Engine for Ground-Based Power

Primary Circuit

The LOKI MMR reactor and helium primary loop will operate within the EBR-II 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. The helium primary loop will transfer heat from the reactor through penetrations in the shielding structure to the primary heat exchanger.

Secondary Side

The supercritical CO2 secondary loop will transfer heat from the primary heat exchanger through penetrations in the DOME containment structure for rejection to the surrounding environment. NANO has partnered with Peregrine Turbine Technologies (PTT) for development of the supercritical CO2 secondary loop.

(b) Reactor Core

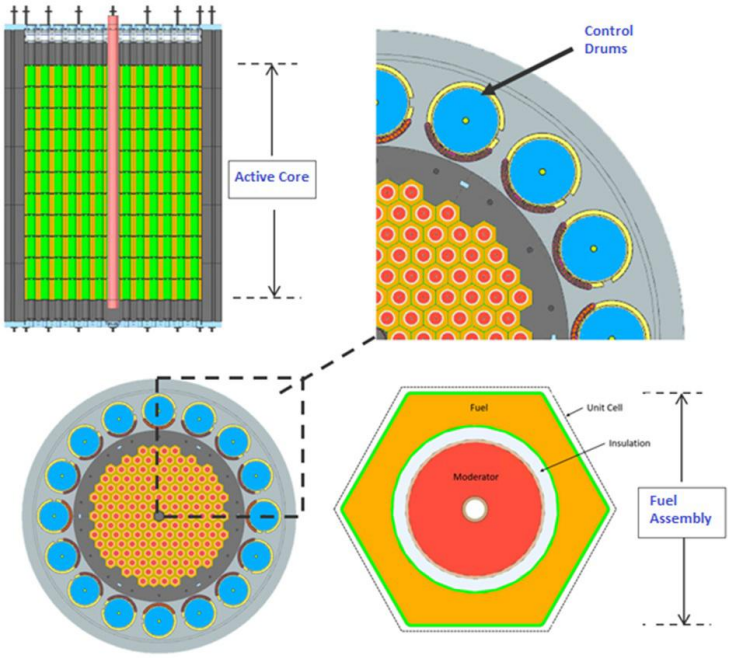


Figure 2. Conceptual arrangement of the LOKI MMR reactor and fuel assembly

The LOKI MMR 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.

The reactor core is controlled through a series of control drums located in the radial graphite reflector and safe shutdown is assured with a redundant and independent safety shutdown rod at the reactor centreline.

Prototype LOKI MMR FCM elements, shown in Figure 1, were previously manufactured at USNC's pilot fuel manufacturing facility in Oak Ridge, TN. The capability to additively manufacture FCM fuel is being developed jointly between Framatome and Standard Nuclear. The method combines binder jet printing and chemical vapor infiltration in a process capable of yielding a high-purity, >93% dense ceramic matrix; attributes essential for high-performance nuclear fuel capable of retaining fission products in off-nominal conditions.



Figure 3. Additively manufactured, chemical vapor infiltrated SiC LOKI fuel element with prototypic size and shape and material. Filled with SiC surrogate powder

The high TRISO packing fraction (>58%) enabled through the additive manufacturing process enables LOKI MMR to use 9.9% enriched uranium (LEU+) for fuel. This significantly de-risks the fuel supply chain. When commercial HALEU is available, NANO will have the option of fueling and/or refueling LOKI MMR reactor cores with HALEU. This flexible approach enables NANO to take advantage of time to market and then to pivot to HALEU when readily available.

(c) Reactivity Control

For reactor control and protection, LOKI MMR 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.

Fission product retention in the LOKI MMR reactor starts at the fuel. The FCM fuel elements consist of Tristructural Isotropic (TRISO) nuclear fuel particles packed within a silicon carbide matrix. The silicon carbide within the fuel particles and the surrounding matrix serves as a very high temperature barrier to prevent release of radioactive fission products.

(d) Reactor Vessel (RV) and Internals

(e) Reactor Coolant System and Steam Generator

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

5. Safety Features

The safety concept for LOKI MMR 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. LOKI MMR's small size means the reactor has a high surface area to volume ratio. Fission product decay heat is conducted into the graphite radial reflector to the reactor pressure vessel. The heat can then be convected and radiated to the surrounding shielding and containment structure. LOKI MMR uses two diverse and redundant methods for insuring shutdown: the control drums arranged in the radial reflector, and a center safety shutdown rod. The center safety shutdown rod, which is fully removed from the reactor during operation, contains enough worth to overcome the worth of the control drums during the reactor lifetime. The control drums will also automatically insert in the event of a reactor trip. The action of either control system will insure reactor shutdown. LOKI MMR leverages multiple defense-in-depth features including a helium purification system, an ASME-rated primary loop pressure boundary, the DOME confinement building, and an operation location with significant distance from population. By relying on FCM fuel and its small size, LOKI MMR can remove all heat related safety systems, allowing for greater simplicity and increased robustness. 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.

(a) Engineered Safety System Approach and Configuration

The safety goals in nuclear facility design and operation are to ensure adequate protection of the public, workers, and environment. A primary assumption for the protection of the public, workers, and environment is to use inherently safer design concepts in the reactor designs with a performance-based graded approach. In a general sense, the performance-based approach can be summarized by identifying four Fundamental Safety Functions (FSFs): 1) reactivity control, 2) heat removal, 3) radioactive material retention, and 4) shielding. The control reactivity and remove heat functions are typically necessary to prevent conditions that would challenge temperature and pressure design limits of the radionuclide confinement barriers and preserve the barrier's integrity. The LOKI MMR design will ensure that these fundamental safety functions are satisfied through a combination of Structures, Systems, and Components (SSCs) with appropriate safety classification and associated evaluations for adequacy of Defense-in-Depth (DID) based on the reactor hazard analysis.

LOKI MMR 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. LOKI MMR 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.

LOKI MMR uses FCM Fuel, which contains robust silicon carbide radionuclide barriers. These barriers start with the use of AGR-based TRISO particle fuel, which are then encapsulated in a fully ceramic silicon carbide matrix. A fuel safety case technical report will be developed and delivered alongside submission of the preliminary documented safety analysis (PDSA). Operational temperatures will be limited to hundreds of degrees centigrade below fuel safety limits to provide additional safety margin, to be verified during test. A central component to a layered approach is the use of successive, multiple physical barriers for protection against release of radioactivity and hazardous materials. Beyond the FCM fuel providing layers of protection against the release of fission products, the American Society of Mechanical Engineers (ASME)-rated primary coolant boundary (Reactor Pressure Vessel (RPV) and piping) provides an additional barrier for radionuclide retention.

LOKI MMR will use helium as a primary coolant to transfer heat from the reactor core to the primary heat exchanger, however the primary coolant will not be necessary for heat removal in loss of forced coolant events. The low power of LOKI MMR, its small size, and use of high temperature capable AGR-based TRISO particle fuel enable decay heat to be conducted to the surface of the RPV and rejected to the ambient environment through natural convection and radiation, even with additional shielding installed.

For shielding, LOKI MMR will utilize shielding that will work in conjunction with the radial and axial neutron reflectors to reduce neutron and gamma radiation to the surrounding environment. The shielding will also reduce activation of the reactor pressure vessel and nearby components, aiding in decontamination and decommissioning. LOKI MMR electrical components, such as the controller and motors, may have shielding to allow the components to remain functional. Operation of LOKI MMR will take place within a shielded structure within the DOME test bed.

The selection of reactor-specific SSCs in the addendums follows the hierarchy of controls listed in DOESTD-1189-2016, such that safety SSCs are derived based on the following order of preference:

- SSCs are preferred over administrative controls.
- Passive SSCs are preferred over active SSCs.
- Preventive controls are preferred over mitigative controls.
- Controls closest to the hazard may provide protection to the largest population of potential receptors, including workers and the public.
- Controls that are effective for multiple hazards can be resource effective.

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

The safety goals will also be fulfilled by an approach that ensures adequate DID by providing multiple independent layers of protection to prevent the release of radioactive and hazardous materials or, if prevention fails, to limit the potential consequences of accidents and to prevent their evolution to more serious conditions. The DOE philosophy for DID is best summarized in DOE G 420.1-1A, “Nonreactor Nuclear Safety Design Guide for use with DOE O 420.1C, Facility Safety.”10 Section 7 discusses what DOE guidelines LOKI MMR will follow for hazard identification while operating at DOME.

The LOKI MMR design approach will adopt the five levels of DID in the DOE G 420.1-1A guidance to ensure adequate plant capability DID in the reactor design:

- Level 1: Prevention of abnormal operation and failures
- Level 2: Control of abnormal operation and detection of failures
- Level 3: Control of accidents within the design basis
- Level 4: Control of severe facility conditions
- Level 5: Mitigation of radiological consequences

(e) Safety Approach and Configuration to Manage DEC

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.

(f) Containment System

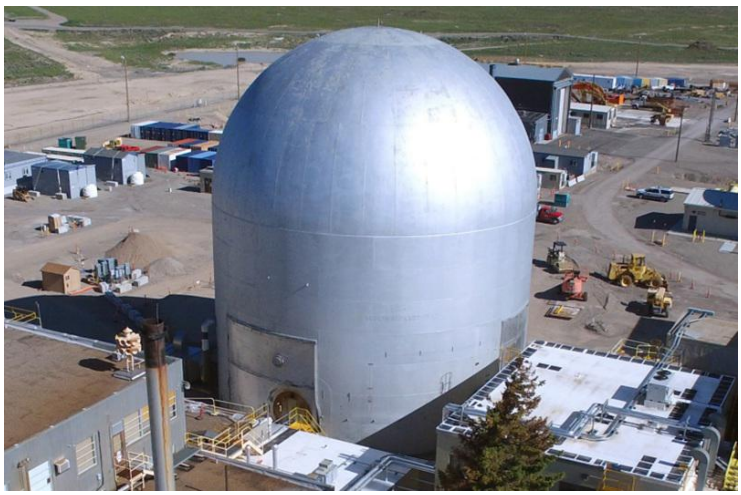


Figure 4. Exterior view of EBR-II containment, the key structure composing the DOME facility

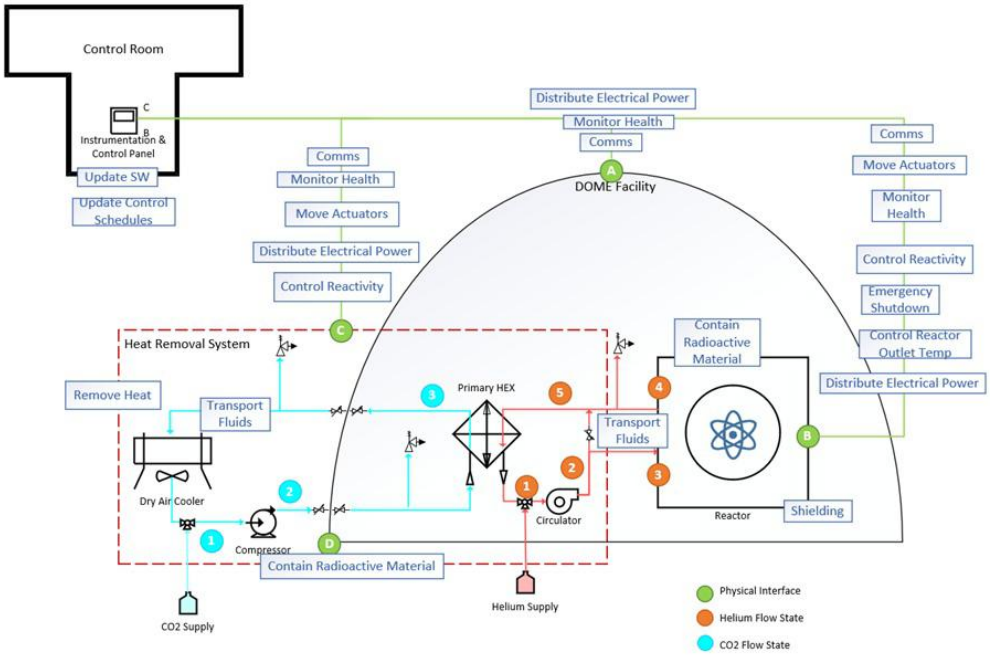


Figure 5. Key Interfaces and Functions of the LOKI MMR System within the DOME Facility

LOKI MMR will be operated within the DOME facility at Idaho National Laboratory. The DOME facility is a reuse of the Experimental Breeder Reactor II (EBR-II) containment. The interior of the containment dome is 25 m tall and 21 m diameter.

6. Plant Safety and Operational Performances

7. Instrumentation and Control Systems

8. Plant Layout Arrangement

LOKI MMR 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 NANO 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 the LOKI MMR demonstration. Its low power and use of TRISO fuel enables the reactor to be passively cooled within the DOME facility without the use of any external systems. The safety of collocated workers at INL's MFC and members of the public at the INL boundary will not be affected if cooling is lost from the reactor during operation. As a demonstration reactor, LOKI MMR will not be connected to a private or local electricity grid. Any power generated during operation will be converted to heat and transferred to the surrounding environment.

9. Testing Conducted for Design Verification and Validation

To operate the LOKI MMR demonstration reactor, NANO will work with INL to integrate within the DOME facility. To allow the reactor to operate within the DOE authorization structure, INL will function as the design authority. In this arrangement, NANO will transfer design basis documentation to INL for finalization and approval. Operators for the LOKI MMR demonstration will train under INL supervised training programs.

Overnight capital cost for the LOKI MMR demonstration system is estimated overnight capital cost of $20 million. Funding for the LOKI MMR demonstration system will be provided entirely by NANO as part of development cost for the LOKI MMR nuclear system product

10.Design and Licensing Status

To enable operation of LOKI MMR, Idaho National Laboratory will submit licensing documentation to the US Department of Energy (DOE) for review and approval. USNC and Idaho National Laboratory 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 and submission of the CSDR is expected in December 2025. Additionally, as part of FEEED, Nano will undergo a quality assurance audit to show compliance with a subset of ASME NQA-1 standards. Due to LOKI MMR's small size and skid mounted architecture, the reactor will be factory assembled and shipped to the operations site following factory acceptance testing. Lessons learned in the design, construction, operation, and decommissioning of LOKI MMR will influence the production of later versions for commercial operation. Later versions of LOKI MMR will have increased performance targets (power, temperature, size, mass) and improved cost targets. Future versions of LOKI MMR will have equivalent or smaller size to the demonstration reactor, allowing continued factory production. Factory production of LOKI MMR systems will allow very high levels of standardization for deployed plants with greatly reduced construction times. Future versions of LOKI MMR will require different levels of reliability and maintainability for systems and equipment.

11.Fuel Cycle Approach

The fuel source for the LOKI MMR 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 LOKI MMR reactor is the same design used in NANO's Nano's KRONOS UIUC and Chalk River projects and will be produced alongside them in the planned fuel production factory operated jointly by Standard Nuclear and Framatome. UCO was chosen for LOKI MMR 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

13.Development Milestones

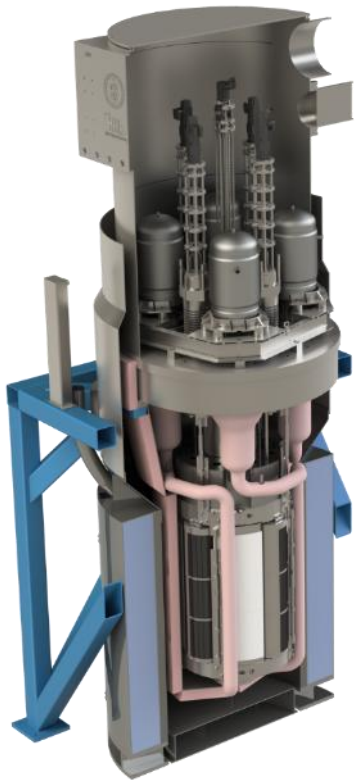
Year	Milestone	Status
2022	Preliminary studies and technological innovation (using previously developed patents)	**Completed**

Year	Milestone	Status
2023	Pre-conceptual design phase and technology validation	**Completed**
2024	Conceptual Design Phase	**Completed**
2026	Detailed Design Phases	**Ongoing**
2027	Procurement of Components and Acceptance Testing	**Planned**
2028	Assembly, Commissioning and Operation at NRIC DOME Facility	**Planned**
2030	Disassembly, Decommissioning, and Disposal	**Planned**

Idaho National Laboratory
United States of America
MARVEL



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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*
Design status	*Equipment manufacturing in progress*
Thermal/electrical capacity, MW(t)/MW(e)	*0.075-0.1 / 0.015-0.027*
NSSS operating pressure (primary/secondary), MPa	*0.39 / 0.22*
Core inlet/outlet coolant temperature (°C)	*450 / 520*
Primary circulation	*Natural Convection*
Fuel type/assembly array	*Uranium Zirconium Hydride*
Number of fuel assemblies in the core	*6*
Fuel enrichment (%)	*19.75%*
Core discharge burnup (GWd/ton)	*1.50 (current; can be increased if needed)*
Refuelling cycle (months)	*> 60*
Fuel cycle requirements/approach	*System life is limited to 2 years*
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*
Seismic design (SSE)	*IBC 2015 Risk Category IV Site Class C*
Design life (years)	*2-40 (depending on transients)*
Plant footprint (m^2)	*8.9*
RV height/diameter (m)	*2.55 / 1.22*
RV weight (metric ton)	*5.94*

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

☐ Electricity	☐ Hydrogen	☐ Industrial process heat
☐ Baseload	☐HTSE - electrolysis	☐ <200°C
☐ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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:

Test, demonstrate, and address issues related to installation, startup, and operation

- 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.

Enable Automated Operation Technologies

- 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.

Enable Application Integration

- 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) Nuclear Steam Supply System

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.

Fuel Characteristics

The major components of the MARVEL fuel system are

(1) U-ZrH1.6 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.

(c) 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.

(d) Reactor Vessel (RV) 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.

(e) Reactor Coolant System and Steam Generator

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.

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.



(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

Compact core; naturally driven, adjustable high-grade heat extraction; small footprint

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.

(a) Engineered Safety System Approach and Configuration

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) Containment System

(g) Spent Fuel Cooling Safety Approach / System

6. Plant Safety and Operational Performances

N/A

7. Instrumentation and Control Systems

N/A

8. Plant Layout Arrangement

N/A

9. 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.

10. 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.

11. Fuel Cycle Approach

N/A

12.Waste Management and Disposal Plan

N/A

13.Development Milestones

N/A

Toshiba Energy Systems & Solutions Japan
Movelux



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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)*
Design status	*Conceptual design*
Thermal/electrical capacity, MW(t)/MW(e)	*10 / 3 – 4*
NSSS operating pressure (primary/secondary), MPa	*0.1 /0.3*
Core inlet/outlet coolant temperature (°C)	*Heat pipe: 680 / 685 Heat Exchanger: 450 / 680*
Primary circulation	*Natural*
Fuel type/assembly array	*Silicide (U 3 Si 2) / Hexagonal*
Number of fuel assemblies in the core	*66 (fuel), 72 (CaH 2)*
Fuel enrichment (%)	*4.8 – 5.0*
Core discharge burnup (GWd/ton)	*1.0*
Refuelling cycle (months)	*Continuous*
Fuel cycle requirements/approach	*Either once-through or closed fuel cycle scheme depending on country's fuel cycle policy*
Reactivity control mechanism	*In-Ga Expansion Module (IGEM)*
Approach to safety systems	*Active / Passive*
Seismic design (SSE)	*0.3 g*
Design life (years)	*10 – 15*
Plant footprint (m^2)	*100*
RV height/diameter (m)	*6 .0 / 2 .0*
RV weight (metric ton)	*TBD*

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 (CaH2) capable of operating at up to 800°C is adopted for the moderator material.

2. Target Application

☒ Electricity	☒ Hydrogen	☒ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

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

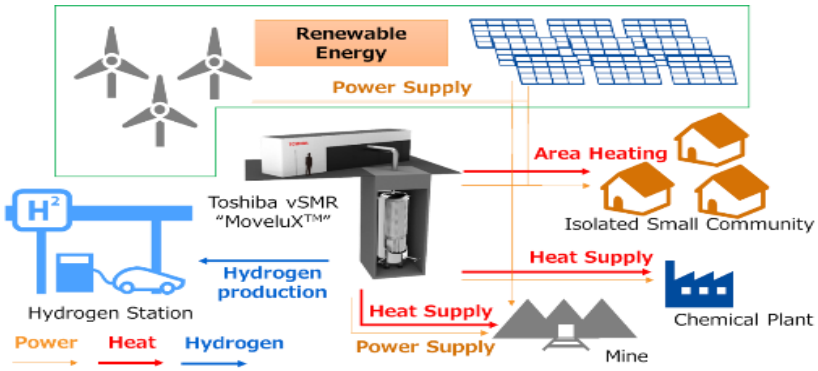


Figure 1. Utilization image of the MoveluX

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.

4. Main Design Features

(a) Nuclear Steam Supply System

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.

Fuel Characteristics

The Uranium silicide fuel has high melting point and large thermal conductivity.

(c) 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).

(d) Reactor Vessel (RV) and Internals

Reactor vessel with atmospheric pressure is adopted

(e) Reactor Coolant System and Steam Generator

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.

(f) Pressuriser

(g) Primary pumps

(h) Distinguishing features

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

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) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

(f) 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.

(g) Spent Fuel Cooling Safety Approach / System

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.

Core Damage Frequency (CDF) : Extremely low.

Refueling : No refueling during the plant life time.

7. Instrumentation and Control Systems

Few I&C devices are installed to the MoveluX reactor system for the reactor start-up, monitoring and active control. Technically, the manned operation is not required during nominal operation in current design concept.

8. Plant Layout Arrangement

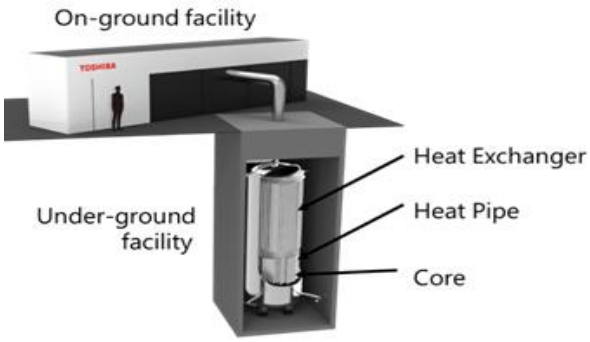


Figure 2. Schematic view of the MoveluX reactor system

9. 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.

10. Design and Licensing Status

MoveluX is at the conceptual design stage.

11. 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.

12. 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.

13.Development Milestones

Year	Milestone	Status
2015	Start fundamental study based on the space reactor design	Completed
2017	Complete reactor type decision	Completed
2019	Start concept design	Completed
2028	Complete concept design and component demonstration	Planned
2030	Complete system demonstration without nuclear fuel	Planned
2035~	FOAK demonstration	Planned

NIKIET, Russian Federation
SHELF-M



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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*
Design status	*Detailed design*
Thermal/electrical capacity, MW(t)/MW(e)	*35 / up to 10*
NSSS operating pressure (primary/secondary), MPa	*14.7 / 3.7*
Core inlet/outlet coolant temperature (°C)	*271 / 308*
Primary circulation	*Natural (for 20 % of rated power) / forced*
Fuel type/assembly array	*Cermet (UO 2 + metallic silumin) / hexagonal*
Number of fuel assemblies in the core	*n.c.*
Fuel enrichment (%)	*Up to 19.7*
Core discharge burnup (GWd/ton)	*90*
Refuelling cycle (months)	*(max local core discharge 162)*
Fuel cycle requirements/approach	*96*
Reactivity control mechanism	*Once-through fuel cycle*
Approach to safety systems	*Control rods driving mechanism*
Seismic design (SSE)	*Combined active and passive*
Design life (years)	*VIII by MSK-64 scale*
Plant footprint (m^2)	*60*
RV height/diameter (m)	*16 800*
RV weight (metric ton)	*Power capsule: 10 / 8*

1. Introduction

The SHELF-M is a modernized SMR for small NPP based on SHELF reactor unit with an increased power capacity of 35 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. Refueling and nuclear waste removal will be carried out at dedicated facilities. Optimized core design allows increasing refueling cycle for up to 8 years.

2. Target Application

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☐ District Heating / Steam	☐ Desalination	☐ >550°C

The SHELF-M is a SMR designed for land-based NPP that intended to supply electric power to remote areas with decentralized power supply. The plant does not require local water sources - residual and decay heat removed by external heat exchangers cooled by atmospheric air. The SHELF-M designed to operate with low personnel numbers with only 15 staff members required to be present of site during normal operation.

3. Design Philosophy

The SHELF-M based on proven technology derived from NIKIET's experience in design and development of marine propulsion PWRs 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.

Main NSSS design principles are construction costs reduction and reliability improvement. Construction costs reduction achieved via modularization – NPP equipment shipped in form of fabricated modules, which are fail-safe tested on specialized facilities before shipment.

Passive systems used in maximum extent in project to decrease probability of personnel errors and equipment malfunctions while increasing reliability of NSSS.

Other rather important principles are insuring manoeuvrability and self-sufficiency. NPP capable of supplying consumer in daily and annual load variations and does not require external measures during normal operation (between scheduled maintenances) and accidents (during 72 hours from the begging of accident)

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 two 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.

RPV and all necessary normal operation and safety systems assembled inside a cylindrical power capsule with 8 m in inner diameter and 10 m high. Capsule's shell serves as a containment. SHELF-M has optimized core with extended fuel cycle (up to 8 years) and higher thermal capacity (35 MW) compared to SHELF.

(b) Reactor Core

The SHELF-M reactor core consists of 241 cylindrical fuel assemblies (FA) of different types with variable fuel enrichment. Fuel assemblies placed in nodes of a regular hexagonal lattice. Control rods are located between FA and united in nine 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-aluminum (silumin) matrix. Fuel rod cladding 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 able to shut down the core in case of severe accident by injecting boron acid in primary coolant. Boron regulation not provided during normal operation.

(d) Reactor Vessel (RV) 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 lids (central and peripheral). All pipelines connected to the lids at the top of the RPV to prevent the core drainage in case of LOCA.

RPV and reactor internals made of low-cobalt steel and alloys to prevent generation of long-lasting radioactive waste.

(e) Reactor Coolant System and Steam Generator

Reactor coolant system based on conventional two-circuit methodology. The SHELF-M primary circuit cooling under normal operating conditions done using forced circulation by two primary coolant circulation pumps (PCCP). Cooling by natural circulation is also possible with thermal output at the 20 % level of thermal capacity.

The SHELF-M utilizes 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 shut off individual section in case of primary coolant leaks.

(f) Pressuriser

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 five separate vessels: gas collector, intermediate pressurizer, end pressurizer and two expansion vessels. Pressurizer connected to the reactor central lid by a pipeline with a restrictor to reduce coolant leak rate in large-break accidents.

(g) Primary pumps

The SHELF-M utilizes two electrical primary coolant circulation pumps (PCCP) located on peripheral lid of the RPV.

(h) Distinguishing features

Autonomous passive decay heat removal system, integral transportable containment

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The SHELF-M 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

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

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, loss of secondary circuit coolant etc.), emergency decay heat removal system (EDHRS) 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 standalone heat exchangers between circuits. The heat transferred to the atmospheric air via heat exchangers, located on the roof of the reactor building.

(e) Safety Approach and Configuration to Manage DEC

The emergency reactor cooling system 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 vessels with coolant under high pressure, which is passively injected into RPV in case of primary coolant pressure drop.

(f) Containment System

The SHELF-M utilizes two containment systems. The first one - safety vessel, located inside power capsule, which 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 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.

(g) Spent Fuel Cooling Safety Approach / System

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 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 45 days once in 8 years.

7. Instrumentation and Control Systems

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

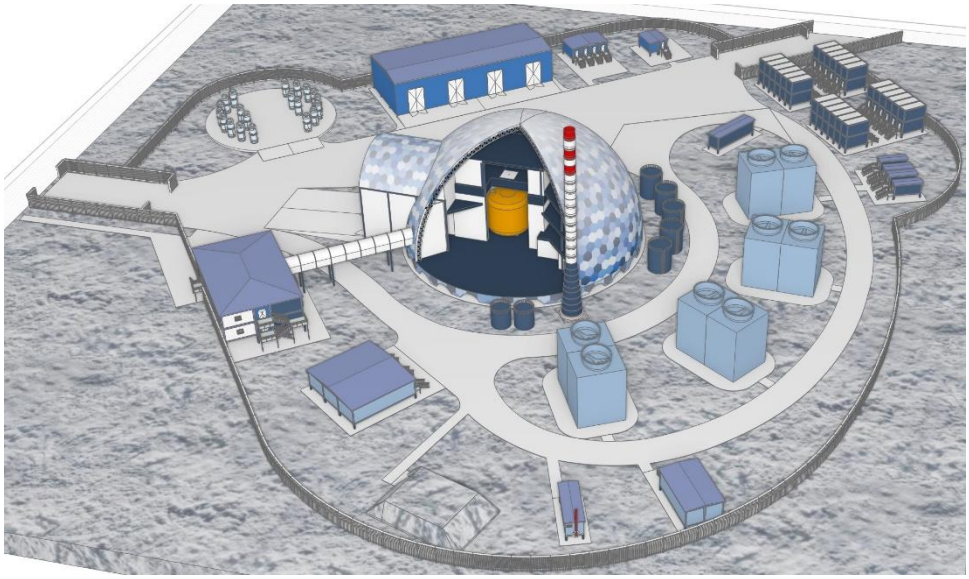
- Normal operation comprises of phased automated initiation, operation at steady power levels in a range of 20 to 100 % of rated power with forced primary coolant circulation, operation at steady power levels at 20 % of rated power with natural coolant circulation, switchover from natural primary coolant circulation to forced circulation and scheduled automated deactivation.
- 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.
- 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 based on lightweight steel frame structure and houses the reactor hall with power capsule, external concrete protection, and the turbine hall. The building 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 is to be started in 2027.

11. Fuel Cycle Approach

The SHELF-M implements a standard PWR fuel cycle. All fuel assemblies unload 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 designed to gather radioactive leaks from reactor cooling systems and storage liquid waste inside power capsule.

Gaseous radioactive waste system 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.

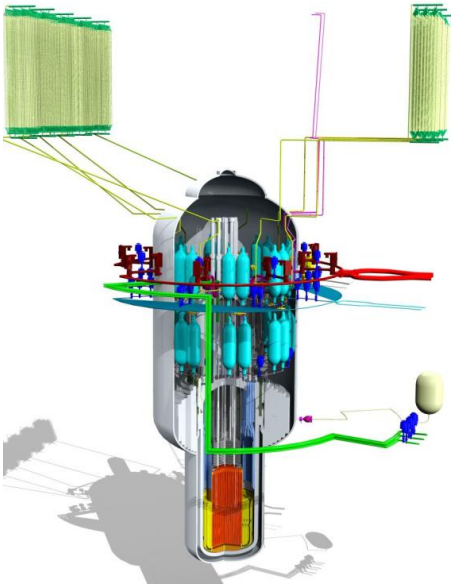
13. Development Milestones

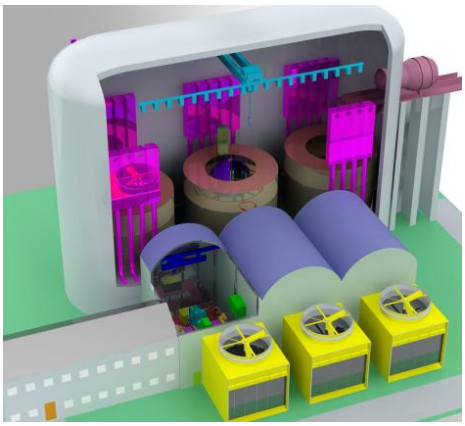
Year	Milestone	Status
2018	Start of the project SHELF-M	Completed
2019	Conceptual design	Completed
2022	State contract for the development of NPP with SHELF-M reactor unit	Completed
2022-2024	Detailed design	Ongoing
2027-2029	Licensing	Planned
2028	Start of construction	Planned
2030	Operation testing	Planned

NIKIET, Russian Federation
UNITHERM



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*NIKIET, Russian Federation*
Reactor type	*PWR*
Coolant/moderator	*High purity water*
Design status	*Conceptual design*
Thermal/electrical capacity, MW(t)/MW(e)	*30 / 6.6*
NSSS operating pressure (primary/secondary), MPa	*16.5*
Core inlet/outlet coolant temperature (°C)	*249 / 330*
Primary circulation	*Natural circulation*
Fuel type/assembly array	*UO2 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*
Fuel cycle requirements/approach	*Traditional*
Reactivity control mechanism	*Soluble boron and control rod insertion*
Approach to safety systems	*Hybrid (passive + active) system*
Seismic design (SSE)	*VIII-IX-MSK 64*
Design life (years)	*30*
Plant footprint (m^2)	*~10 000*
RV height/diameter (m)	*9.8 / 2.9*
RV weight (metric ton)	*32*

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

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☒ Desalination	☐ >550°C

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.

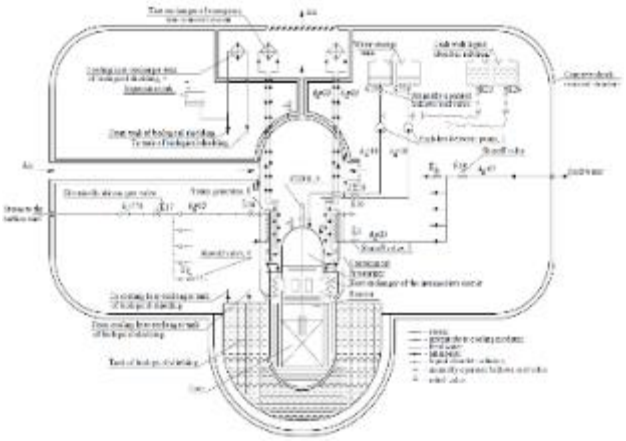


Figure 1: General view of the nuclear steam supply system and associated systems.

(d) Reactor Vessel (RV) 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 and Steam Generator

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).

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.

(f) Pressuriser

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

(g) Primary pumps

(h) Distinguishing features

Autonomous passive reactor decay removal system; guard vessel; iron-water biological shielding; and the biological shielding tanks

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) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

(e) Safety Approach and Configuration to Manage DEC

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.

(f) 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

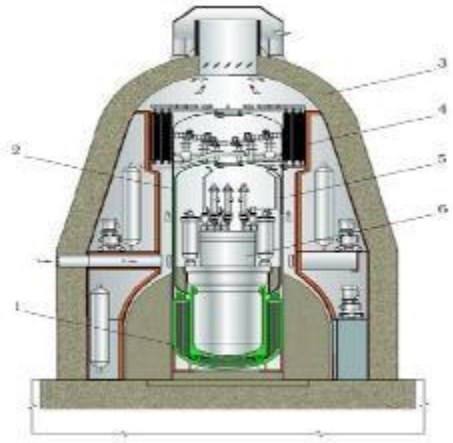


Figure 2: Containment system

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.

(g) Spent Fuel Cooling Safety Approach / System

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 κV ± 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

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. Testing Conducted for Design Verification and Validation

N/A

10. 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.

11. Fuel Cycle Approach

The duration of the campaign reactor core is 15 years.

12. 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.

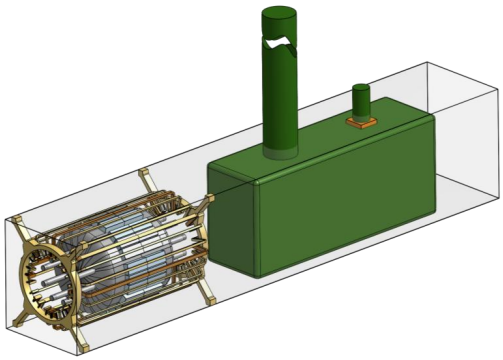
13. Development Milestones

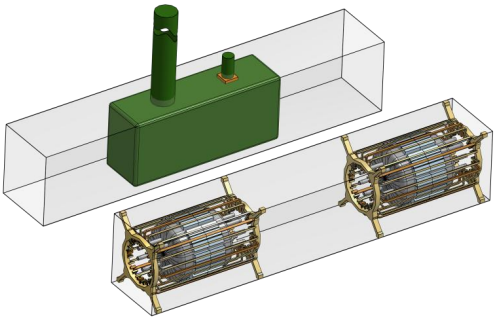
Year	Milestone	Status
1994	The NPP design on the basis of the UNITHERM concept has become the laureate of the competition on SMR designs established by the Russian Nuclear Society	Completed
2012	Technical proposal on the UNITHERM reactor facility (WDR stage)	Completed
2015	Technical proposal for a SMR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

Nano Nuclear Energy,, United States of America, ZEUS



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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*NANO Nuclear Energy*
Reactor type	*Solid core micro reactor*
Coolant/moderator	*None/Graphite*
Design status	*Conceptual Design*
Thermal/electrical capacity, MW(t)/MW(e)	*0.38/0.1*
NSSS operating pressure (primary/secondary), MPa	*No primary pressure / 0.8*
Core inlet/outlet coolant temperature (°C)	*N/A*
Primary circulation	*N/A*
Fuel type/assembly array	*UO2*
Number of fuel assemblies in the core	*N/A*
Fuel enrichment (%)	*≤ 10%*
Core discharge burnup (GWd/ton)	
Refuelling cycle (months)	
Fuel cycle requirements/approach	*240 (no on-site refuelling)*
Reactivity control mechanism	*once-through*
Approach to safety systems	*Control rods; burnable absorbers.*
Seismic design (SSE)	*Passive 72 h*
Design life (years)	
Plant footprint (m^2)	
RV height/diameter (m)	*20*
RV weight (metric ton)	*< 10000*

1. Introduction

NANO Nuclear's Zeus nuclear reactor is a 100 kWe Micro Modular Reactor (MMR). It features a unique design that does not rely on any primary fluid coolant, but instead, heat is transferred to the edge of the core by conduction through a solid moderator matrix. This way any risk associated with primary coolant deviations from nominal conditions (loss of coolant, loss of flow, boiling, etc.,) is eliminated. Furthermore, without a fluid coolant, all the related components (pump/blower, pipes, tanks, etc.) and related systems (chemistry and volume control) are eliminated, greatly simplifying the design and reducing the overall size. The second main design constraint and philosophy is that the overall system, including the reflector and the conversion system and excluding shielding, is designed to fit in a 45-foot High-Cube container (length 13.716 m, width 2.500 m, height 2.896 m).

Zeus is loaded with conventional uranium dioxide fuel, with low enrichment levels, and can operate at full power for 20 years without refueling.

Zeus' secondary loop relies on an open-air Brayton cycle for converting the thermal energy into electricity. Heat exchangers are placed outside of the core vessel on its cylindrical lateral surfaces keeping the primary system completely sealed. The

Zeus power unit is designed to generate 380 kWt to drive a single compressor-turbine engine. Alternatively, a single turbomachinery setup could be configured to serve two Zeus units.

2. Target Application

☒ Electricity	☐ Hydrogen	☐ Industrial process heat
☒ Baseload	☐HTSE - electrolysis	☐ <200°C
☒ Load-following	☐PEM - polymer membrane	☐ 200–550°C
☒ District Heating / Steam	☐ Desalination	☐ >550°C

This power plant is designed for easy transferability, making it ideal for remote areas needing electricity or heat where grid connection is not economically viable.

3. Design Philosophy

Design philosophy can be individuated into two main features:

- Eliminating primary fluid coolants and relying entirely on solid-state heat conduction. Safety is inherently passive, using pure conduction and radiation to safely dissipate decay heat without the need for moving parts or active interventions, even during severe accident scenarios.
- Modularity and portability are achieved by packaging the entire reactor and its open-air Brayton power conversion system to fit within a standard 45-foot High-Cube shipping container.

4. Main Design Features

(a) Nuclear Steam Supply System

This system is not applicable to the Zeus Nano Reactor, as its solid core design eliminates the need for any primary fluid coolant. Instead of a traditional steam supply system, it utilizes an open-air Brayton cycle in the secondary loop to convert thermal energy into electricity.

(b) Reactor Core

The core uses conventional uranium dioxide fuel with enrichment levels between 7.5% and 10.0%, arranged across 12 concentric rings within a solid graphite matrix. The active region measures 120 cm in height and 160 cm in diameter. Inner cylindrical graphite moderator of 40 cm in diameter. It targets a burnup of ~10 MWd/kgU, over a 20-year effective full power lifespan. Reactor thermal power of approximately 380 kWt, which yields the 100 kWe electrical output at a roughly 26% overall efficiency.

(c) Reactivity Control

Reactivity is mechanically controlled by cross-shaped control rods made of boron carbide. Each control rod is driven by individual bellows-based mechanisms. Additionally, to manage reactivity over the core's lifetime, burnable poison is included within the central graphite moderator block.

(d) Reactor Vessel (RV) and Internals

The reactor features a hermetically sealed cylindrical central housing equipped with lateral expansion bellows to accommodate differential thermal expansions. The internal arrangement includes graphite slabs separated by helium.

(e) Reactor Coolant System and Steam Generator

Traditional fluid coolant loops and steam generators are not applicable. Heat is transferred entirely through solid-state conduction to the vessel edge, where highly porous ceramic silicon carbide foam heat exchangers transfer the thermal energy directly to the secondary loop air. The highly porous silicon carbide foam heat exchangers rely on varying levels of porosity to optimize the airflow, combining a 30 pores-per-inch fine material near the inlet with an 8 pores-per-inch coarse material. The secondary cooling air enters this foam structure at roughly 300 oC and is heated to approximately 700 oC before expanding in the turbine.

(f) Pressuriser

A pressuriser is not applicable to the Zeus design. The total absence of a primary fluid coolant means that pressure, volume, and chemistry control systems are eliminated.

(g) Primary pumps

Primary pumps are not applicable for this reactor. Because there is no primary fluid to circulate, the design fundamentally eliminates all moving primary systems, including pumps, blowers, and associated piping.

(h) Distinguishing features

A unique aspect of this 100 kWe reactor is its reliance on fully passive, solid-state heat conduction. This completely eliminates the risks typically associated with primary fluid coolants, such as loss-of-coolant, loss of flow, or boiling.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

N/A

(b) Decay Heat Removal System

(c) Emergency Core Cooling System

(d) Safety Approach and Configuration to Manage DBC

N/A

(e) Safety Approach and Configuration to Manage DEC

N/A

(f) Containment System

N/A

(g) Spent Fuel Cooling Safety Approach / System

N/A

6. Plant Safety and Operational Performances

The core is engineered for long-term service, featuring a long core lifetime. It is targeted to operate continuously at full power for 20 years without refueling. The system is designed to produce reliable and stable baseload power, resulting in a limited load-following capability.

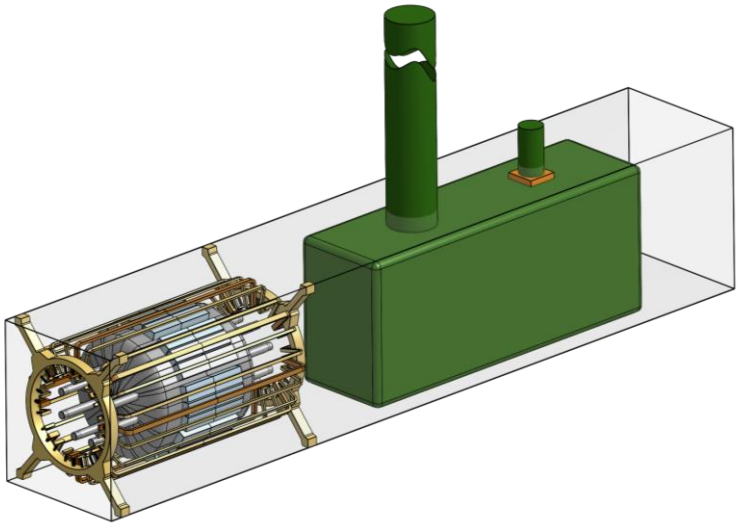
Safety is ensured by two redundant and independent sets of shutdown rods, which are designed for passive insertion upon loss of electricity. Furthermore, during a Loss of Heat Sink (LOHS) accident, the reactor passively dissipates 100% of its decay heat through thermal conduction, radiation, and natural air circulation driven by a chimney draft.

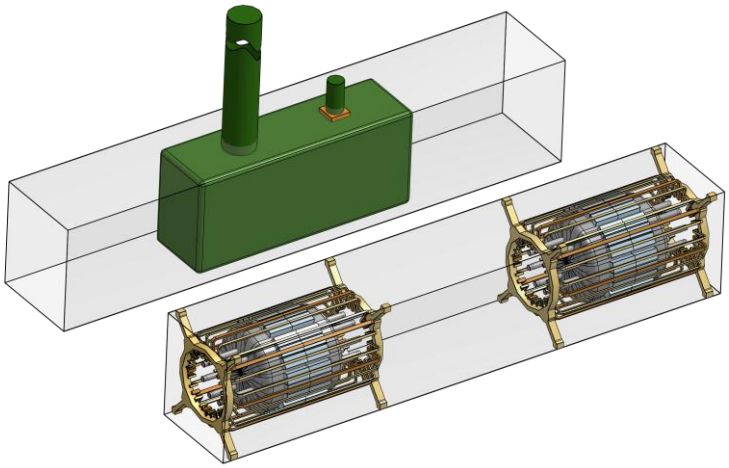
7. Instrumentation and Control Systems

N/A

8. Plant Layout Arrangement

The Zeus Reactor's power plant layout is designed entirely around the dimensions of a standard 45-foot High-Cube container. The plant can be deployed in two alternative configurations: a single-core arrangement that houses one reactor core and its associated turbomachinery assembly together within a single container, or a two-core arrangement. In the two-core setup, the reactors are placed inside one container to share a single turbomachinery unit; because each core has an approximate length of 4.2 meters, this dual-core configuration may require a second, separate container specifically for the transport of the energy conversion system.





9. Testing Conducted for Design Verification and Validation

Key tests to validate the feasibility and assess the conceptual design include the assessment of the moderator, specifically its thermal conduction performance as a function of temperature and fluence. Furthermore, the heat exchanger foam modules and the characteristics of the vessel expansion joint are also subject to main tests.

10. Design and Licensing Status

Current status is Conceptual Design. The design philosophy is fixed, preliminary optimization of materials, design solutions and strategies are assessed.

11. Fuel Cycle Approach

The Zeus reactor employs a single-batch fuel management strategy, designed to operate continuously for 20 effective full power years (EFPY) without any refueling. The core utilizes conventional, low-enriched uranium dioxide (UO2) fuel.

12. Waste Management and Disposal Plan

N/A

13. Development Milestones

N/A

ANNEX
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

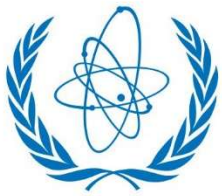
PLOFC Pressurized Loss of Forced Cooling

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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