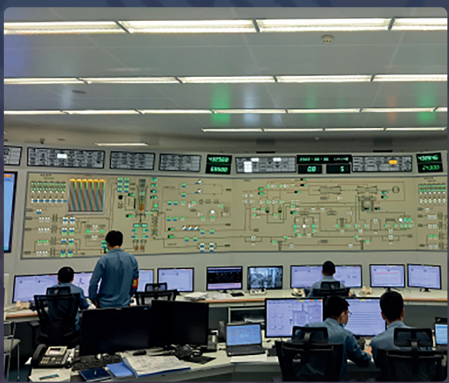
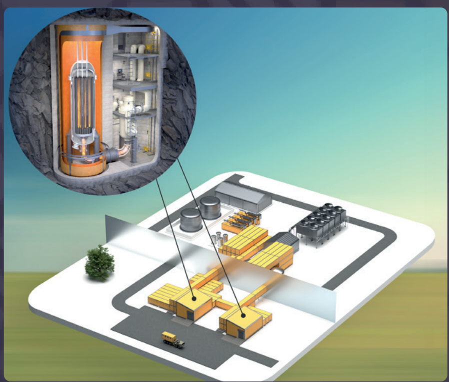
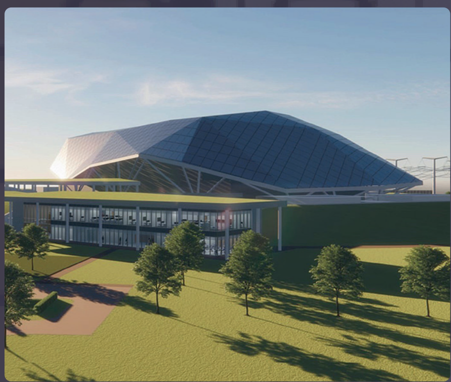
Advances in Small Modular Reactor Technology Developments

A Supplement to:
IAEA Advanced Reactors Information System (ARIS)
2022 Edition

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ADVANCES IN SMALL MODULAR REACTOR TECHNOLOGY DEVELOPMENTS

2022 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 expanding countries interested in SMRs. In this regard, balanced and objective information to all Member States on technology status and development trends for advanced reactor lines and their applications are collected, assessed and provided through publication of status reports and other technical documents.

Member States, both those launching their nuclear power programme and those with an existing nuclear power programme, keep expressing their interest in information about advances in SMR design and technology developments, as well as global trend of their deployment. The IAEA's Division of Nuclear Power, which has been facilitating Member States in addressing common technologies and issues for SMRs, plays a prominent role in convening international scientific forums and technical cooperation in this field for the interested Member States. The activities on SMRs are further supported by specific activities on advanced water-cooled, liquid metal-cooled fast neutron spectrum, molten salt, and high temperature gas-cooled reactors technology developments, as well as their non-electric applications.

The driving forces in the development of SMRs are their specific characteristics. They can be deployed incrementally to closely match increasing energy demand resulting in a moderate financial commitment for countries or regions with smaller electricity grids. SMRs show the prospect 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-electric applications such as district heating, providing heat for industrial processes, hydrogen production or sea-water desalination. Process heat or cogeneration results in significantly improved thermal efficiencies leading to a better return on investment. Some SMR designs may also serve niche markets, for example by deploying microreactors to replace diesel generators in small islands or remote regions.

Booklets on the status of SMR technology developments have been published biennially since 2012. The objective is to provide Member States with a concise overview of the latest status of SMR designs. This booklet is reporting the advances in design and technology developments of SMRs of all the major technology lines within the category of SMRs. It covers land-based and marine-based water-cooled reactors, high temperature gas-cooled reactors, liquid metal-cooled fast neutron spectrum reactors, molten salt reactors, and a sub-category called microreactors with electrical power typically up to 10 MW(e). For the first time also that the booklet provides some insights on the economic challenges in deployment of SMRs, a summary on enabling design features to facilitate SMRs' decommissioning and a summary on experimental testing for design verification and validation. The brief design description of SMRs is provided by the responsible institute or organization and is reproduced, with permission, in this booklet.
This booklet is intended as a supplement to the IAEA Advanced Reactor Information System (ARIS), which can be accessed at http://aris.iaea.org. Previous editions of this booklet published in support of ARIS are listed in Annex X.

This publication was developed by Nuclear Power Technology Development Section, Division of Nuclear Power of the IAEA's Department of Nuclear Energy in cooperation with Member States. The IAEA officers responsible for this publication were Y. Zou and M.H. Subki of the Division of Nuclear Power.

TABLE OF CONTENTS

INTRODUCTION 1
PART I.1. WATER COOLED SMALL MODULAR REACTORS (LAND BASED) 9
CAREM (CNEA, Argentina) 11
ACP100 (CNNC,China) 15
CANDU SMR™ (Candu Energy Inc, Canada) 19
CAP200 (SPIC/SNERDI, China) 23
DHR400 (CNNC, China) 27
HAPPY200 (SPIC, China) 31
NHR200-II (Tsinghua University and CGN, China) 35
TEPLATOR™ (UWB Pilsen & CIIRC CTU, Czech Republic) 39
NUWARD™ (EDF, France) 43
IMR (Mitsubishi Heavy Industries, Japan) 47
i-SMR (KHNP & KAERI, Republic of Korea) 51
SMART (KAERI, Republic of Korea and K.A.CARE, Saudi Arabia) 55
RITM-200N (JSC "Afrikantov OKBM", Russian Federation) 59
VK-300 (NIKIET, Russian Federation) 63
KARAT-45 (NIKIET, Russian Federation) 67
KARAT-100 (NIKIET, Russian Federation) 71
RUTA-70 (NIKIET, Russian Federation) 75
STAR (STAR ENERGY SA, Switzerland) 79
Rolls-Royce SMR (Rolls-Royce SMR Ltd, United Kingdom) 83
VOYGR™ (NuScale Power Corporation, United States of America) 87
BWRX-300 (GE-Hitachi Nuclear Energy, USA and Hitachi-GE Nuclear Energy, Japan) 91
SMR-160 (Holtec International, United States of America) 95
Westinghouse SMR (Westinghouse Electric Company LLC, United States of America) 99
mPower (BWX Technologies, Inc., United States of America) 103
OPEN20 (Last Energy Inc., United States of America) 107
PART I.2. WATER COOLED SMALL MODULAR REACTORS (MARINE BASED) 111
KLT-40S (JSC "Afrikantov OKBM", Russian Federation) 113
ACPR50S (CGNPC, China) 117
ACP100S (CNNC, China) 121
BANDI-60 (KEPCO E&C, Republic of Korea) 125
ABV-6E (JSC "Afrikantov OKBM", Russian Federation) 129
RITM-200M (JSC "Afrikantov OKBM", Russian Federation) 133
VBER-300 (JSC "Afrikantov OKBM", Russian Federation) 137
SHELF-M (NIKIET, Russian Federation) 141
PART II. HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS 145

HTR-PM (Tsinghua University, China) 147
STARCORE (StarCore Nuclear, Canada, United Kingdom and United States of America) 151
JIMMY (JIMMY ENERGY SAS, France) 155
GTHTR300 (JAEA Consortium, Japan) 159
GT-MHR (JSC “Afrikantov OKBM”, Russian Federation) 163
MHR-T Reactor (JSC “Afrikantov OKBM”, Russian Federation) 167
MHR-100 (JSC “Afrikantov OKBM”, Russian Federation) 171
AHTR-100 (Eskom Holdings SOC Ltd., South Africa) 177
PBMR®-400 (PBMR SOC Ltd., South Africa) 181
HTMR100 (STL Nuclear (Pty) Ltd., South Africa) 185
EM^2 (General Atomics, United States of America) 189
FMR (General Atomics, United States of America) 193
Xe-100 (X-energy, LLC, United States of America) 197
SC-HTGR (Framatome Inc., United States of America) 201
PeLUIt/RDE (BRIN, Indonesia) 205
HTR-10 (Tsinghua University, China) 209
HTTR (JAEA, Japan) 213
PART III. LIQUID METAL COOLED FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS 217
BREST-OD-300 (NIKIET, Russian Federation) 219
ARC-100 (ARC Clean Energy, Canada) 223
4S (Toshiba Energy Systems & Solutions Corporation, Japan) 227
MicroURANUS (UNIST, Republic of Korea) 231
LFR-AS-200 (*new*cleo srl, Italy) 235
SVBR (JSC AKME Engineering, Russian Federation) 239
SEALER-55 (LeadCold, Sweden) 243
Westinghouse Lead Fast Reactor (Westinghouse Electric Company, USA) 247
PART IV. MOLTEN SALT SMALL MODULAR REACTORS 251
IMSR400 (Terrestrial Energy Inc., Canada) 253
SSR-W (Moltex Energy, Canada) 257
smTMSR-400 (SINAP, CAS, China) 261
CMSR (Seaborg Technologies, Denmark) 265
Copenhagen Atomics Waste Burner (Copenhagen Atomics, Denmark) 269
FUJI (International Thorium Molten-Salt Forum, Japan) 273
THORIZON (THORIZON B.V., Netherlands) 277
SSR-U (Moltex Energy, United Kingdom) 281
KP-FHR (Kairos Power, United States of America) 285
Mk1 PB-FHR (UC Berkeley, United States of America) 289
MCSFR (Elysium Industries, USA) 293

LFTR (Flibe Energy, United States of America) 297
ThorCon (ThorCon International, United States of America and Indonesia) 301
PART V. MICROREACTORS 305
Energy Well (Centrum výzkumu Řež s.r.o., Czech Republic) 307
MoveluX (Toshiba Energy Systems & Solutions Corporation, Japan) 311
ELENA (NRC "Kurchatov Institute", Russian Federation) 315
UNITHERM (NIKIET, Russian Federation) 319
AMR (STL Nuclear (Pty) Ltd, South Africa) 323
LFR-TL-30 (*new*cleo Ltd., United Kingdom) 327
U-Battery (Urenco, United Kingdom) 331
Aurora Powerhouse Product Line (Oklo Inc., United States of America) 335
HOLOS-QUAD (HolosGen LLC, United States of America) 337
MARVEL Research Microreactor (Idaho National Laboratory, United States of America) 341
MMR® (Ultra Safe Nuclear Corporation, United States of America) 345
Westinghouse eVinci™ Micro Reactor (Westinghouse Electric Company LLC, USA) 349
ANNEX I. Summary of Global SMR Technology Development and Deployment 353
ANNEX II. Power Range of SMR Designs 358
ANNEX III. Comparison of Main Technical Characteristics among several SMR Designs 364
ANNEX IV. Non-Electric Applications using Small Modular Reactors 370
ANNEX V. Economic Challenges in Deployment of Small Modular Reactors 374
ANNEX VI. Fuel Cycle Approach adopted in SMRs 378
ANNEX VII. Spent Fuel, Waste Management and Disposal Plans adopted for SMRs 385
ANNEX VIII. Enabling Design Features to Facilitate SMR Decommissioning 390
ANNEX IX. Experimental Testing for Design Verification and Validation 394
ANNEX X. Bibliography 401
ANNEX XI. Acronyms 405
ANNEX XII. Acknowledgement 411

INTRODUCTION

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

Climate change presents us with a stark challenge, that is to reduce greenhouse gas emissions much faster than has been done so far or face the increasingly catastrophic consequences of an inexorably warming planet. The world needs to harness all low-carbon sources of energy to meet the Paris Agreement goal of limiting the rise in global temperatures to well below 2°C above pre-industrial levels. Use of renewables such as wind and solar power is growing, while nuclear power continues to make a significant contribution to energy supply, energy security and grid stability. Of all the low-carbon energy sources, nuclear power is one of the few, if not the only one, that can generate at scale all the main energy carriers: electricity, heat, and hydrogen. Advanced nuclear reactor technologies, particularly Small Modular Reactors (SMRs) including microreactors, are designed to not only produce base load and dispatchable carbon-free electricity but also to supply other clean energy products needed to decarbonize energy-intensive sectors such as the transport sector, the building sector, industrial heat applications, and desalination. SMRs are also designed with enhanced operability to enable flexible operation with variable renewables.

There is increasing interest in SMRs and their applications. During the International Conference on Climate Change and the Role of Nuclear Power held in September 2019, SMRs were considered by many Member States as a potential viable nuclear option to contribute in mitigating the climate change. Following on this conference, in April 2021, the IAEA launched the IAEA Platform on Small Modular Reactors and their Applications (the SMR Platform), a mechanism that coordinates the IAEA's activities in this field and provides a 'one-stop shop' for Member States and other stakeholders. The SMR Platform offers expertise from the entire Agency, encompassing all aspects relevant to the development, early deployment, and oversight of SMRs and their applications.

SMRs are advanced reactors with a power capacity of typically up to 300 MW(e) per unit, which is about one-third of the generating capacity of traditional nuclear power reactors and whose components and systems can be shop fabricated and then transported as modules to the sites for installation as demand arises. Most of the SMR designs adopt advanced safety features and are deployable either as a single or multi-module plant. SMRs are under development for all principal reactor technology lines: water-cooled reactors, high temperature gas-cooled reactors, liquid metal-cooled fast neutron spectrum reactors, molten salt reactors, and microreactors. The key driving forces of SMR development are fulfilling the need for flexible power generation for a wider range of users and applications, replacing ageing fossil-fired units, enhancing safety performance, and offering better economic affordability.

SMRs are envisioned for niche electricity or energy markets where large reactors would not be viable. SMRs could provide cogeneration for heat, hydrogen production, desalination for small electricity grids, remote and off-grid areas, and enabling hybrid nuclear-renewables energy systems. Through modularisation technology, SMRs target at the economies of serial production with shorter installation or construction time. Near-term deployable SMRs are expected to have safety performance comparable or better to that of evolutionary reactor designs.

With smaller footprints, SMRs are expected to have flexibility in siting and allow them to be tailored to the energy needs of regional or industrial clusters. The modularity and advanced safety features make SMRs attractive to industries and countries with smaller grid sizes and little to no expertise of operating nuclear plants. Some transportable turnkey systems are being developed and deployed that are completely built in a shipyard factory, delivered to remote sites, or exported to other countries as a marine plant for plug and play for electricity and heat supply. The possibility of siting SMRs in remote, off-grid communities enhance the access of such community to clean electricity and heat. These advantages relate well with the floating power plants concepts based on SMRs.

Though significant advancements have been made in various SMR technologies in recent years, some technical issues still attract considerable attention in the industry. These include control room staffing and human factor engineering for multi-module SMR plants, applicability of existing codes and standards, manufacturing approach for novel components, and back-end solutions for fuel cycle. Some potential deployment advantages of SMRs such as reduced size of Emergency Planning Zone (EPZ) or single operator monitoring several modules are under discussion by nuclear regulators. Although SMRs are designed for lower upfront capital cost per unit, their economic competitiveness is still to be proven.

There are more than eighty (80) SMR designs under development and deployment at different stages in 18 Member States. Several major milestones have been reached in SMR technology deployment. The Akademik Lomonosov floating power unit in the Russian Federation with two-module KLT-40S was connected to the grid in December 2019 and started commercial operation in May 2020. The HTR-PM demonstrator in China was connected to the grid in December 2021 and is expected to reach full power operation by the end of 2022. The CAREM25 in Argentina is under construction and is expected to reach first criticality in 2026. The construction of ACP100 in China started in July 2021 and is targeted to start commercial operation by the end of 2026. The construction of BREST-OD-300 in Russian Federation began in June 2021 and is planned to be completed in 2026. The NuScale Power Module™ in the United States has received Standard Design Approval from U.S. NRC in September 2020. The NRC has directed to issue a final rule that certifies NuScale's SMR design for use in the United States.

This booklet provides a brief introductory information and technical descriptions of various SMR designs and technologies under different stages of development and deployment. To assist the readers in better understanding the status of development and deployment, **Table 1** lists all the eighty-three (83) designs included in this booklet, along with the output capacity, reactor type, designer organisation, country of origin and status of development or deployment.

The 2022 edition comprises five (5) parts. Parts I to IV are arranged in the order of the different types of coolants, with Part I divided into two sub-parts (land based and marine based). Part V is dedicated to microreactors, a sub-category of SMR which either has a relatively smaller output capacity (typically less than 10 MWe) or has specific applications, such as powering remote regions. The part of microreactors includes technologies that use different coolants and/or fuel design, such as heat pipe.

Table 1 Design and Status of SMRs included in this Booklet

Design	Output MW(e)	Type	Designer	Country	Status
PART I.1: WATER COOLED SMALL MODULAR REACTORS (LAND BASED)					
CAREM	30	Integral PWR	CNEA	Argentina	Under construction
ACP100	125	Integral PWR	CNNC/NPIC	China	Under construction
CANDU SMR™	300	PHWR	Candu Energy Inc.	Canada	Conceptual Design
CAP200	> 200	PWR	SPIC/SNERDI	China	Basic Design
DHR400	400 MW(t)	PWR (pool type)	CNNC	China	Basic Design
HAPPY200	200 MW(t)	PWR	SPIC	China	Detailed Design
NHR200-II	200 MW(t)	Integral PWR	Tsinghua University and CGN	China	Basic Design
TEPLATOR™	< 150 MW(t)	HWR	UWB Pilsen & CIIRC CTU	Czech Republic	Conceptual Design
NUWARD™	2 × 170	Integral PWR	EDF	France	Conceptual Design
IMR	350	PWR	MHI	Japan	Conceptual Design Completed
i-SMR	170	Integral PWR	KHNP and KAERI	Republic of Korea	Conceptual design
SMART	107	Integral PWR	KAERI and K.A.CARE	Republic of Korea and Saudi Arabia	Detailed Design
RITM-200N	55	Integral PWR	JSC Afrikantov OKBM, Rosatom	Russian Federation	Detailed Design Completed
VK-300	250	BWR	NIKIET	Russian Federation	Detailed Design
KARAT-45	45 – 50	BWR	NIKIET	Russian Federation	Conceptual Design
KARAT-100	100	BWR	NIKIET	Russian Federation	Conceptual Design

Design	Output MW(e)	Type	Designer	Country	Status
RUTA-70	70 MW(t)	PWR (pool type)	NIKIET	Russian Federation	Conceptual Design
STAR	10	LWR (pressure tube)	STAR ENERGY SA	Switzerland	Basic design
Rolls-Royce SMR	470	PWR	Rolls-Royce SMR Ltd.	UK	Detailed Design
VOYGR™	4/6/12 × 77	Integral PWR	NuScale Power Corporation	USA	Equipment Manufacturing in progress
BWRX-300	270 – 290	BWR	GE-Hitachi Nuclear Energy and Hitachi-GE Nuclear Energy	USA and Japan	Detailed Design
SMR-160	160	PWR	Holtec International	USA	Preliminary Design Completed
Westinghouse SMR	> 225	Integral PWR	Westinghouse Electric Company LLC	USA	Conceptual Design Completed
mPower	2 × 195	Integral PWR	BWX Technologies, Inc	USA	Conceptual Design
OPEN20	22	PWR	Last Energy Inc.	USA	Detailed Design
PART I.2: WATER COOLED SMALL MODULAR REACTORS (MARINE BASED)					
KLT-40S	2 × 35	PWR	JSC Afrikantov OKBM	Russian Federation	In Operation
ACPR50S	50	PWR (loop type)	CGNPC	China	Detailed Design
ACP100S	125	Integral PWR	CNNC/NPIC	China	Basic Design
BANDI-60	60	PWR	KEPCO E&C	Republic of Korea	Conceptual Design
ABV-6E	6 – 9	PWR	JSC Afrikantov OKBM, Rosatom	Russian Federation	Final design
RITM-200M	50	Integral PWR	JSC Afrikantov OKBM, Rosatom	Russian Federation	Basic Design Completed
VBER-300	325	Integral PWR	JSC Afrikantov OKBM, Rosatom	Russian Federation	Licensing Stage
SHELF-M	up to 10	Integral PWR	NIKIET	Russian Federation	Basic Design
PART II: HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS					
HTR-PM	210	HTGR (pebble bed)	INET, Tsinghua University	China	In operation
STARCORE	14/20/60	HTGR (prismatic)	StarCore Nuclear	Canada	Pre-Conceptual Design
JIMMY	10 – 20 MW(t)	HTGR (prismatic)	JIMMY ENERGY SAS	France	Detailed Design
GTHTR300	100 – 300	HTGR (prismatic)	JAEA Consortium	Japan	Basic Design
GT-MHR	288	HTGR (prismatic)	JSC Afrikantov OKBM	Russian Federation	Preliminary Design Completed
MHR-T	4 × 205.5	HTGR	JSC Afrikantov OKBM	Russian Federation	Conceptual Design
MHR-100	25 – 87	HTGR	JSC Afrikantov OKBM	Russian Federation	Conceptual Design
AHTR-100	50	HTGR (pebble bed)	Eskom Holdings SOC Ltd.	South Africa	Conceptual Design Completed
PBMR-400	165	HTGR (pebble bed)	PBMR SOC Ltd.	South Africa	Preliminary Design Completed
HTMR100	35	HTGR (pebble bed)	STL Nuclear (Pty) Ltd.	South Africa	Basic Design
EM²	265	GFR	General Atomics	USA	Conceptual Design
FMR	50	GFR	General Atomics	USA	Conceptual Design
Xe-100	82.5	HTGR (pebble bed)	X-Energy LLC	USA	Basic Design
SC-HTGR	272	HTGR (prismatic)	Framatome, Inc.	USA	Preliminary Design
PeLUIt / RDE	40 MW(t)	HTGR (pebble bed)	BRIN	Indonesia	Conceptual Design
HTR-10	2.5	HTGR (pebble bed)	INET, Tsinghua University	China	Operable

Design	Output MW(e)	Type	Designer	Country	Status
HTTR	30 MW(t)	HTGR (prismatic)	JAEA	Japan	In operation
PART III: LIQUID METAL COOLED FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS					
BREST-OD-300	300	LMFR (pool type)	NIKIET	Russian Federation	Under Construction
ARC-100	100	LMFR (pool type)	ARC Clean Energy	Canada	Preliminary Design
4S	10	LMFR (pool type)	Toshiba Energy Systems & Solutions Corporation	Japan	Detailed Design
MicroURANUS	20	LBE-cooled Reactor	UNIST	Republic of Korea	Conceptual Design
LFR-AS-200	200	LMFR	*new*cleo srl	Italy	Conceptual Design
SVBR	100	LMFR	JSC AKME Engineering	Russian Federation	Detailed Design
SEALER-55	55	LMFR	LeadCold	Sweden	Conceptual Design
Westinghouse LFR	450	LMFR (pool type)	Westinghouse Electric Company, LLC.	USA	Conceptual Design
PART IV: MOLTEN SALT SMALL MODULAR REACTORS					
IMSR400	2 × 195	MSR	Terrestrial Energy Inc.	Canada	Detailed Design
SSR-W	300	MSR (static fuelled)	Moltex Energy	Canada	Conceptual Design
smTMSR-400	168	MSR	CAS/SINAP	China	Pre-Conceptual Design
CMSR	100	MSR	Seaborg Technologies ApS	Denmark	Conceptual Design
Copenhagen Atomics Waste Burner	20 MW(t)	MSR	Copenhagen Atomics	Denmark	Detailed Design
FUJI	200	MSR	ITMSF	Japan	Preliminary Design Completed
THORIZON	40 – 120	MSR	THORIZON	Netherlands	Conceptual Design
SSR-U	16	MSR	Moltex Energy	UK	Basic Design
KP-FHR	140	FHR	KAIROS Power, LLC.	USA	Conceptual Design
Mk1 PB-FHR	100	FHR	UC Berkeley	USA	Pre-Conceptual Design
MCSFR	50 / 200 / 400 / 1200	MSR (fast spectrum)	Elysium Industries	USA	Conceptual Design
LFTR	250	MSR	Flibe Energy, Inc.	USA	Conceptual Design
ThorCon	250	MSR	ThorCon International	USA and Indonesia	Preliminary Design Completed
PART V: MICROREACTORS					
Energy Well	8	FHTR	Centrum výzkumu Řež	Czech Republic	Pre-Conceptual Design
MoveluX	3 – 4	Heat Pipe (sodium)	Toshiba Energy Systems & Solutions Corporation	Japan	Conceptual Design
ELENA	0.068	PWR	National Research Centre “Kurchatov Institute”	Russian Federation	Conceptual Design
UNITHERM	6.6	PWR	NIKIET	Russian Federation	Conceptual Design
AMR	3	HTGR (prismatic)	STL Nuclear (Pty) Ltd.	South Africa	Pre-conceptual design
LFR-TL-30	30	LMFR	*new*cleo Ltd.	UK	Conceptual Design
U-Battery	4	HTGR	Urenco	UK	Conceptual Design
Aurora	1.5 – 50	LMFR	OKLO, Inc.	USA	Detailed Design
HOLOS-QUAD	10	HTGR	HolosGen LLC	USA	Detailed Design
MARVEL	0.015 – 0.027	LMFR	Idaho National Laboratory	USA	Equipment manufacturing in progress
MMR™	> 5 and > 10	HTGR	Ultra Safe Nuclear Corporation	USA	Basic Design
Westinghouse eVinci™	2 – 3.5	Heat Pipe	Westinghouse Electric Company, LLC.	USA	Conceptual Design Completed

Part I.1: Land-based water-cooled SMRs. This part presents notable water-cooled SMR designs from various configurations of Light Water Reactor (LWR) and Heavy Water Reactor (HWR) technologies. These designs make advantage of mature technology as most of the large power plants in operation consist of water-cooled reactors. There are twenty-five (25) land-based water-cooled SMR designs from twelve (12) Member States described in this booklet. These designs include integral PWR, PHWR, compact PWR, loop-type PWR, BWRs, and pool-type PWR for district heating. An integral PWR with natural circulation, CAREM, is under construction for first criticality in 2026. Another integral PWR with design simplification, ACP100, started construction in 2021 and is targeting commercial operation in 2026. More designs are being prepared for near-term deployment, including NuScale VOYGR and BWRX-300 in the USA, Rolls-Royce SMR in the UK and NUWARD in France.

Part I.2: Marine-based water-cooled SMRs. This part presents concepts that can be deployed in a marine environment, either as barge-mounted floating power unit or immersible underwater power unit. This sub-category provides many flexible deployment options. There are eight (8) marine-based water-cooled SMRs. Some of them have been deployed in nuclear icebreaker ship. KLT-40S, deployed in the Akademik Lomonosov floating nuclear power plant which started commercial operation in Pevek of Russian Federation in May 2020, is from this sub-category and is the first SMR design connected to the grid.

Part II: High temperature gas-cooled SMRs: This part provides information on modular HTGRs under development and in operation. HTGRs provide high temperature heat (≥750°C) that can be utilised for more efficient electricity generation, a variety of industrial applications as well as cogeneration. Fourteen (14) HTGR-type SMRs are described in this booklet, including China's HTR-PM, which was connected to grid in December 2021 and targets at full power operation by the end of 2022. This part also includes three (3) HTGR test reactors, two of which have already been in operation for technology testing purposes respectively in Japan and China for over twenty years.

Part III: Liquid metal-cooled fast neutron spectrum SMRs. This part presents eight (8) SMR designs that adopt fast neutron spectrum with liquid metal coolants, including sodium, pure lead and lead-bismuth eutectic. Tangible advances in technology development and deployment on SMRs in this category have been made. The BREST-OD-300, a lead-cooled fast neutron reactor, is in the process of construction which is planned to be completed in 2026 at a site in Seversk, Russian Federation. This is a demonstration prototype project for future design with large power to enable a closed nuclear fuel cycle.

Part IV: Molten salt SMRs. This part highlights thirteen (13) SMR designs from molten salt fuelled and cooled advanced reactor technology, which is also one of the six Generation IV designs. MSRs promise many advantages including enhanced safety due to inherent property of molten salt, low-pressure single-phase coolant system that eliminates the need of large containment, a high temperature system that results in high efficiency, and flexible fuel cycle. Several MSR designs are conducting preliminary licensing activities or preliminary engagement with regulators in Canada, Denmark, Netherlands, the UK and the USA.

Part V: Microreactors. This booklet carries on 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). Different types of coolant, including light water, helium, molten salt and liquid metal, are adopted by microreactors. Heat pipe is another proposed cooling system option. Several designs are undertaking licensing activities in Canada and the USA for near-term deployment. In 2019, a site initial application was submitted by Global First Power (GFP) to Canadian Nuclear Safety Commission (CNSC) for a single small modular reactor using USNC's Micro Modular Reactor (MMR) technology at Chalk River. Microreactors serve future niche electricity and heat markets, such as powering micro-grids and remote off-grid areas, restoring power quickly in communities affected by natural disasters and supporting faster restoration of critical services (e.g., hospitals, water supply) and seawater desalination. Twelve (12) microreactor designs are included and discussed in this booklet.

For this booklet, an effort has been made to present all SMR designs within the above categories. Each technical description consists of an introduction, design philosophy, target applications, a table of major design parameters, main design features, safety features, plant safety and operational performances, instrumentation and control system, plant layout arrangement, design and licensing status, fuel cycle approach, waste management and disposal plan, and development milestones. For the first time in this biennial booklet, many designs also provide information on testing conducted for design verification and validation. Even though some SMR designs don't present reportable new milestones or advances, they are still included in the booklet for the benefit of Member States' readers by providing lessons learned in design development. Not all reactor designs presented in this booklet can be strictly categorised as SMRs. Some strongly rely on proven technologies of operating large capacity reactors, while others do not apply modular design. They are presented in this booklet by reason of completeness and foreseen niche markets.

This booklet is enriched with a set of annexes that provide readers with various figures and tables to understand the essential technical aspects of SMR designs. The description of each annex is given below. The concepts that have been mentioned in these annexes are just examples useful to elaborate on possible options. The summaries provided have not undergone an official review by the IAEA. The views expressed do not necessarily reflect those of the IAEA or its Member States but remain the responsibilities of the contributors.

Annex I provides figures that summarise global SMR technology development and deployment. It includes a world map of SMR developers, general deployment timeline that extends to early 2030s, government and private sectors on SMR technology development, SMR designs across the world's regions, and a chart that presents stage of design or deployment of SMRs in terms of their output capacity.

Annex II provides figures showing power ranges of SMR designs of above-mentioned five (5) categories.

Annex III provides tables comparing main technical characteristics among several SMR designs of the same category.

Annex IV recaps SMR designs for different non-electric applications. Typical reactor core exit coolant temperature of different reactor types is illustrated alongside the temperature of various non-electric applications.

Annex V provides a summary on the economic challenges in deployment of SMRs. It provides an overview of the economic challenges facing SMR during the key phases of a typical project – from the project development phase to construction and commissioning to the operation phase – with a focus on costs and cost drivers, funding and financing issues, and economic impact.

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

Annex VII provides a summary of spent fuel, waste management and disposal plans adopted for SMRs. It contains a table on the adopted approaches, i.e., volume reduction and conditioning, waste processing, storage approach, spent fuel pool cooling mechanism, spent fuel take back option, and market potential.

Annex VIII provides a summary on enabling design features to facilitate SMRs' decommissioning. It introduces the decommissioning aspect of SMR designs. It shares lessons learned and aspects related to SMR designs. Besides, a general discussion on SMRs for decommissioning presents interconnected factors in this regard.

Annex IX provides a summary on experimental testing for design verification and validation. The advanced design features of SMRs of major technology lines are verified by a comprehensive technology validation program that includes safety tests and performance tests. This annex features a table that lists experimental testing performed for validation of some integral PWR designs.

Annex X summarises other booklets previously published in support of the IAEA's Advanced Reactor Information System (ARIS, http://aris.iaea.org).

Annex XI contains a list of commonly used acronyms.

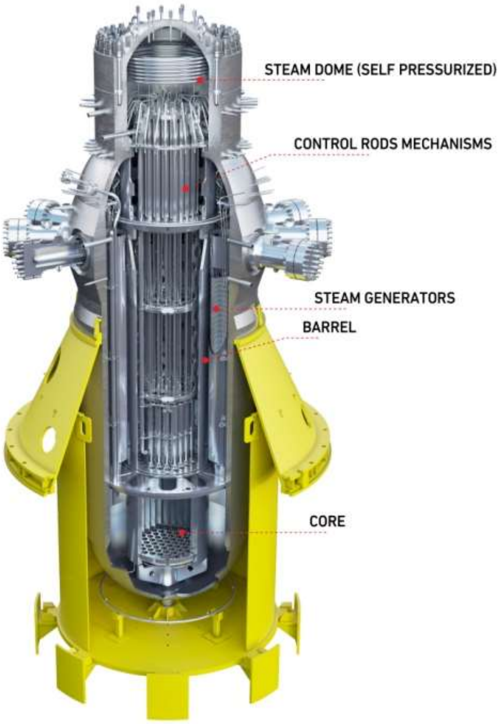
Annex XII refers to the acknowledgement to the organisations which provided design information and the organisations which helped develop this edition; and to the IAEA colleagues who provided inputs to the annexes or reviewed the contents.

This booklet is a supplement to the IAEA's ARIS which provides more detailed information on different SMRs under development and deployment. The technical description and major technical parameters were provided by the design organisations without validation or verification by the IAEA. All figures, illustrations and tables in technical description of each design were also provided by the design organisations.

It is hoped that this booklet will be useful to the Member States with a general interest in SMRs, as well as to the countries looking for structured technical information of one or more specific SMR designs. It should also furtherly promote contributions to and the use of the IAEA's ARIS.

PART I.1. WATER COOLED SMALL MODULAR REACTORS (LAND BASED)

CARREM (CNEA, Argentina)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNEA, Argentina
Reactor type	Integral PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	100 / ~30 (CAREM25)
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	12.25 / 4.7 (CAREM25)
Core Inlet/Outlet Coolant Temperature (°C)	284 / 326 (CAREM25)
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	61 (CAREM25)
Fuel enrichment (%)	3.1% (CAREM25)
Core Discharge Burnup (GWd/ton)	24 (CAREM25)
Refuelling Cycle (months)	14 (CAREM25)
Reactivity control mechanism	Control rod driving mechanism (CRDM) only
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	36 000 (CAREM25)
RPV height/diameter (m)	11 / 3.2 (CAREM25)
RPV weight (metric ton)	267 (CAREM25)
Seismic Design (SSE)	0.25g (CAREM25)
Fuel Cycle Requirements or Approach	390 full-power days and 50% of core replacement (CAREM25)
Distinguishing features	Core heat removal by natural circulation, pressure suppression containment
Design status	Under construction (CAREM25)

1. Introduction

CAREM is a national SMR development project, based on LWR technology, coordinated by Argentina's National Atomic Energy Commission (CNEA) in collaboration with leading nuclear companies in Argentina with the purpose to develop, design and construct innovative small nuclear power plants with high level of safety and economic competitiveness. CAREM is an integral PWR type NPP, based on indirect steam cycle with features that simplify the design and support the objective of achieving a higher level of safety.
CAREM25 is the demonstration prototype of CAREM SMR, and was developed using domestic technology, at least 70% of the components and related services for CAREM were sourced from Argentinean companies.

2. Target Application

CAREM is designed as an energy source for electricity supply of regions with small demands. It can also support seawater desalination processes to supply water and energy to coastal sites.

3. Design Philosophy

CAREM is a natural circulation based indirect-cycle reactor with features that simplify the design and improve safety performance. Its primary circuit is fully contained in the reactor vessel and it does not need any primary recirculation pumps. The self-pressurization is achieved by balancing steam production and condensation in the vessel, without a separate pressurizer vessel. CAREM design reduces the number of sensitive components and potentially risky interactions with the environment. Some of the significant design characteristics are: Integrated primary cooling system; Self-pressurized; Core cooling by natural circulation; In-vessel control rod drive mechanisms; and Safety systems relying on passive features.

4. Main Design Features

(a) Nuclear Steam Supply System

CAREM is an integral reactor. Its high-energy primary system (core, steam generators, primary coolant and steam dome) is within a single pressure vessel. Primary cooling flow is achieved by natural circulation, which is induced by placing the steam generators above the core. Water enters the core from the lower plenum. After being heated, the coolant exits the core and flows up through the chimney to the upper steam dome. In the upper part, water leaves the chimney through lateral windows to the external region. It then flows down through modular steam generators, decreasing its enthalpy.

(b) Reactor Core

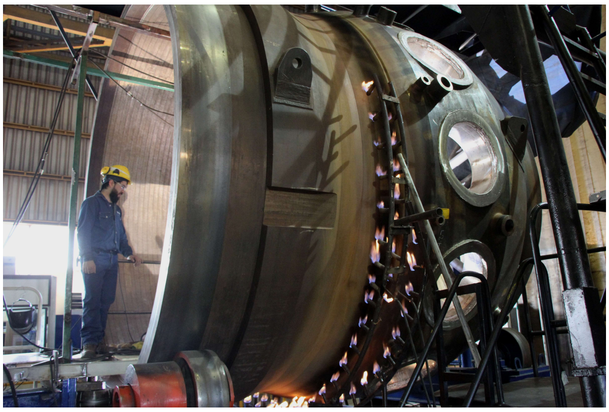
The reactor core of CAREM25 has hexagonal cross section fuel assemblies. There are 61 fuel assemblies with 1.4 meters active length. Each fuel assembly contains 108 fuel rods with 9 mm outer diameter, 18 guide thimbles and one instrumentation thimble. The fuel is 1.8% - 3.1% enriched UO_2. The fuel cycle can be tailored to customer requirements, with a reference design for the prototype of 390 full-power days and 50% of core replacement.

(c) Reactivity Control

Core reactivity is controlled using Gd_2O_3 as burnable poison in specific fuel rods and movable absorbing elements belonging to the power adjustment and reactivity control system. Neutron poison in the coolant is not used for reactivity control during normal operation and in reactor shutdown. Each absorbing element consists of a cluster of rods linked to a structural element (‘spider’), so the whole cluster moves as a single unit. Absorber rods fit into the guide tubes.

(d) Reactor Pressure Vessel and Internals

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



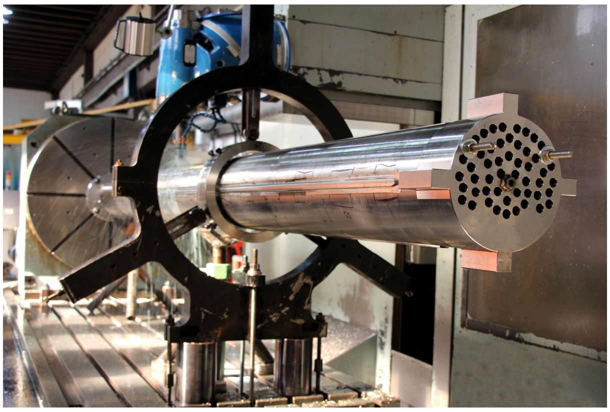
CAREM25 RPV fabrication

(e) Reactor Coolant System

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

(f) Steam Generator

In CAREM25, twelve (12) identical mini-helical vertical steam generators of the once-through type are placed equidistant from each other, along the inner surface of the RPV. Each consists of a system of 6 coiled piping layers, 52 parallel pipes of 26 m active length. They are used to transfer heat from the primary to the secondary circuit, producing superheated dry steam at 4.7MPa. The secondary system circulates upwards within the tubes, while the primary coolant moves in counter-current flow. The steam generators are designed to withstand the primary pressure. The entire secondary side is designed to withstand primary pressure up to isolation valves (including the steam outlet/water inlet headers) in case of SG tube breakage.



CAREM25 steam generator fabrication

(g) Pressurizer

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

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

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

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

(b) Decay Heat Removal System

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

(c) Emergency Core Cooling System

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

(d) Spent Fuel Cooling Safety Approach/System

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

(e) Containment System

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

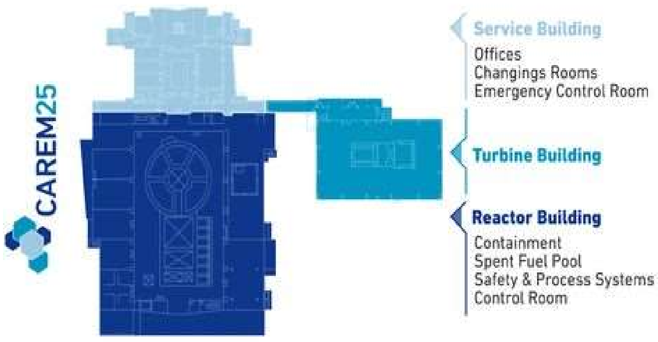
6. Plant Safety and Operational Performances

The natural circulation of coolant produces different flow rates in the primary system according to the power generated and removed. Under different power transients, a self-correcting response in the flow rate is obtained. Due to the self-pressurizing of the RPV, the system keeps the pressure close to the saturation pressure. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps.

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement



CAREM plant layout

CAREM25 construction site

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

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

11. Fuel Cycle Approach

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

12. Waste Management and Disposal Plan

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

13. Plant Economics

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

14. Development Milestones

1984	CAREM concept was presented in Lima, Peru, during the IAEA Conference on SMRs and was one of the first of the new generation reactor designs. CNEA officially launched the CAREM project
2006	Argentina Nuclear Reactivation Plan listed the CAREM25 project among priorities of national nuclear development
2009	CNEA submitted its preliminary safety analysis report (PSAR) for CAREM25 to the ARN
2011	Start-up of a high pressure and high temperature loop for testing the innovative hydraulic control rod drive mechanism
2011	Site excavation work beginning
2012	Civil design beginning
2013	Construction license
2014	Civil works started
2024	Electromechanical erection (start)
2026	First criticality

ACP100 (CNNC/NPIC, China)

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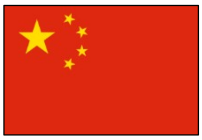


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNNC/NPIC, China
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	385 / 125
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary),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

1. Introduction

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

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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

(c) Reactivity Control

The reactivity is controlled by means of control rods, solid burnable poison and soluble boron dissolved in the primary coolant. There are 20 control rods, with a magnetic force type control rod driving mechanism (CRDM).

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

(f) Steam Generator

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

(g) Pressurizer

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

(h) Primary pumps

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The ACP100 is designed with several passive safety features and severe accident mitigation features. Enhanced safety and physical security of ACP100 are made possible by arranging the nuclear steam supply system and spent fuel pool underground. When the spent fuel pool is filled with spent fuel of 10 years, the pool cooling water can cope for seven (7) days of cooling in the case of accident before boiling dry and uncovering fuel. Severe accident prevention and mitigation are achieved through passive reactor cavity flooding preventing RPV melt, passive hydrogen recombination system preventing containment hydrogen explosion and maintaining the containment integrity after severe accidents, automatic pressure relief system and RPV off-gas system to remove non-condensable gas gathered at RPV head after accidents.

(b) Decay Heat Removal System

The PDHRS prevents core meltdown in the case of design basis accident (DBA) and beyond DBA, such as station black out, complete loss of feedwater, small-break LOCA (i.e., to prevent the change of beyond DBA to severe phase). The PDHRS of ACP100 consists of one emergency cooler and associated valves, piping, and instrumentation. The emergency cooler is located in the in-containment refuelling water storage tank, which provides the heat sink for the emergency cooler. The decay heat is removed from the core by natural circulation. The PDHRS provides core cooling for seven (7) days without operator intervention or long term with IRWST makeup water collected by gravity force from the steam condensed in containment.

(c) Emergency Core Cooling System

The emergency core cooling system (ECCS) consists of two coolant storage tanks (CST), two safety injection tanks (SIT), an in-containment refuelling water storage tank (IRWST) and associated injection lines. The ACP100 has a safety related direct current (DC) power source to support accident mitigation for up to 72 hours, along with auxiliary power units to recharge the battery system for up to seven (7) days. After LOCA accidents, the steam in containment is condensed continuously at containment internal face thus the heat is conducted to containment, which is cooled by PAS, thus ensuring the containment integrity.

(d) Spent Fuel Cooling Safety Approach / System

The In-Refuelling Water Storage Tank (IRWST) is a passive water tank, resting on the internal structure base slab. During refuelling operations, it provides water for refuelling cavity, internals storage compartment and refuelling transfer canal to complete the refuelling operation. Under the condition of LOCA and the steam pipe rupture, it provides water for emergency reactor core cooling. In the severe accidents, water in it floods the internal structure under the balanced water level due to gravity. During the operation of reactor automatic depressurization system, it absorbs the sprayed steam from the RCS. During the operation of the passive residual heat removal cooler, it works as the heat sink of the passive residual heat removal system. The reactor pool is used during refuelling operation or inspection of reactor internals. The reactor pool consists of two compartments which can be separated by bulkhead: reactor cavity and internals storage compartment adjacent to the reactor.

(e) Containment System

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

6. Plant Safety and Operational Performances

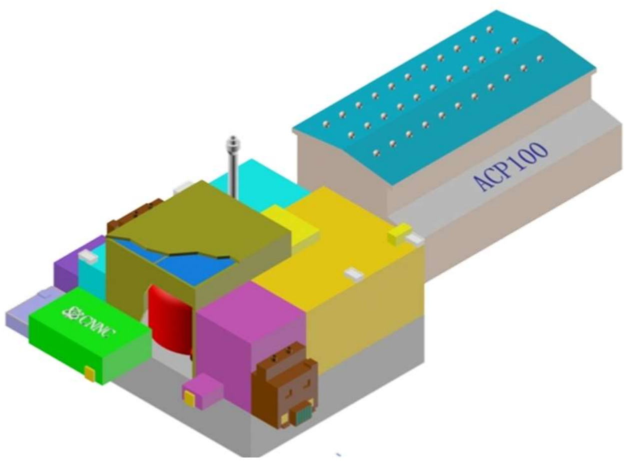
Nuclear safety is always the first priority. The ultimate goal of nuclear safety is to establish and maintain an effective defence that can effectively protect people, community, and environment from radioactive disaster. To be specific, the design and operation of ACP100 ensures that radiation dose to the workers and to the members of the public do not exceed the dose limits and kept it as low as reasonably achievable. Accident prevention measures ensure that radioactive consequences are lower than limited dose in terms of all the considered accident sequences and even in the unlikely severe accidents, mitigation of accidents induced influences can be ensured by implementing emergency plan. The design of ACP100 incorporates operational experience of the state-of-the-art design. Proven technology and equipment are adopted as much as reasonably possible.

7. Instrumentation and Control System

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

8. Plant Layout Arrangement

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



Plant layout arrangement

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

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

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

FCD in July 2021

11. Fuel Cycle Approach

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

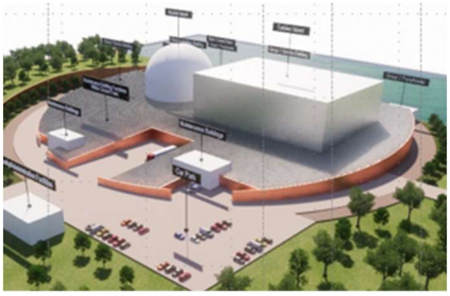
12. Waste Management and Disposal Plan

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

13. Development Milestones

2016	Generic reactor safety review for ACP100 by IAEA finished.
2017	CNNC signed an agreement with the Changjiang municipal government in Hainan Province to host the first of a kind (FOAK) ACP100 demonstration unit.
2018	Preliminary safety assessment report (PSAR) finished.
2019	PSAR submitted to National Nuclear Safety Authority, Site Preparation started.
2020	Apply for authorize to Changjiang nuclear power site, Hainan, China.
2021	FCD on Changjiang nuclear power site, Hainan, China.
2026	Target commercial operation.

CANDU SMR™ (Candu Energy Inc, Canada)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Candu Energy Inc. Member of the SNC-Lavalin Group, Canada
Reactor type	Pressurized Heavy Water Reactor
Coolant/moderator	Heavy Water (D_2O)
Thermal/electrical capacity, MW(t)/MW(e)	960 / 300
Primary circulation	Forced
NSSS Operating Pressure (primary/secondary), MPa	9.9 / 4.6
Core Outlet Coolant Temperature (°C)	310
Fuel type/assembly array	37 elements
Number of fuel assemblies in the core	2 064
Fuel enrichment (%)	Natural Uranium; not enriched
Core Discharge Burnup (GWd/ton)	5.8
Refuelling Cycle (months)	On-line
Reactivity control mechanism	Zone controllers, mechanical adjusters
Approach to safety systems	Combined active and passive
Design life (years)	70 years
Plant footprint (m^2)	21 000
RPV height/diameter (m)	NA; Calandria
RPV weight (metric ton)	300
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Natural Uranium, once-through
Distinguishing features	Natural Uranium fuel (no enrichment); high degree of localization
Design status	Conceptual design

1. Introduction

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

2. Target Application

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

3. Design Philosophy

The CSMR is based on a decades-long proven design that is licensed in many countries operating CANDU reactors. It meets modern regulatory requirements, with dedicated post-Fukushima features.
The CSMR does not require enriched fuel or a fuel-qualification program because it relies on natural uranium in the same fuel bundle employed in multiple CANDU6® reactors. To achieve this it uses heavy water for

moderation and cooling.

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

(f) Steam Generator

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

(g) Pressurizer

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

5. Safety Features

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

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

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

(a) Engineered Safety System Approach and Configuration

All plant systems are assigned to one of two groups (group 1 or group 2) according to the CSMR grouping and separation approach. Group 1 includes the power production systems and delivers safety functions for group 2 support. Group 2 includes the safety related systems required for mitigation of accidents and group 2 systems

are qualified or protected to provide safety functions in the event of severe external events. There is additional separation within the groups.
The CSMR has two separate shutdown systems. These are two fully-capable fast-acting means of shutdown for use at the third level of defence in depth, fully independent of each other and of the reactor regulating system which acts at the second level of defence in depth.

(b) Decay Heat Removal System

Decay heat removal is accomplished in CSMR by application of several provisions in the design, including three group 2 systems, as follows:
The large inventory in the HTS, including the pressurizer inventory, provides heat removal for normal operating transients. The HTS layout enhances natural circulation to the steam generators, which have a large inventory. The steam generators have normal make-up capability and back-up feedwater.
When the steam generators are not available, or are not effective for heat removal, heat can be removed from the HTS using the shutdown cooling system. Under emergency conditions this system can be valved in at full HTS pressure and temperature.

(c) Emergency Core Cooling System

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

(d) Containment System

The basic function of the containment system is to form a continuous, pressure-confining envelope about the reactor core and primary cooling system in order to limit the release of radioactive material to the external environment resulting from an accident. This accident could be either a failure of fuel cooling, or an accident which releases radioactive material into the containment without a rise in containment internal pressure.
To achieve this overall function, the containment system includes the following related safety functions:
i. Isolation: to ensure closure of all openings in containment when an accident occurs.
ii. Pressure/activity reduction: to control and assist in reducing the internal pressure and free radioactive material released into containment by an accident.
iii. Hydrogen control: to limit concentrations of hydrogen/deuterium within containment after an accident to prevent detonation.
iv. Monitoring: to monitor conditions within containment and the status of containment equipment, before, during and after an accident.
In addition to its safety role, the containment structure also serves the following functions:
i. To limit the release of radioactive materials from the reactor to the environment during normal operations.
ii. To provide external shielding against radiation sources within containment during normal operations and after an accident.
iii. To protect reactor systems against external events such as tornados, floods, etc.
The containment system includes a reinforced concrete containment structure (the reactor building) with a reinforced concrete dome and an internal steel liner, access airlocks, equipment hatch, building air coolers for pressure reduction, and a containment isolation system.

6. Plant Safety and Operational Performances

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

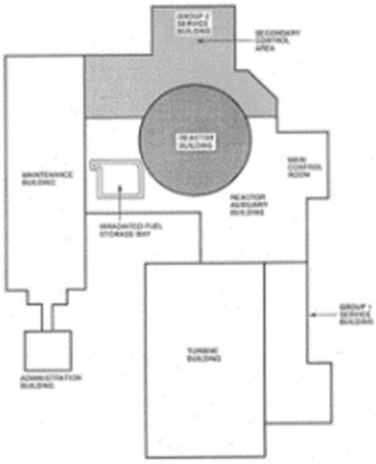
7. Instrumentation and Control Systems

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

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

8. Plant Layout Arrangement

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



Plant Layout

9. Design and Licensing Status

Candu Energy Inc. has requested the CNSC to initiate the formal Vendor Design Review (VDR) process for the CSMR design, which will be conducted based on the regulatory requirements applicable to the new build projects in Canada. The design of CSMR is at the Conceptual Design stage.

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

The CSMR implements systems and equipment for handling radioactive wastes consistent with the state-of-the-art in Canada. The radioactive waste management systems are designed with the objective of limiting routine releases in accord with the ALARA principle. Facilities are provided for interim storage, or controlled release, of all radioactive gaseous, liquid and solid wastes.
It is anticipated that most of the of low-level and intermediate-level solid waste produced over the reactor's 70-plus year lifetime will be stored at the waste management area, with high-level waste being stored on-site in a dry storage facility. Candu Energy Inc. is working on establishing the plan to safely store, handle and dispose of all the spent fuel including the on-site storage, long term storage and potential design for new long-term storage containers that meet the requirements for the deep geological repository design of the Nuclear Waste Management Organization (NWMO).

12. Development Milestones

2020	Design Concept ready for contracting
CED + 1	Conceptual Design complete (CED = Contract Effective Date)
CED + 3	Detailed Design complete
CED + 5	First Concrete for first unit
CED + 8	First Unit in service

CAP200 (SPIC/SNERDI, China)

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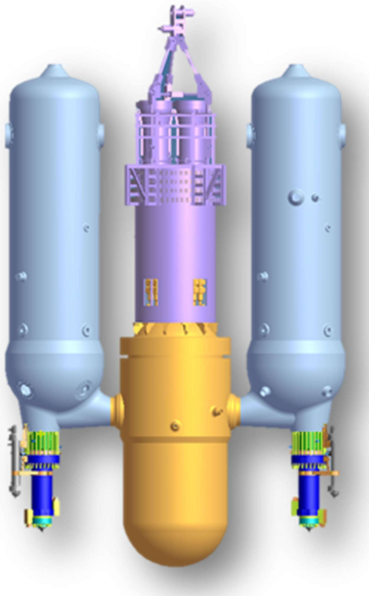


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	SPIC/SNERDI, China
Reactor type	PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	660 / >200
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.5 / 6.06
Core inlet/outlet coolant temperature (°C)	289 / 313
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	89
Fuel enrichment (%)	4.2 (Average)
Refuelling cycle (months)	24
Core discharge burnup	37 (Average)
Reactivity control mechanism	Control rod drive mechanism and soluble boron
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	–
RPV height/diameter (m)	8.845 / 3.280
RPV weight (metric ton)	200
Seismic design (SSE)	0.30 g
Fuel cycle requirements/approach	UO_2 fuel and low-leakage core design
Distinguishing features	Compact layout; Passive safety; Underground containment
Design status	Basic design

1. Introduction

The China Advanced Passive pressurized water reactor 200MW(e) (CAP200) is one of the serial research and development products of PWRs adopting passive engineered safety features initiated by Shanghai Nuclear Engineering Research and Design Institute (SNERDI). The design of CAP200 is based on more than 45 years of experience in PWR R&D, and more than 20 years of experience in PWR construction and safe operation in China. It is the outcome of accumulated experience and achievements of the world's first batch of AP1000 units and the R&D of CAP1400. Furthermore, it adopts safety enhancement measures based on lessons learnt from the Fukushima event.

2. Target Application

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

3. Design Philosophy

CAP200 is a small PWR which is designed with improved safety, flexibility and environmental friendliness, and is also comparable with other SMRs on economy. Compared with large PWRs, CAP200 has a number of advantages such as higher inherent safety, lower frequency of large radioactivity release, longer time of post-accident no operator intervention, smaller environmental impact, lower site restrictions, shorter construction period and smaller financing scale as well as lower financial risk. The main features of this reactor are as follows:

- Compact primary system: Steam generators (SGs) are connected to reactor pressure vessel (RPV) directly and main pipes are eliminated. Compact layout of system and components results in lower risk and possibility of loss of coolant accidents and smaller primary system footprint.
- Modular design and fabrication: main modules can be fabricated in factory and transported to the site for installation. Construction period can be shortened because of high modularization.

Redundant and diverse safety features: redundant and diverse active and passive safety features are deployed, which ensures the reactor core safety and extremely low risk of large radioactivity release.

4. Main Design Features

(a) Nuclear Steam Supply System

The NSSS of CAP200 consists of reactor pressure vessel (include IHP), steam generators, reactor coolant pumps, pressurizer and auxiliary systems. Due to the cancellation of RCS pipe, the radioactivity containment capability of CAP200 is better than traditional PWR.

(b) Reactor Core

The CAP200 reactor core adopts two different types of control rod assemblies. There are 37 assemblies in total. The high worth assemblies are known as “black” rods. These are used for shutdown and large swings of reactivity. The grey rods with lower worth are used for load-follow during operation of a fuel cycle to avoid adjusting soluble boron concentration, which results in a substantial reduction in wastewater generation and treatment for PWR required to execute load-follow operation.

(c) Reactivity Control

Core reactivity is controlled by both soluble boron and control rods. CAP200 is capable of load following without boron dilution. Nevertheless, the control rods need repositioning when boron dilutes. This method suppresses the excess reactivity with soluble boron as conventional PWR does, but to a large degree simplifies the conventional boron system and the dilution operation.

(d) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) of CAP200 is designed with reference to that of CAP1400. The difference is in the inlet and outlet of RPV. The pressure nozzles are used in the case of CAP200, to connect RPV and SGs replacing main pipes. The vessel is cylindrical with a removable flanged hemispherical upper head and hemispherical bottom head. The area with highest neutron fluence of active core region of the vessel is completely forged and free of welding which enhances the confidence of 60 years’ design life and decreases the task of in-service inspection. As a safety enhancement, neutron and temperature detectors enter the reactor through upper head of reactor vessel which eliminates penetrations in the lower head of RPV. This reduces the possibility of a LOCA and uncovering of the core The core is positioned as low as possible in the vessel to decrease re-flood time in the event of an accident. Furthermore, this arrangement is very helpful for successful execution of IVR. The reactor internals, installed in RPV, provide support, protection, alignment and position for the core and control rods to guarantee safe and reliable operation of reactor. The reactor internals consist of the lower internals and the upper internals. The core shroud made of stainless steel is welded which eliminates the occurrence of reactor core damage induced by loosening of baffle bolts. The reactor internals design of CAP200 takes the inner pressure nozzle installation into consideration.

(e) Reactor Coolant System

The primary circuit is designed to achieve a high safety and performance record. The system consists of two heat transfer circuits, each with a steam generator and a reactor coolant pump installed directly onto the steam generator; eliminating the primary piping between the pump and steam generator, and also the primary piping between the reactor vessel and steam generator. The reactor vessel and steam generators are connected directly through the interface nozzles. In addition, the system includes the pressurizer, interconnecting piping, valves, and instrumentation for operational control and safeguards actuation. A simplified support structure for the primary systems reduces in-service inspections and improves accessibility for maintenance. The reactor coolant system pressure boundary provides a barrier against radioactivity release and is designed to provide a high degree of integrity throughout operation of the plant.

(f) Steam Generator

The steam generator (SG) of CAP200 is vertical U-tube type with 3861 tubes made of thermally treated nickelchromiumiron Alloy 690 on a triangular pitch, to fulfil the requirement of heat transfer capacity. The design parameter of steam exit from SG is 6.06 MPa and 183.6 kg/s for thermal design flow rate with no tube plugged, respectively. Pressure nozzles are used to connect RPV and RCP directly with SG. The inner duct of the pressure nozzle connecting RPV with SG side is welded to the partition board of SG water chamber. The nozzles are designed with two ducts. For the nozzles between RPV and SGs, hot water flows in the inner circular duct and cold water in the outer annular duct. Stainless steel bars have been installed in bend area to preclude occurrence of damaging flow induced vibration under all conditions of operation. The feedwater spray nozzle is at top of the feed ring and adopts the separated start-up feedwater pipe which eliminates thermal stratification and prevents occurrence of water hammer. The SG channel head is divided into three parts, hot channel, cold channel, and a third channel through which the coolant is pumped back into reactor vessel.

(g) Pressurizer

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

(h) Primary pumps

Both leak-tight canned motor pump and wet coil pump are possible choices for the RCP of CAP200. There is rich experience of use of canned motor pump and wet coil pump in nuclear power plants, and the current advanced large nuclear reactors are also employing canned motor pump or wet coil pump. The RCP is designed to produce a head of 56.3m at design flow rate of 12000 m^3/h with a cold leg temperature of 289℃. The reactor coolant pump has no shaft seals, eliminating the potential seal failure LOCA, which significantly enhances safety and reduces pump maintenance. The pumps have an internal flywheel to increase the pump rotating inertia and thereby providing a slower rate-of-flow coastdown to improve core thermal margins following the loss of electric power.

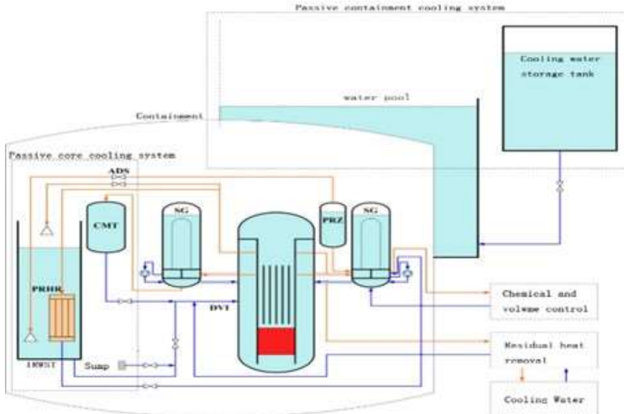
5. Safety Features

(a) Engineered Safety System Approach and Configuration

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

(b) Passive core cooling system

The main function of passive core cooling system is to provide emergency core cooling during postulated design basis accidents by supplement and boration to RCS after non-LOCA accidents and safety injection to core after LOCA. Passive core cooling system forms core decay heat removal pathway together with passive containment cooling system. For CAP200, Passive Safety Injection System combined with the Passive Decay Heat Removal System is referred as the Passive Core Cooling System.



(c) Spent Fuel Cooling Safety Approach / System

The Spent Fuel Cooling System is not required to operate to mitigate design basis events. In the event the spent fuel pool cooling system is unavailable, spent fuel cooling is provided by the heat capacity of the water in the pool. With the use of makeup water from safety related sources, the pool level is maintained above the spent fuel assemblies for at least 72 hours.

(d) Passive Containment Cooling System

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

6. Plant Safety and Operational Performances

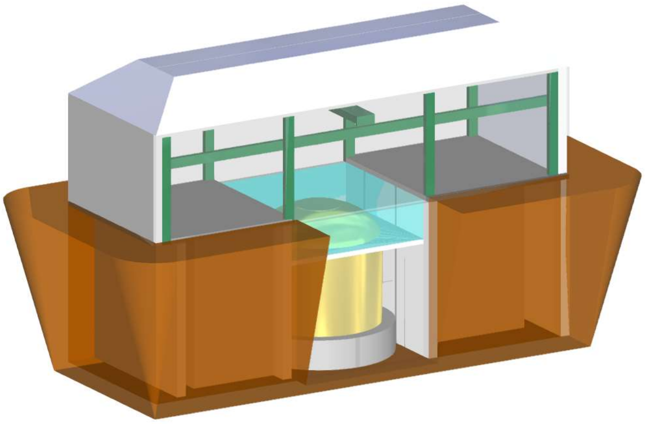
Moderating and maximizing the time response of event loads relative to their limits is a focal point in improving the reactor inventory and cooling safety functions. The total inventory and its distribution throughout the system factor into this assessment. Further, reserve primary coolant from interfacing safety systems, most notably the RWST, can extend these time response periods both temporally and to a broader range of off-normal plant states. The arrangement of reactor core and steam generator thermal centers is crucial to the plant's capability to remove heat by natural circulation following a loss of forced circulation. For CAP200, these two components are vertically separating within an integral pressure vessel.

7. Instrumentation and Control System

The instrumentation and control (I&C) system provides the capability to monitor, control and operate plant systems. It functions to (1) control the normal operation of the facility, (2) ensure critical systems operate within their designed and licensed limits, and (3) provide information and alarms in the control room for the operators. Important operating parameters are monitored and recorded, during both normal operations and emergency conditions to enable necessary operator actions. The I&C system is implemented using modern, scalable digital technology.

8. Plant Layout Arrangement

The reactor building is placedunderground, which is suitable for various geological conditions. It can adapt to the site conditions of most potential sites and islands. The size of Nuclear Island is minimized by both system simplification and adoption of passive design. Compact RCS especially eliminates main pipes and allows reactor building to be smaller. The number of system component is reduced by adoption of large capacity equipment, common use of single equipment for different systems, for example, replacing the polar crane by a travelling crane to use both for reactor building and auxiliary building, etc.



9. Testing Conducted for Design Verification and Validation

The design of passive safety systems of CAP200 takes advantage of the development results of CAP1400 which has been fully validated by Advanced Core-cooling Mechanism Experiment (ACME) and other tests. The Integral Effect Test is considered to be developed in order to give further demonstration to the performance of passive safety systems of CAP200. Hydraulic simulation test of reactor structure and flow-induced vibration test of reactor vessel internals are being implemented to study the uniformity of the inlet flow distribution of the reactor core, the reactor pressure drops, the bypass flow characteristics and the flow-induced vibration response characteristics due to the hydrodynamic loads and interaction with coolant flow.

10. Design and Licensing Status

Basic design started.

11. Fuel Cycle Approach

There are 89 UO2 fuel assemblies in in the core. This core design assumes an end-of-cycle (EOC) exposure of 703 effective full-power days (EFPD). The feed fuel in this design uses two distinct assembly enrichments. The selection of the enrichments and placement within the core are meant to minimize neutron leakage from the core while maintaining acceptable axial and radial power distributions and peaking factors.

12. Waste Management and Disposal Plan

Waste Management System consists of liquid, gaseous, and solid radwaste systems. The liquid radwaste system is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences. The gaseous radwaste system is designed to receive hydrogen bearing and radioactive gases generated during process operation and provides the capability to reduce the amounts of radioactive nuclides. The solid radwaste system is designed to collect, process, package and temporarily store the wastes like spent ion exchange resins, spent filter cartridges, dry active wastes and etc. After stored for several years, the waste will be transported to the disposal site qualified to receive radioactive waste for geological disposal.

13. Development Milestones

2014	Conceptual design started
2015	Conceptual design finished
2018	Further deepen conceptual design
2022	Basic design started

DHR400 (CNNC, China)

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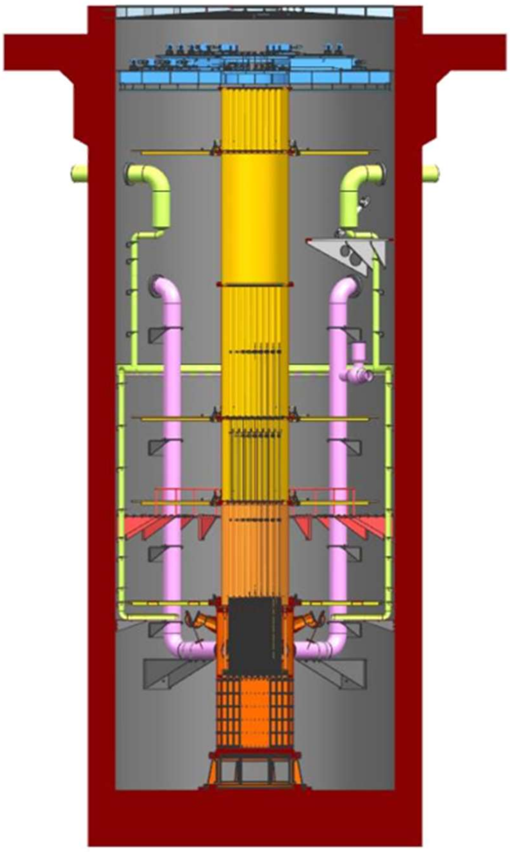


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	China National Nuclear Corporation (CNNC), China
Reactor type	Pool type reactor
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	400 / no electricity
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.3 (Core inlet pressure)
Core Inlet/Outlet Coolant Temperature (ºC)	68 / 98
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	69
Fuel enrichment (%)	< 5
Core Discharge Burnup (GWd/ton)	31
Refuelling Cycle (months)	15
Reactivity control mechanism	Control rod drive mechanisms
Approach to safety systems	Inherent safety features with large water inventory in the reactor pool / Passive core residual heat removal system / Passive heat pipe system
Design life (years)	60
Plant footprint (m^2)	100 000
Pool depth/diameter (m)	28 / 10
Seismic Design (SSE)	0.3g
Distinguishing features	Inherent safety features / Economical / Heat load tracking capability
Design status	Basic design

1. Introduction

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

2. Target Application

DHR400 is a pool type reactor for district heating with a thermal power of 400 MW. The reactor is designed to replace traditional small regional heating plants using coal or natural gas to minimize air pollution and reduce carbon emission.

3. Design Philosophy

The DHR400 is designed based on the pool type research reactor. It operates under low temperature with atmospheric pressure above the pool surface. With the deep reactor pool, appropriate core outlet temperature is achieved by the static pressure of the water layer. The design configuration precludes typical reactor accidents including the loss of coolant accident (LOCA), control rod ejection accident and the loss of decay heat removal capability. DHR400 is moderated by low pressure water, inherent reactor shutdown at abnormal situations is assured with the large negative temperature reactivity coefficient. The DHR400 design also adopts proven pool type reactor technology, design simplification, convenient maintenance, highly automation and minimal operators to obtain enhanced safety and improved economic efficiency.

4. Main Design Features

(a) Nuclear Steam Supply System

Not Applicable (N/A).

(b) Reactor Core

The core of DHR400 consists of 69 fuel assemblies. Each fuel assembly is 2.15m long and its design is modified from the standard 17×17 PWR fuel assembly with 264 fuel rods, and 9 supporting tubes. The fuel is UO_2 with Gd as a burnable absorber. The ^{235}U enrichment is below 5%. The reactor operates 150 days per year (typical 5-month winter period in northern China) and a three-batch refuelling is conducted off power on a 15-month refuelling cycle, that is, the reactor is refuelled every 3 years.

(c) Reactivity Control

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

(d) Reactor Pool and Internals

The reactor pool is a cylinder with an inside diameter of 10 m and an overall height of 28 m, containing the reactor core and in-pool structures inside its 25 m depth of water. The pool is buried underground with an elevation of its bottom at -27m. The pool is made of reinforced concrete with an inner layer of duplex stainless steel. The upper head includes a carbon steel truss and a stainless-steel plate, connected to the concrete wall of the pool, and provides support for the control rod driven mechanism and the control rod guide tubes. Below the upper head there is a plenum, which is connected to an engineered venting system to exhaust vapour and other gases. Internal structures are designed to support the core and isolate the water inside the core from the water in the pool. The large water inventory in the pool provides large thermal inertia and a long response time, thus enhances the resistance to system transients and accidents. These features ensure that the core will not melt down under any accident.

(e) Primary Coolant System

During normal operations, the 68ºC inlet water enters the core from the bottom, the water is then heated to 98ºC. The outlet water enters the 4 outlet pipes through the rising barrel right above the core and then enters 4 separate primary pump room through the pool wall. The water entering the primary pump room flows into the primary heat exchanger. Inside the primary heat exchanger, the primary water is cooled to 68ºC by the secondary side water and then pumped back to the upper part of the reactor pool. The water then flows down to the bottom and enters the core again. No boiling will occur during normal operations.

(f) Secondary Coolant System

The secondary coolant system is an isolated sealed intermediate loop that separates the primary loop and the tertiary loop while transfer heat from the primary to the tertiary. 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 water will contaminate the tertiary loop.

(g) Primary Heat Exchanger

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal System

In normal shutdown situation, decay heat is removed by primary coolant system. When the primary pump is shutdown, the decay heat is removed from the reactor core to the reactor pool with natural circulation. And the heat of the pool water is transferred to the environment with pool water cooling system and passive heat pipe system.

(c) Emergency core cooling system

No emergency core cooling system is employed in the DHR system. The reactor pool with 1800 tons of water provides an immense heat sink. Even under the circumstances that all cooling system failed, core cooling can be maintained with the evaporation of the pool water for several days. So, emergency core cooling system is not necessary for DHR.

(d) Spent fuel cooling safety Approach/System

Spent fuel pool cooling and purification system is designed to remove the decay heat released from the spent fuel and maintain the pool water temperature within an acceptable limit.

(e) Confinement System

There are three barriers precluding a radioactive release to the environment in DHR, including the fuel coating, the reactor pool and the confinement building on top of the pool. Due to the low operating temperature and atmospheric pressure above the reactor pool, there are no high-pressure events and instead of a containment, a confinement building is sufficient for protection. The reactor is located underground and the reactor core is submerged in 1800 tons of water, making DHR400 highly resistant to external events. Additional protection is provided by the reactor confinement building above the pool.

(f) Waste Gas Treatment System

A waste gas treatment system is adopted to store and decay short life gaseous fission products.

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control System

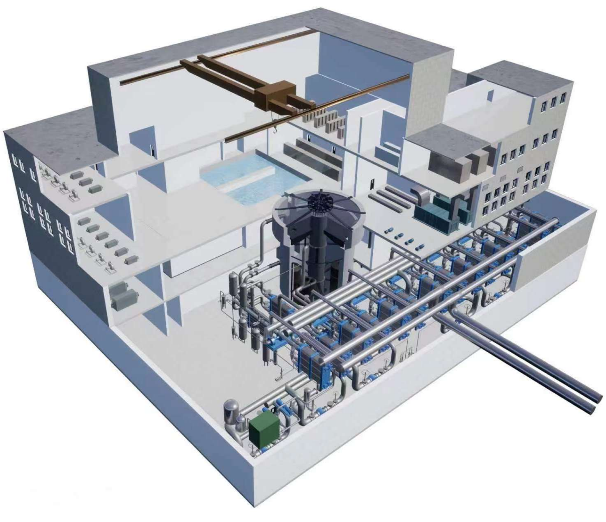
The instrument and control (I&C) system of the reactor will adopt distributed instrument control system to provide consistent and effective human-machine interface, improve the availability, reliability, operability, and human-machine interface design of the system, and reduce the requirements for spare parts with a considerable design flexibility.

8. Plant Layout Arrangement

The layout of the DHR400 is illustrated below. The main building is arranged in a single unit, which combines the reactor confinement building, auxiliary building, and exhaust tower. The reactor confinement building is the core of the main building, and the auxiliary building includes the heat supply building, the radioactive waste treatment and storage building and other auxiliary buildings. It is arranged around the reactor building to facilitate the connection between the buildings.

(a) Reactor Confinement Building

The reactor confinement building includes reactor pool, spent fuel pool, storage pool, reactor hall and primary loop system room.



Plant Layout

(b) Heating Building

The heat supply building includes the water pump room and the secondary heat exchanger room, which are located at one side of the reactor confinement building and transfer the core heat from the primary loop to the heat supply network.

(c) Radioactive waste treatment and storage Building

The radioactive waste treatment and storage building is located on the other side of the reactor confinement building. It collects and stores radioactive liquid, solid and gaseous wastes generated during operation and maintenance to meet the requirements of radioactive waste management.

(d) Other auxiliary buildings

Other auxiliary buildings include new fuel room, circuit purification system room, spent fuel pool cooling system room, equipment cooling water room, supply and exhaust fan room, electrical equipment room, main control room, electronic equipment room, physical protection centre, etc.

(e) Exhaust tower

The exhaust tower is located at the exhaust fan room on the 5.000m floor, with a total height of 40m. From the elevation of 11.000m to 24.000m, it is a reinforced concrete structure, and from the elevation of 24.000m to 40.000m, it is a steel circular exhaust duct. The radioactive gas (radioactive aerosol and radioactive iodine) from the radioactive area of the main building is first filtered and then discharged at high altitude through the exhaust tower to meet the environmental radiation safety requirements.

9. Design and Licensing Status

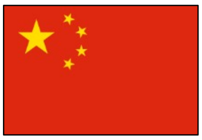
The basic design of DHR has just completed (some parameters might change with the optimization of DHR400 design). Seeking for construction license in 2023.

10. Waste Management and Disposal Plan

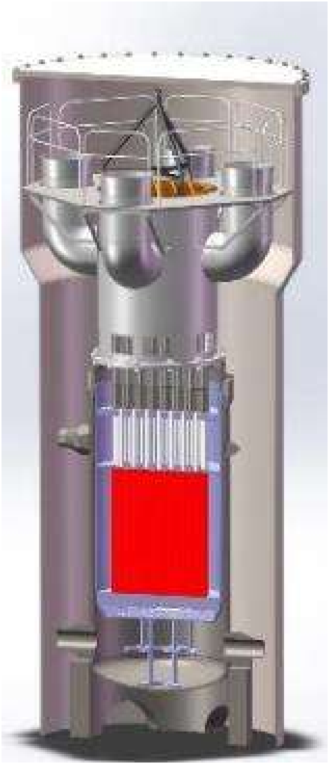
The radioactive waste management system is designed for the storage and treatment of radioactive waste. The system includes radioactive waste liquid collection and treatment system, radioactive waste resin collection and treatment system, solid waste collection system and radioactive waste gas collection and treatment system.

11. Development Milestones

2023	Construction licence
2026	Commercial operation



HAPPY200 (SPIC, China)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	State Power Investment Corporation, Ltd. (SPIC), China
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	200 / 0 (thermal power only)
Primary circulation	Forced (2 pumps)
NSSS Operating Pressure (primary/heating), MPa	0.6 / 0.8
Core Inlet/Outlet Coolant Temperature (^oC)	80 / 120
Fuel type/assembly array	UO_2 / Square 17x17
Number of fuel assemblies in the core	37
Fuel enrichment (%)	2.76 avg / 4.45 max
Core Discharge Burnup (GWd/ton)	40
Refuelling Cycle (months)	18
Reactivity control mechanism	Rods
Approach to safety systems	Active / Passive
Design life (years)	60
Plant footprint (m^2)	1150
RPV height/diameter (m)	6.0 / 2.25
RPV weight (metric ton)	26
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	LEU / once through
Distinguishing features	Pool-loop combined type reactor, heat generator.
Design status	Detailed Design

1. Introduction

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

2. Target Application

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

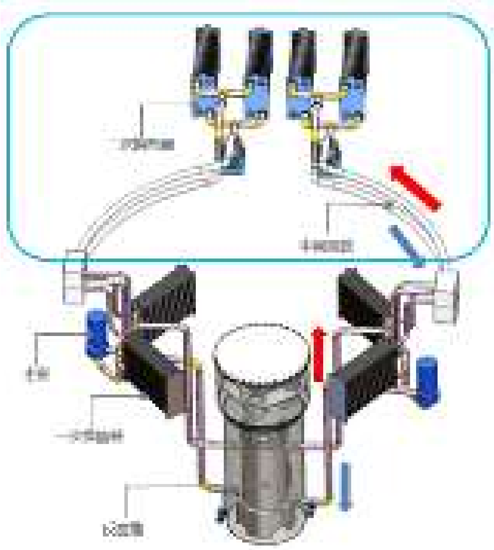
3. Design Philosophy

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

4. Main Design Features

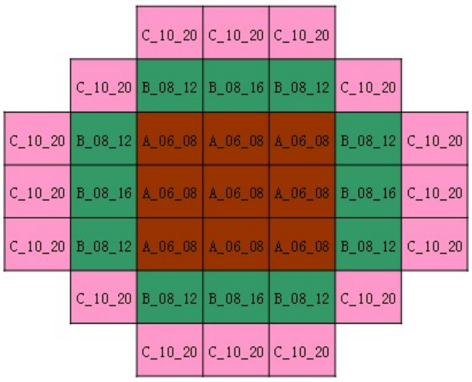
(a) Primary Circuit

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



(b) Reactor Core and Fuel

The core of the HAPPY200 consists of 37 fuel assemblies. Each fuel assembly is 2.1 m long and its design is modified from a standard 17×17 PWR fuel assembly with 264 fuel rods. The fuel is UO_2 with Gd as a burnable absorber. The ^{235}U enrichment is below 5%. The reactor operates 180 days per year (typical 6-month winter period in northern China) and a three-batch refuelling is conducted off power on an 18-month refuelling cycle.



(c) Secondary Side

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

(d) Reactivity Control

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

(e) Reactor Pressure Vessel and Internals

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal System

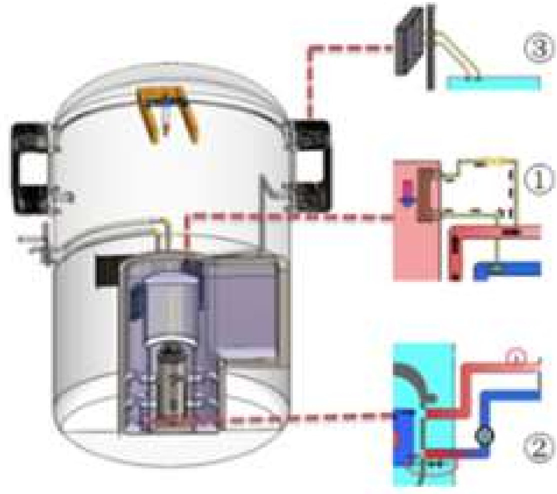
The passive residual heat removal system is designed to remove the residual heat from the reactor core during unexpected operational occurrences and events, especially for the non-LOCA events. It consists of two series vertical tube heat exchangers connected with the hot legs and cold legs of each loop.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

(c) Emergency Core Cooling System

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

(d) Containment System

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



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

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

(a) Engineered Safety System Approach and Configuration

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

(b) Electrical building

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

(c) Fuel building

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

(d) Auxiliary building

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

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

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

11. Fuel Cycle Approach

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

12. Waste Management and Disposal Plan

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

13. Development Milestones

2015	Start market investigation and concept design (changes)
2016	Concept design completed
2019	Start pre-project work in north of China
2025	To start construction of a proto-type NPP in China
2027	Target commissioning and commercial operation dates

NHR200-II (Tsinghua University and CGN, China)

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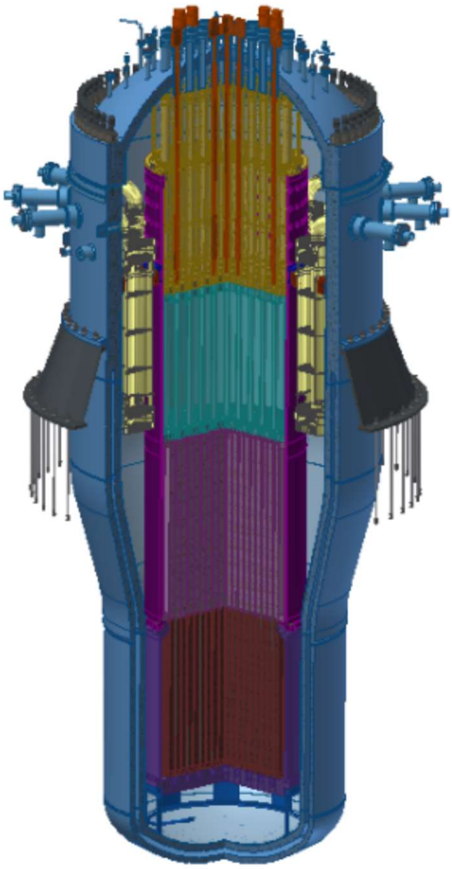


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Tsinghua University and CGN, China
Reactor type	Integral PWR
Coolant/moderator	Light Water / Light Water
Thermal/electrical capacity, MW(t)/MW(e)	200 MW(t)
Primary circulation	Natural Circulation
NSSS operating pressure (primary/secondary), MPa	8
Core inlet/outlet coolant temperature (°C)	232 / 280
Fuel type/assembly array	UO_2 pellet / square 9 × 9
Number of fuel assemblies in the core	208
Fuel enrichment (%)	1.8, 2.67, 3.4
Refuelling cycle (months)	24
Core discharge burnup (GWd/ton)	30.8 (average)
Reactivity control mechanism	The hydraulic control rod drive mechanisms
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	–
RPV height/diameter (m)	13.5 / 4.94 (max, external diameter)
RPV weight (metric ton)	388
Seismic design (SSE)	0.3 g
Fuel cycle requirements/approach	in-out refuelling
Distinguishing features	Natural Circulation, Passive Safety
Design status	Basic design

1. Introduction

NHR200-II is an integrated light water reactor adopting full-power natural circulation in the primary loop and passive safety features which is developed by Tsinghua University and China General Nuclear Group (CGN). NHR200-II is based on the 5 MWt experimental Nuclear Heating Reactor (NHR) and 200 MWt Nuclear Heating Reactor (NHR200-I) which are designed by Tsinghua University.

2. Target Application

NHR200-II is a multipurpose reactor. It can be used for district heating, industrial steam supply, steam/water cogeneration, heat-electricity cogeneration and seawater desalination.

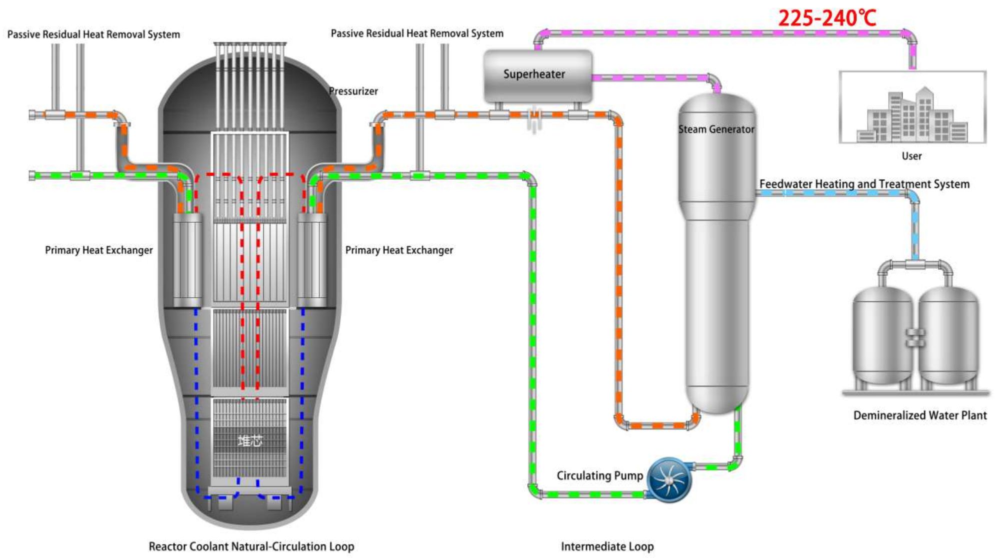
3. Design Philosophy

- Integrated reactor
- Full-power natural circulation
- Passive safety feature and simplification systems
- Near to the city or customers

4. Main Design Features

(a) Nuclear Steam Supply System

The integrated nuclear steam supply system (NSSS) design consists of the reactor core and fourteen primary heat exchangers. The heat of core is transferred to Steam Generator (SG) by the intermediate loop.



Simplified diagram of NSSS.

The simplified diagram of NSSS is shown as above.

(b) Reactor Core

The core of NHR200-II consists of 208 fuel assemblies and 52 control rod assemblies. Each fuel assembly consists of a fuel bundle and a Zircaloy-4 channel which surrounds the fuel bundle. The fuel is UO_2 with Gd_2O_3 as a burnable absorber homogeneously mixed within the fuel for select rod locations. The control rod assemblies are inserted into the gap between four fuel assemblies. The reactor has a refueling cycle of 24 months.

(c) Reactivity Control

Reactivity control is achieved through 52 control rod assemblies. The control rod assembly consists of a sheathed cruciform array of stainless steel tubes filled with boron carbide powder. The hydraulic control rod drive mechanisms (CRDMs), which are designed by INET, are placed in the reactor pressure vessel (RPV) and can eliminate the possibility of postulated rod ejection accidents.

(d) Reactor Pressure Vessel and Internals

The RPV consists of a cylindrical steel vessel with an inside diameter of approximately 4.3m and an overall height of approximately 13.5m. The main pipes are eliminated and small-size nozzles are placed on the higher part of the vessel, or on the upper head, so the feasibility of a primary coolant loss accident would be much lower. And the lower part of the vessel has two layers. For the most serious condition, a break occurring on the bottom of the RPV, the coolant will be contained by the second layer of the vessel, assuring that the reactor core can always be covered by water. 14 primary heat exchangers are placed between the core barrel and RPV, vertically higher than the core.

(e) Reactor Coolant System

The NHR200-II core, which is located in the lower part of the RPV, is cooled through natural circulation without reactor coolant pumps. When flowing through the reactor core, the coolant is heated and flows upward into the riser. After flowing through the primary heat exchangers, where it is cooled by the water of the intermediate loop, the core coolant increases in density, naturally flows downward and returns to the core inlet. The pressure of the reactor coolant system is automatically maintained by the nitrogen/steam space in the top of the RPV.

(f) Steam Generator

Each NHR200-II has 14 primary heat exchangers located in the RPV and 2 steam generators located outside the containment. Two intermediate loops, each of which connects one steam generator and seven primary heat exchangers, are deployed to carry heat from the reactor and transfer it to the water/steam of the second loop. The primary heat exchanger is tube-in-tube type and the core coolant flows inside the inner tubes and outside the outer tubes, while the intermediate loop water flows in the gaps between the inner and outer tubes.

(g) Pressurizer

Not Applicable. No separate pressurizer is existed in NHR200-II. And the pressure of the reactor coolant system is automatically maintained by the nitrogen/steam space in the top of the RPV.

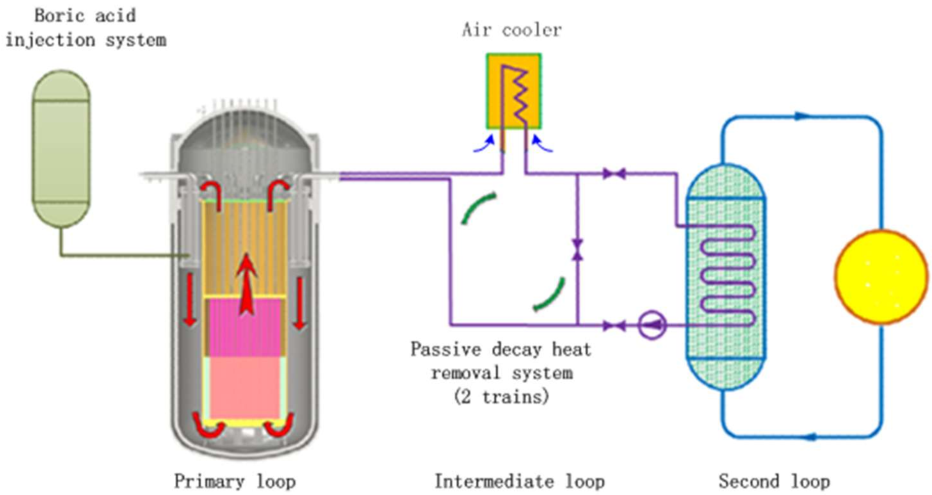
(h) Primary pumps

The NHR200-II is a kind of natural circulation reactor.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

As NHR200-II has been designed with a number of advanced and innovative features, only a few engineered safety systems are needed, including the decay heat removal system, the passive boric acid injection system, and the containment system. The passive safety systems are shown below.



Passive Engineered Safety Systems

(b) Decay Heat Removal System

Two trains of passive decay heat removal loops are deployed independently. Each train contains of 7 primary heat exchangers (second side), a set of air coolers and associated pipes and valves. The core decay heat after reactor shutdown is finally dispersed into the atmosphere by natural circulation. There is no need of a power supply, so the decay heat removal will be ensured. The core decay heat can also be dispersed into the plant service water system via the heat exchangers of the chemical and volume control system, which is not a safety class system.

(c) Emergency Core Cooling System

Not Applicable.

(d) Containment System

The reactor and some auxiliary systems are placed in a small concrete containment. According to our design, the containment heat removal system is not needed. At the same time, a higher leak rate is permitted because the accident radioactive source term is much lower than that of the nuclear power plants in operation.

6. Plant Safety and Operational Performances

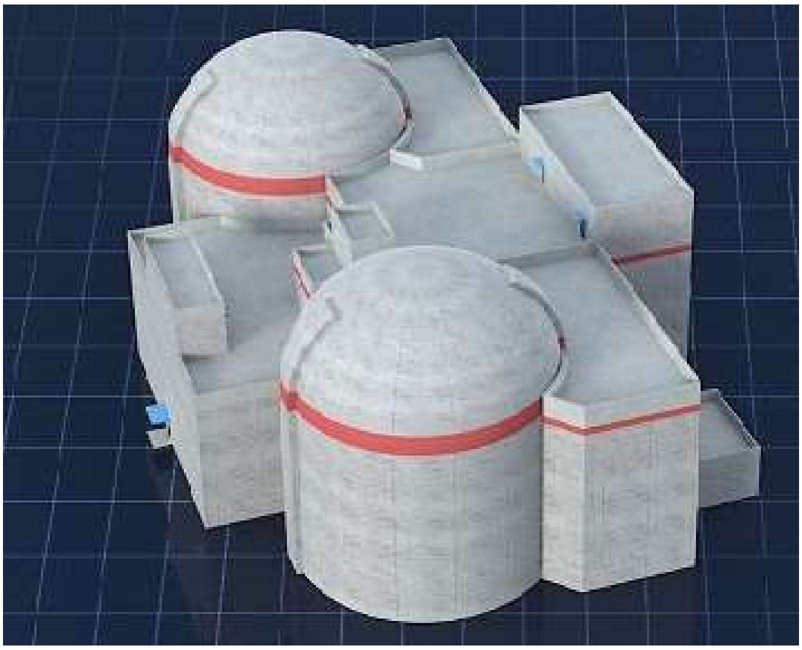
According to the basic design, a preliminary PSA of internal events of NHR200-II has been carried out, the result of which shows that the core damage frequency (CDF) is less than 1E-8/reactor-year. Because the systems are highly simplified, the operation of NHR200-II is so easy that only 5 operators are needed in a two-reactor-module control room.

7. Instrumentation and Control System

The instrumentation and control system of NHR200-II 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 the reactor building and nuclear auxiliary buildings. The spent fuel storage pool is housed in the containment, so the extra fuel storage building is eliminated. For different applications, different equipment can be arranged in the conventional island, such as steam turbo-generators, heat exchanger stations, or seawater desalination equipment.



Nuclear island and conventional island layout

9. Design and Licensing Status

The basic design of NHR200-II has been completed and the preliminary safety analysis report (PSAR) is published. In Jan. 2018, based on the preliminary feasibility study results, the feasibility study of the demonstration nuclear heat-supply plant in Hebei Province was permitted by China National Energy Administration.

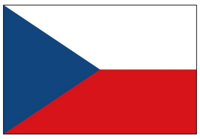
10. Waste Management and Disposal Plan

Waste management approach and disposal plan is similar to other nuclear power plants, and different waste management and disposal technology could be supplied for different site.

11. Development Milestones

1989	5MWt experimental reactor was constructed
1996	The design of NHR200-I was completed
2006	INET began to develop NHR200-II
2016	Basic design of NHR200-II was completed
2022	Plan to complete the preliminary work of NHR200-II demonstration project

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



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	UWB Pilsen & CIIRC CTU Prague, Czech Republic
Reactor type	Channels in Reactor Vessel
Coolant/moderator	Heavy Water (D_2O)/ Heavy Water (D_2O)
Thermal/electrical capacity, MW(t)/MW(e)	<150 / does not produce electricity
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	< 2 / < 2
Core Inlet/Outlet Coolant Temperature (°C)	< 175 / < 192
Fuel type/assembly array	VVER-440 / hexagonal with 126 fuel pins
Number of fuel assemblies	55
Fuel enrichment (%)	Spent fuel (< 1.2 wt% ^{235}U equivalent) or slightly enriched fresh fuel
Core Discharge Burnup (GWd/ton)	2.3
Refuelling Cycle (months)	10 months with the online option
Reactivity control mechanism	Control rods, Moderator height
Approach to safety systems	Inherent and passive safety with built/in decay heat sink
Design life (years)	60
Plant footprint (m^2)	≤ 2 000
RPV height/diameter (m)	8 / 4
RPV weight (metric ton)	N/A
Seismic Design (SSE)	0.2 g
Fuel cycle requirements / Approach	LEU - reuse of LWR spent FAs, possibility to run on fresh SEU
Distinguishing features	District heating zero CO_2 source with zero fuel cost, heat-storage for decay heat removal.
Design status	Conceptual design

1. Introduction

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

2. Target Application

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

3. Main Design Features

(a) Design Philosophy

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

(b) Nuclear Steam Supply System

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

(c) Reactor Core

The TEPLATOR core consists of equally spaced channels filled with spent nuclear fuel from LWR reactors. More customisation options are possible. The initial one is the reuse of VVER-440 spent nuclear fuel. In that case, totally 55 fuel assemblies are placed in a large-pitched hexagonal array. Typical VVER-440 spent nuclear fuel uses 3.6 wt% ^{235}U initial enrichment, 35 GWd/ton average burnup and 30 years of cooling. Alternative use of slightly enriched fresh fuel is possible. Each fuel assembly is placed in a coolant tube filled with heavy water or alternatives for temperatures up to 192 °C. The atmospheric pressure of a heavy water moderator eliminates the need for a thick and expensive pressure vessel, the temperature of the moderator will be lower than 60 °C. The TEPLATOR is a heat generator with a typical operation time of 10 months each year (regular European heating season).

(d) Reactivity Control

Two independent systems are deployed. Reactivity control under normal operation is achieved by changing control rods positions. The second reactivity control system is based on moderator pumping in or draining out the reactor calandria. The safety shut-down system is based on a combination of the control rods and moderator draining systems.

(e) Calandria and Internals

The internals of TEPLATOR consist of the fuel channels, channel outlets, control rods, control rods drive mechanism, I&C systems, axial and radial reflector, and bottom collector. The coolant (heavy water) is distributed through the bottom collector to individual channels. The calandria works on ambient pressure, it does not need to be pressurized. The space between fuel channels and calandria is filled with heavy water, which serves as a moderator. The total volume of heavy water in calandria is around 30 m^3. A graphite reflector surrounds the core both in the axial and radial directions.

(f) Reactor Coolant System

The primary coolant (D_2O) enters the core of TEPLATOR at the temperature of 175 °C. It flows through the fuel channel, and then it leaves the individual channels at 192 °C at the channel outlet. This outlet is attached to the collector where the primary coolant is collected. From the collector, the coolant is distributed to the three heat exchangers where it heats the secondary heat transfer fluid (HTF). The primary coolant flows through the pump, then through the pipe on the inside of the calandria to the bottom collector where it is distributed again to the individual channels. Roughly 20 m^3 of D_2O is required in the primary circuit.

(g) Primary Heat Exchanger

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

(h) Pressurizer

The pressurizer is linked to the heavy water management systems, and it operates at low pressure (2 MPa).

(i) Secondary side

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

4. Safety Features

General safety features

The TEPLATOR operating conditions (e.g., fuel/coolant temperature, pressure, linear heat production) are much lower than those for which the used FAs were certified and used in LWRs. The safety features establish defence-in-depth against radiological hazards. The TEPLATOR leverages the inherent safety characteristics of the basic LWR reactor design and supplements them with passive and active safety features that emphasizes proven and result in an improvement in safety. The TEPLATOR secondary circuit provides large volumes of fluid that are available to provide cooling to the core in the event of accidents, including by passive means.

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

(a) Engineered Safety System Approach and Configuration

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

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

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

(c) Containment System

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

5. Plant Safety and Operational Performances

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

6. Instrumentation and Control Systems

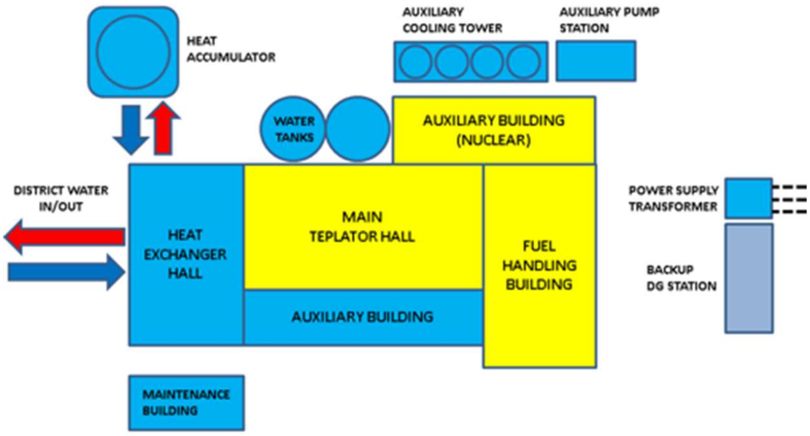
The instrumentation and control (I&C) system provides the capability to monitor, control and operate plant systems. The I&C system is implemented using mature technology; i.e., known and tested LWR detectors and systems.

7. Plant Layout Arrangement

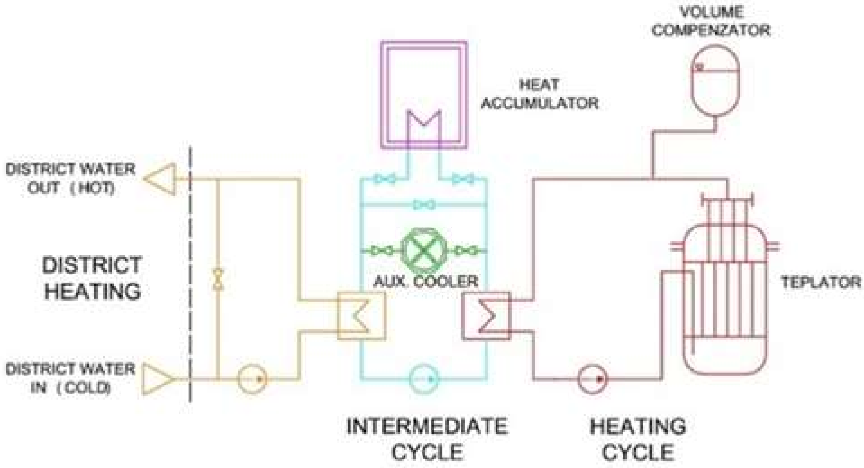
The plan layout of the TEPLATOR is illustrated below.

(a) Reactor Building

The TEPLATOR facility consists of one main structure which further contains nuclear and non-nuclear sectors/buildings. Nuclear sectors are the main TEPLATOR hall, the fuel handling building, and the auxiliary nuclear building. Non-nuclear sectors are the heat exchanger hall and the auxiliary building. Other buildings and structures within the facility layout are



a heat accumulator, water storage tanks, auxiliary cooling towers with a pumping station, transformers of power supply and backup diesel generators.



(a) Balance of Plant

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

8. Design and Licensing Status

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

9. Fuel Cycle Approach

The unique feature of TEPLATOR is the reuse of spent nuclear fuel from commercial LWRs which is normally considered a waste. This can be achieved due to significantly lower operation parameters (well under limits) when compared to the conditions in large LWRs. Based on heating or cooling demand, the core can be operated from 10 months each year with subsequent refuelling of fuel assemblies. Usage of fresh SEU assemblies is possible and increases the heat price by roughly 20%.

10. Waste Management and Disposal Plan

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

11. Development Milestones

2019- 2020	Preliminary studies and pre-conceptual design
2020-2021	Conceptual design phase and technology validation
2021-2024	Design phase - Basic design Detail design / Licensing
2024-	Start of DEMO unit construction, first unit in service in 2028+

NUWARD^{TM} (EDF, France)

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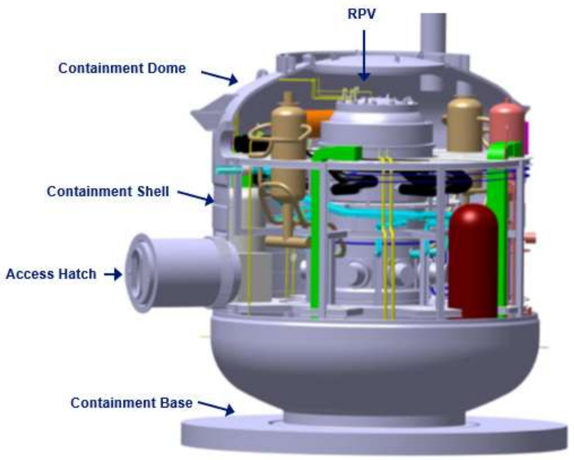
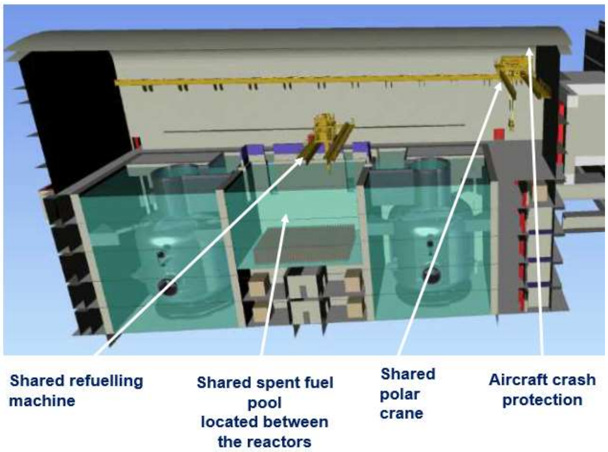




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	EDF, France with major contributions from CEA, Naval Group, Framatome, TechnicAtome, and Tractebel-Engie
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	2 x 540 / 2 x 170
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15 / 4.5
Core Inlet/Outlet Coolant Temperature (°C)	280 / 307
Fuel type/assembly array	UO_2 / 17x17 square pitch arrangement
Number of fuel assemblies in the core	76
Fuel enrichment (%)	<5
Refuelling Cycle (months)	24 (half core)
Reactivity control	Control rod drive mechanism (CRDM), solid burnable poisons
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	3500, nuclear island including fuel storage pool
RPV height/diameter (m)	15 / 5
RPV weight (metric ton)	310
Seismic Design (SSE)	0.3g
Distinguishing features	Integrated NSSS with pool submerged containment, boron-free in normal operation and in all Design Basis Conditions (DBC), semi-buried nuclear island
Design status	Conceptual Design

1. Introduction

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

2. Target Application

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

3. Design Philosophy

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

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

4. Main Design Features

(a) Nuclear Steam Supply System

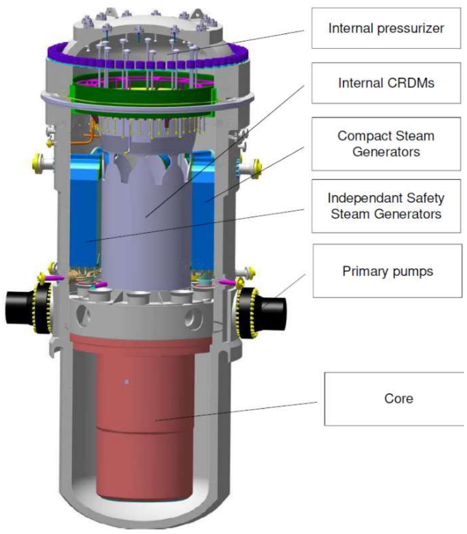
The NUWARD™ reactor is a fully integrated PWR reactor, housing a single unique vessel containing all the main reactor coolant system components, including the CSGs, the pressuriser and the CRDM.

(b) Reactor Core

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

(c) Reactivity Control

The reactivity is controlled by means of control rods and solid burnable poison. The reactor boron-free reactivity control allows for simplification of auxiliary systems design and operation in both normal conditions and DBCs, as well as a drastic reduction in effluents produced from operation.



Reactor Cross-section View

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System and Steam Generator

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

(f) Pressuriser

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

(g) Primary pumps

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

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

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

(b) Safety Approach and Configuration to Manage DBC

NUWARD™ incorporates 2 trains of passive heat removal, via a natural circulation transfer system of the decay heat from the core to the water surrounding the unit containment (the reactor pool), through two dedicated Safety CSGs (S-CSG) independent from the operational or the operational six CSGs. Each train can be actuated by 2 diversified channels (diversified sensors and I&C) and redundant (backup spare) actuators. The water surrounding the unit containment ensures the heat removal function for more than 3 days without the need for an external ultimate heat sink. The passive vessel heat removal system is considered a D-passive system according to IAEA classification. A set of 2 redundant low-pressure safety injection accumulators provide the make-up of reactor coolant water inventory in case of LOCA. NUWARD™ includes safety features to prevent criticality risks. The use of an internal CRDM eliminates the occurrence of a rod ejection accident. Dedicated safety systems to manage DBC are provided for each of the two reactor units.

(c) Safety Approach and Configuration to Manage DEC

DEC-A systems include Low flowrate depressurisation system and active water injection. This system provides for the removal of the decay heat in case of a postulated common mode failure of redundant trains of passive DBC safety systems; high pressure borated water injection is available to cope with Anticipated Transients Without Scram (ATWS) accidents. DEC-B systems include the low flowrate depressurisation system; Flooding of the vessel pit in order to provide IVR of corium; and nitrogen injection to manage the risk of hydrogen combustion.

(d) Containment System

The NUWARD™ design includes steel containment as the 3rd barrier, which is immersed in water. The minimised LOCA break size and the efficient passive heat removal system result in a limited peak pressure inside the steel containment which is passively cooled by the surrounding water. The containment is protected against hydrogen accumulation risk in DBCs by passive recombiners.

(e) Spent Fuel Cooling Safety Approach / System

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

6. Plant Safety and Operational Performances

The design target value for the lifetime capacity factor is above 90%, with the major planned refuelling only outages scheduled for 20 days every 24 months. The reference refuelling strategy is to replace half of a core every 2 years. The plant provides a storage capacity of spent fuel assemblies for 10 years of operation.

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement

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

NUWARD™ example plant layout

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

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

11. Fuel Cycle Approach

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

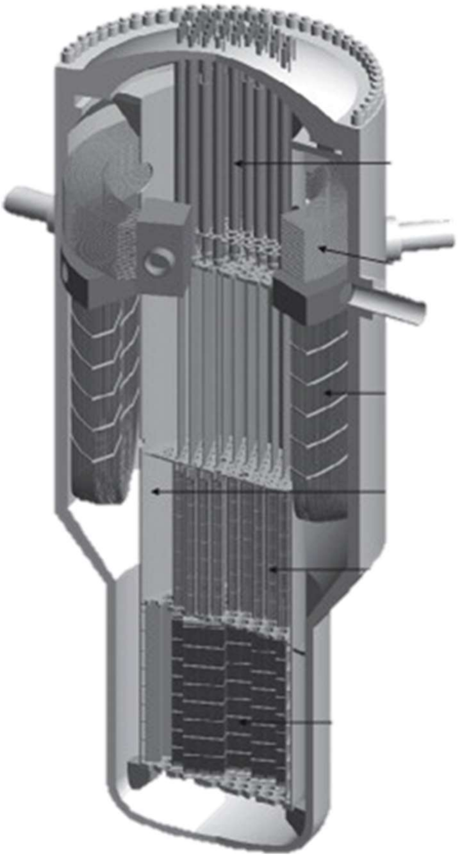
12. Waste Management and Disposal Plan

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

13. Development Milestones

2012 – 2016	Preliminary studies and technological innovation (using previously developed patents).
2017 – 2019	Pre-conceptual design phase and technology validation
2019 – 2022	Conceptual Design Phase (and preparation for pre-licensing)
2023 – 2026	Basic Design Phase
From 2025	Commercialisation
2026 – 2030	Detailed Design Phase
By 2030	Target first concrete for the FOAK in France

IMR (Mitsubishi Heavy Industries, Japan)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Mitsubishi Heavy Industries, Ltd. (MHI), Japan
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	1 000 / 350
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 5.0
Core Inlet/Outlet Coolant Temperature (°C)	329 / 345
Fuel type/assembly array	UO_2 pellet / 21x21 square
Number of fuel assemblies in the core	97
Fuel enrichment (%)	4.8
Core Discharge Burnup (GWd/ton)	> 40
Refuelling Cycle (months)	26
Reactivity control mechanism	Control rods drive mechanism
Approach to safety systems	Hybrid (Passive + Active) system
Design life (years)	60
Plant footprint (m^2)	4900
RPV height/diameter (m)	17 / 6
RPV weight (metric ton)	–
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Similar to existing PWR plants
Distinguishing features	Integral PWR with natural circulation; employs two types of in-vessel steam generator
Design status	Conceptual design completed

1. Introduction

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

2. Target Application

The IMR is primarily designed as a land-based modular power station to generate electricity. Because of its modular characteristics, it is suitable for large-scale power stations consisting of several modules and also for small distributed-power stations, especially for small grids. IMR can also be used for cogeneration of electricity and district heating, seawater desalination, process steam production and so forth. IMR adopts structures, systems and components that require no large-scale infrastructure. This facilitates regulatory authority's approval for the construction and operation of power plant.

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

The IMR core consists of 97 fuel assemblies in 21×21 array with an average enrichment of 4.95 % and produces an output of 1000 MW(t). The refuelling interval is 26 effective full-power months. The power density is about 40% of current PWRs but the fuel lifetime is 6.5 years longer, so that an average discharged burnup of 46 GWd/ton can be attained, which is approximately the same as in current PWRs. The cladding material is Zr–Nb alloy to assure integrity at a temperature of 345°C and over the long reactor lifetime. To maintain the core thermal margin and to achieve a long fuel cycle, the core power density is reduced to one-third of that conventional PWRs. The design-refuelling interval is three (3) years in three (3) batches of fuel replacement. The fuel rod design is the same as that for a conventional PWR.

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

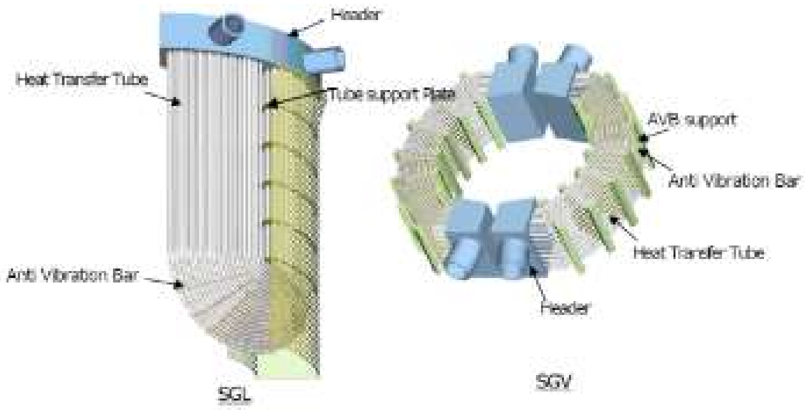
The upper part of the RPV inside diameter is about 6 m in order to accommodate the in-vessel SGs. The inside diameter of the lower part of the RPV is reduced to about 4 m in order to minimize the cold-side water volume. In order to eliminate the necessity for the consideration of LOCA, the largest diameter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

(e) Reactor Coolant System

In the HHTS, IMR employs natural circulation and a self-pressurized primary coolant system, altogether resulting in a simple primary system design without reactor coolant pumps and pressurizer, it also reduces maintenance requirements. The HHTS reduces the size of RPV. The coolant starts boiling in the upper part of the core, and two-phase coolant in bubbly flow flows up in the riser and condenses in the SGs. This design approach increases coolant flow rate and thus reduces the height of RPV to transport the heat from the core. The IMR primary cooling system design under bubbly flow makes it easy to employ PWR design technologies.

(f) Steam Generator

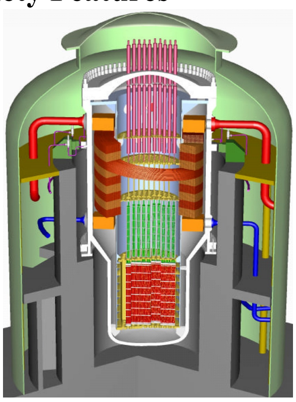
The IMR adopts two types of SG. The first one is the SG in vapour region (SGV) located above the water level in the RPV. The energy transported by vapour formation generates secondary steam through SGV. As the vapour in the RPV is condensed by SGV, controlling the feedwater flow rate to SGV controls the RPV pressure. The other is the steam generator in liquid region (SGL) of the RPV. The energy transported by liquid temperature rise generates secondary steam through SGL. Because the core inlet temperature can be controlled by the amount of heat removal through SGL, the core power can be controlled by feedwater flow rate to SGL. By this method, the movement of the control rods for controlling reactor power will be minimized. For SGL, a U-type tube bundle is adopted, since it is necessary to minimize pressure drops on both the primary and secondary sides to maintain good natural circulation performance. A C-type steam generator is adopted for SGV to optimize space utilization in the vapour part of the RPV.

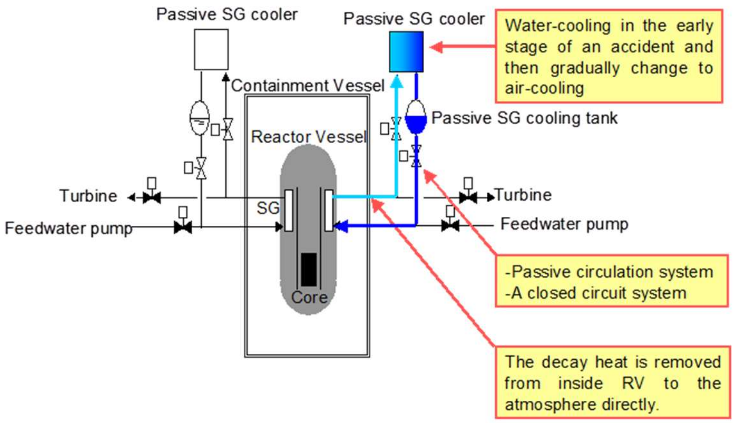


(g) Pressurizer

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

5. Safety Features





(a) Engineered Safety System Approach and Configuration

By adopting an integral type primary system, accidents that may cause fuel failure, such as loss of coolant accidents (LOCA), rod ejection (R/E), loss-of-flow (LOF) and locked rotor (L/R), are eliminated in IMR. Since the diameter of the pipes connected to the RPV is limited to less than 10 mm, the water level in the RPV can be maintained at normal levels by water injection from the charging pumps. There are two trains of the SDHS. If a malfunction such as SG tube leakage occurs, system functions are maintained. The capacity of chemical and volume control system (CVCS) is provided via eight 3-inch pipes connected to the RPV. No ECCS and containment cooling/spray systems are required in IMR. Safety injection systems are eliminated by adopting the SDHS and by limiting the nozzle diameter connected to the primary system. Hence, containment spray system is also eliminated. The auxiliary feedwater system is used for startup and shutdown procedures during normal operation. The auxiliary feedwater system is not a safety system. When the auxiliary feedwater system becomes unavailable, the SDHS is actuated. The IMR adopts simplified support systems, such as the component cooling water system (CCWS), the essential service water system (ESWS) and the emergency AC power system. These are designed as non-safety grade systems powered by a stand-alone diesel generator.

(b) Decay Heat Removal System

The SDHS is activated to remove the decay heat from the RPV to the atmosphere. Even if water leakage occurs and the charging pumps fails to operate, water leakage would be terminated automatically when the pressures inside and outside the RPV are equalized. In the passive steam generator cooler (PSGC), decay heat is removed by water-cooling in the early stage of the accident and then, the heat transfer mode is gradually replaced by air-cooling. Therefore, water, power and operators are not necessary for maintaining the plant safety.

(c) Containment System

A compact containment vessel (CV) is made possible due to the integrated primary system and simplified auxiliary systems. The IMR uses reinforced concrete containment. A higher design pressure of the containment is to meet the safety requirement that water leakage from RPV shall be terminated automatically. Since this CV is about one size larger than the RPV, it is expected to resist high pressure. The reactor containment facility is part of the engineered safety systems, which include SDHS. The containment system is designed to suppress or prevent the possible dispersion of large quantities of radioactive materials.

6. Plant Safety and Operational Performances

The IMR is designed to operate automatically within the range of 20 to 100 % of rated output power by the reactor control system. Even in the low output range below 20%, the control system can control the reactor automatically in the low power-operating mode. The primary system pressure and reactor power are controlled by feedwater and control rods.

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement

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

(a) Reactor Building

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

(b) Balance of Plant

The advanced BOP system allows the utilization of produced heat for non-electrical applications such as process heat, mining (oil sand extraction) and desalination. The turbine generator, turbine, condenser, moisture separator and reheater (MSR) and their auxiliary equipment are installed in the turbine building. The turbine generator is arranged with its axis in line with the reactor.

9. Design and Licensing Status

The IMR conceptual design study was initiated in 1999 by MHI. A group led by MHI including Kyoto University, the Central Research Institute of the Electric Power Industry and the Japan Atomic Power Company developed related key technologies through two projects, funded by the Japanese Ministry of Economy, Trade and Industry (2001–2004 and 2005–2007). Validation testing, research and development for components and design methods, and basic design development are required before licensing.

10. Fuel Cycle Approach

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

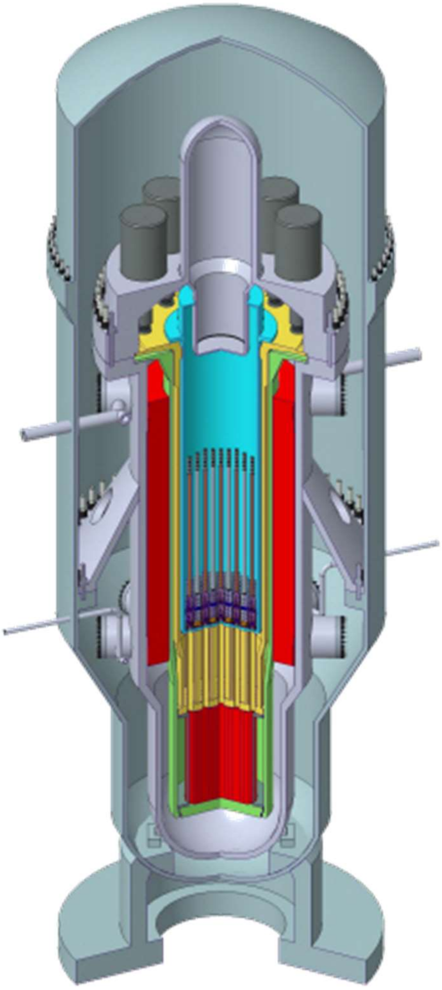
11. Waste Management and Disposal Plan

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

12. Development Milestones

1999	MHI started conceptual design study for IMR.
2001-2004	An industry-university group led by MHI, including Kyoto University, Central Research Institute of Electric Power Industries (CRIEPI), the Japan Atomic Power Company (JAPC), and MHI were developing related key technologies through two projects, funded by the Japan Ministry of Economy, Trade and Industry. In the first project, the feasibility of the HHTS concept was tested through experiments.
2005-2007	In the second project, the thermal-hydraulic data under natural circulation conditions for the HHTS design were obtained by four series of simulation tests using alternate fluids.
2009-2011	Startup transient tests to verify the startup flow instability were studied
2019	MHI is developing a new Small Reactor based on the IMR experiences with funding support by the Ministry of Economy, Trade and Industry.

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



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	KHNP&KAERI Republic of Korea
Reactor type	Integral PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	540 / 170
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.0
Core inlet/outlet coolant temperature (ºC)	295.5 / 320.9
Fuel type/assembly array	UO_2 pellet / 17×17 square
Number of fuel assemblies in the core	69
Fuel enrichment (%)	< 5.0
Refuelling cycle (months)	24
Core discharge burnup (GWd/ton)	< 62.0
Reactivity control mechanism	Control rod, burnable absorber rods, moderator temperature
Approach to safety systems	Fully passive
Design life (years)	80
Plant footprint (m^2)	NA
RPV height/diameter (m)	23 / 5
RPV weight (metric ton)	650
Seismic design (SSE)	0.5g
Fuel cycle requirements/approach	Conventional LWR requirements applied
Distinguishing features	At least 72 hours for core cooling without AC/DC power
Design status	Conceptual design

1. Introduction

In order to promote sustainable development and respond to climate change, the i-SMR is on development in Republic of Korea with improved safety, economy and flexibility compared to existing nuclear power plants. The i-SMR is an innovative small modular PWR producing 540MW 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, eight Reactor Coolant Pumps (RCPs), a helical once-through Steam Generator (SG), and a Pressurizer (PZR) are installed in a single Reactor pressure Vessel (RV) and equipped in a compact steel Containment Vessel (CV). Due to the integral arrangement, the possibility of a Large Break Loss of Coolant Accident (LB-LOCA) is inherently eliminated.

2. Target Application

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

3. Design Philosophy

The i-SMR implements design simplification by integrating the primary coolant system and passive safety systems without external safety water injection, least operating action by a high-level autonomous control.

4. Main Design Features

(a) Nuclear Steam Supply System

The i-SMR is an integral type PWR with compact steel containment vessel. This configuration eliminates a LB-LOCA from the design bases events. The Nuclear Steam Supply System (NSSS) consists of the reactor core, SG, RCPs, Control Element Drive Mechanisms (CEDMs), and reactor internals in the RV. The primary cooling system is based on forced circulation by the RCPs during normal operation. The system has natural circulation capability in emergency conditions that ensures 72 hours coping time for core cooling without AC/DC power. Moreover, air cooling for decay heat removal after water source depletion provides for unlimited long-term cooling capability.

(b) Reactor Core

The core of i-SMR consists of 69 fuel assemblies. The active height of fuel assembly is 2.4 m and 17x17 square fuel assembly containing gadolinia mixed in uranium dioxide with less than 5% enrichment. The refuelling scheme is a two-batch application with 24-month cycle length.

(c) Reactivity Control

Reactivity control during normal operation is achieved by control rods and moderator temperature. The i-SMR adopts boron-free operation and eliminates significant amount of component and maintenance activities related to boron generation and concentration. Burnable poison rods are introduced to give flat radial and axial power profiles, which results in an increased thermal margin of the core. The i-SMR adopts an internal control element drive mechanism which excludes the rod ejection accident. A large number of control rods in i-SMR core assures a relatively high control rod worth.

(d) Reactor Pressure Vessel and Internals

The reactor vessel assembly of the i-SMR contains its major primary components such as a core, a SG, a PZR, eight RCPs, and CEDMs in a single RV.

(e) Reactor Coolant System

The RCS transfers heat generated from the core to the secondary system through the SG and plays a role of 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 Occurrences (AOO).

(f) Steam Generator

The SG of i-SMR, which is installed inside the RV, is a helical once-through type. Feedwater flow upward through the tube side of SG. The tubes of the SG are helical-designed to provide the maximum heat transfer area in a limited space. The primary and coolant flow inside the RV is treated as single path for the normal operation.

(g) Pressurizer

The pressurizer for i-SMR is located at top-side of the RV, since this configuration have advantages for pressurizing the reactor coolant in a subcooled state. The pressurizer is designed as the steam pressurizer which saturated steam and water are co-existed in pressurizer region. The steam pressurizer has an advantage that the simple control schemes are provided by two-phase phenomena during the transients.

(h) Primary pumps

The i-SMR has eight RCPs vertically installed at upper part of the Reactor Closure Header (RCH). Each RCP is an integral unit consisting of a canned asynchronous three phase motor. Rotational speed of the pump rotor is measured by sensors installed in the upper part of the motor. Since a canned motor pump does not require pump seals, possibility of the Small Break Loss of Coolant Accident (SB-LOCA) associated with a pump seal failure is basically eliminated.

5. Safety Features

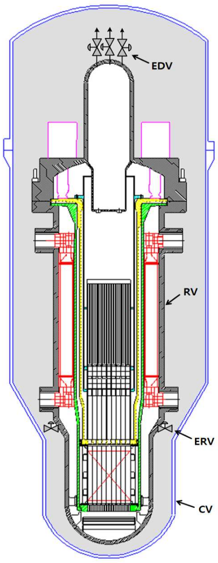
(a) Engineered Safety System Approach and Configuration

Safety approach for design and operation of the i-SMR is based on the defense-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 fully passive safety systems. The safety systems of i-SMR, a sensible mixture of proven technologies and advanced design features, are designed to function automatically on demand without critical operator action. Under a postulated design basis accident, the safety systems cool the RCS below the safe shutdown condition within 36 hours and keep the core undamaged for 72 hours without any corrective actions by operators. Heat from the RCS can be dissipated to the heat sinks, such as the Emergency Cooling

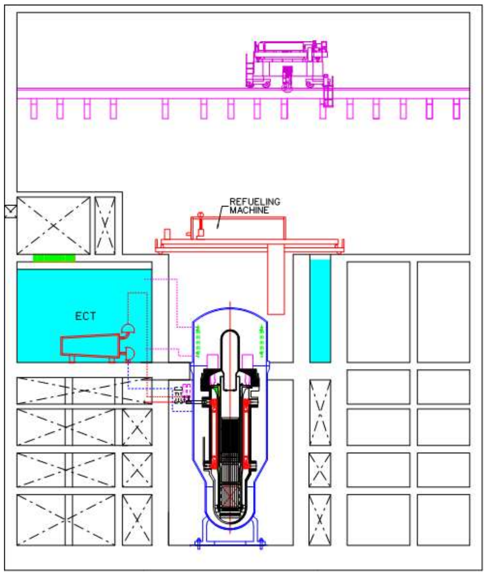
Tank (ECT) and the Reactor Building (RB) atmosphere. Also the safety systems maintain the safe shutdown condition for at least 72 hours under the complete loss of onsite and offsite AC power.

(b) Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) is composed of Emergency Depressurization Valves (EDVs) and Emergency Recirculation Valves (ERVs). Three EDVs and two ERVs open when pressure difference between the RV and CV reaches the predetermined setpoint. When LOCA occurs, the coolant releases to the CV annulus between the RV and the CV through EDVs. When ERVs open, water flows into the RCS through ERVs due to the hydrostatic head difference and forms natural recirculation path.
Decay heat is removed by a Passive Auxiliary Feedwater System (PAFS) and Passive Containment Cooling System (PCCS) in non-LOCA and LOCA scenarios, respectively.



Schematic diagram of the ECCS



PAFS and PCCS schematics

(c) Decay Heat Removal System

After reactor trip, when normal decay heat removal mechanism utilizing the secondary system is not working properly, the PAFS brings the RCS to safe shutdown condition within 36 hours after accident initiation and maintains the safe shutdown condition for at least another 36 hours. The PAFS consists of inlet pipes, PAFS tubes, outlet pipes, and actuation valves. In normal operation, actuation valves are closed and main steam flows to the turbine. When the PAFS signal occurs, the PAFS actuation valve and main steam isolation valve are closed. Hence, the steam generated in the SG flows from the main steam line to the PAFS tube and condensed due to the cold ECT water and return to the feedwater line. Therefore, safety function is performed for at least 72 hours without any corrective action by operator or the aid of AC/DC power for the design basis accidents. The safety function of the PAFS is maintained continuously for a long-term period, coupled with the ECCS without refilling the ECT.

(d) Containment Cooling System

The CV is composed of stainless steel, and divided to upper and lower regions of distinct diameters. The CV is supported by support lugs located beneath the CV. Outer surface of the CV is exposed to the RB inner atmosphere. The CV provides penetration of PCCS piping connected between the PCCS Heat exchangers (HX) and the ECT, and main steam and feedwater piping, etc. Double isolation valves mounted on the CV are installed to prevent LOCA outside containment events for the RCPB lines penetrating the CV.
The PCCS consists of inlet pipes, the PCCS HX which is located inside the containment, and return pipes. When LOCA occurs, the hot steam contacts the PCCS HX outer surface and condensation occurs. Heated water inside the PCCS HX flow upward to the ECT and natural circulation flow path formed. In the PCCS system, there are no actuation signal and operator action required to perform its safety function.

6. Plant Safety and Operational Performances

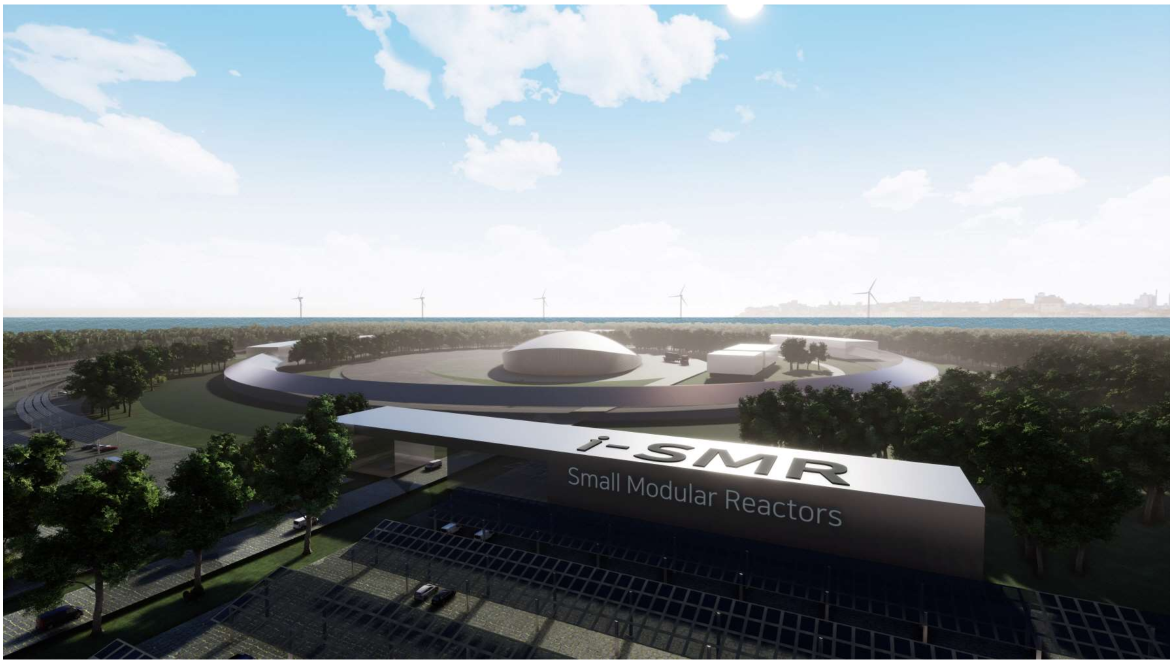
The safety of the i-SMR is achieved in a core damage frequency less than 10-9 and a large release frequency less than 10-10 per year. The operating experience of KHNP have been applied to the i-SMR with the expected available factor no less than 95%. Load following operation of i-SMR is simpler than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

7. Instrumentation and Control System

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 Power Cells safely under all operating conditions (except MCR fire) and maintain it in a safe condition under all operating conditions.
The HMIS is developing to satisfy all regulatory requirements such as independence, redundancy, defense-in-depth and diversity and to improve the economics and operability.

8. Plant Layout Arrangement

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



Plant layout and arrangement

9. 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

10. Fuel Cycle Approach

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

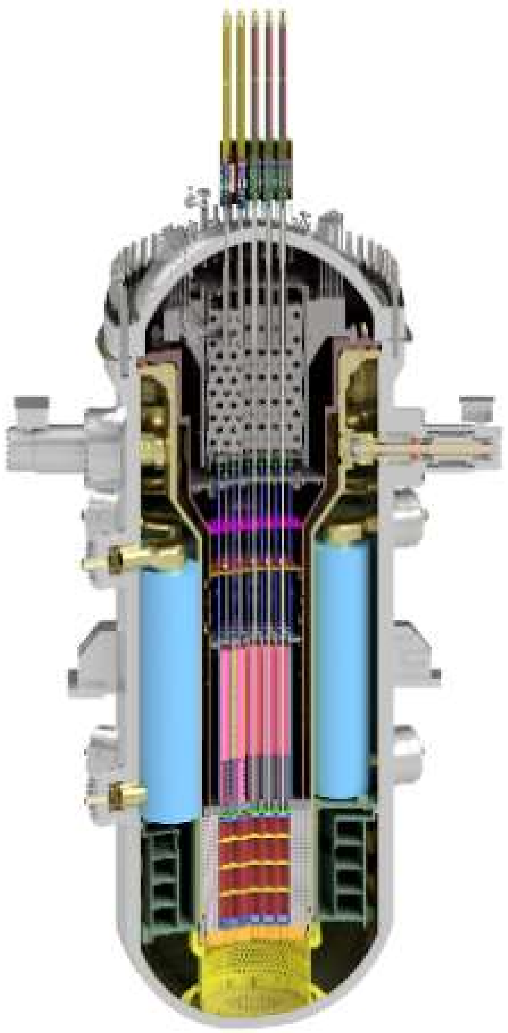
11. Waste Management and Disposal Plan

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

12. Development Milestones

2023	Basic design completed
2028	Standard design approval

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



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	KAERI, Republic of Korea and K.A.CARE, Kingdom of Saudi Arabia
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	365 / 107
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15 / 5.8
Core inlet/outlet coolant temperature (°C)	296 / 322
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	57
Fuel enrichment (%)	< 5
Refuelling cycle (months)	30
Core discharge burnup (GWd/ton)	< 54
Reactivity control mechanism	Control rod drive mechanisms and soluble boron
Approach to safety	Passive
Design life (years)	60
Plant footprint (m^2)	90 000
RPV height/diameter (m)	18.5 / 6.5
RPV weight (metric ton)	1070 (including coolant)
Seismic design (SSE)	> 0.3g with 0.18g of automatic shutdown
Fuel cycle requirements/approach	Conventional LWR requirements applied (spent fuel storage capacity: 30 years)
Distinguishing features	Coupling with desalination and process heat application, integrated primary system
Design status	Detailed design

1. Introduction

The System-integrated modular advanced reactor (SMART) is an integral PWR with a rated electrical power of 107 MW(e) from 365 MW(t). SMART adopts advanced design features to enhance safety, reliability and economics. The advanced design features and technologies were verified and validated during the standard design approval review. To enhance safety and reliability, the design configuration incorporates inherent safety features and passive safety systems. The design aim is to achieve improvement in economics through system simplification, component modularization, reduction of construction time and high plant availability.

2. Target Application

The SMART is a multi-purpose reactor for electric power generation, desalination, district heating, and process heat for industries. SMART has been developed to be suitable for small or isolated grids. SMART has a unit output large enough to meet the demands of electricity and fresh water for a city population of 100 000.

3. Design Philosophy

The SMART design adopts an integrated primary system, modularization and advanced passive safety systems to improve the safety, reliability and economics. Safety performance of SMART is assured by adopting passive safety systems together with severe accident mitigation features. Improvement in economics is achieved through system simplification, in-factory fabrication, reduction of construction time and high plant availability.

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

The low power density design with a slightly enriched UO_2 fuelled core ensures a thermal margin of greater than 15%. In the core, there are 57 fuel assemblies of 2 m long, standard 17x17 square of UO_2 ceramic fuel with less than 5% enrichment, similar to standard PWR fuel. A two-batch refuelling 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 which has been widely used in the commercial nuclear power plants (NPPs).

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

(f) Steam Generator

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

(g) Pressurizer

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

(h) Primary pumps

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

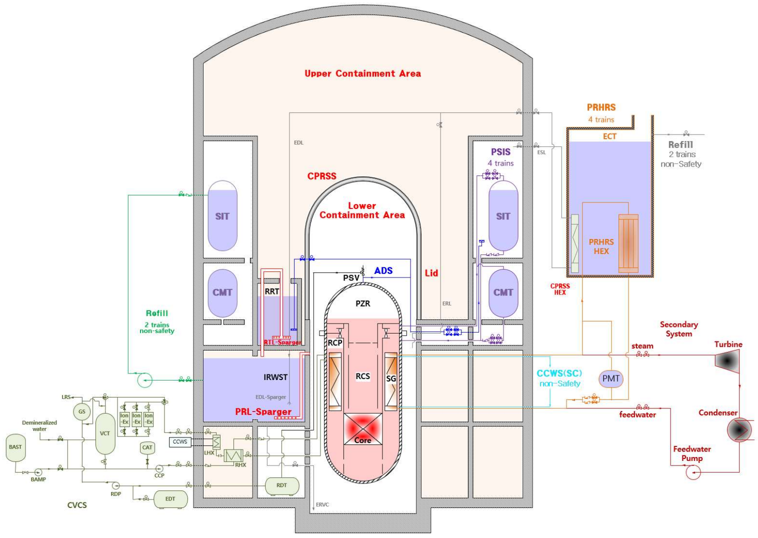
Safety systems of SMART are designed to function automatically on demand. These consist of the PRHRS, PSIS, and CPRSS. Additional safety systems include the ADS and pressurizer safety valves, and a severe accident mitigation system.

(b) Decay Heat Removal System

After the reactor is shutdown, when the normal decay heat removal mechanism utilizing the secondary system is not operable for any reason, the PRHRS brings the RCS to a safe shutdown condition within 36 hours after accident initiation and maintains the safe shutdown condition for at least another 36 hours. Therefore, the safety function operates for 72 hours without any corrective action by operators for the postulated design basis accidents. The safety function of PRHRS is maintained continuously for a long-term period when the emergency cooldown tank (ECT) is replenished periodically by a refilling system designed according to Regulatory Treatment of Non-Safety System (RTNSS) requirements.

(c) Emergency Core Cooling System

The PSIS provides emergency core cooling following postulated design basis accidents. Emergency core cooling is performed using 4 core make-up tanks (CMTs) and 4 safety injection tanks (SITs). Core cooling inventory is maintained through passive safety injection from CMTs and SITs. The CMTs that are full of borated water provide makeup and borating functions to the RCS during the early stage of an SBLOCA or non-LOCA. The top and bottom of CMT are connected to the RCS through the pressure balance line and safety injection line, respectively. The safety injection function of the PSIS is maintained long term as the SITs are replenished periodically.



Emergency Core Cooling System

(d) Containment System

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

6. Plant Safety and Operational Performances

The core damage frequency of SMART is estimated at about 2E-7 per reactor year for internal events. Power manoeuvring operation of SMART is simpler than that of large PWR because only a single bank movement and small insertion is required to induce small reactivity change. This feature minimizes coolant temperature change, relatively high lead bank worth due to a small number of fuel assemblies and the short core height leading to rapidly damping the xenon oscillation. The daily load following performance simulation of SMART core shows that radial peaking factor, 3D peaking factor and the axial offset were satisfied within design limit.

7. Instrumentation and Control System

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

8. Plant Layout Arrangement

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

Plant Layout Arrangement

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

Korea Atomic Energy Research Institute (KAERI) received the standard design approval for SMART from the Korean Nuclear Safety and Security Commission (NSSC) in July 2012. A safety enhancement program to adopt passive safety system in SMART began in March 2012, and the testing and verification of the PRHRS and PSIS were completed in the end of 2015. In September 2015, a pre-project engineering (PPE) agreement was signed between the Republic of Korea and the Kingdom of Saudi Arabia for deployment of SMART. This PPE project was successfully completed in February 2019. In 2019, Korea Hydro & Nuclear Power Co., Ltd. (KHNP)/KAERI/K.A.CARE co-applied for standard design approval of the PPE design and First-of-a-Kind (FOAK) plant construction in Saudi Arabia will follow in due course.

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

March 1999	Conceptual design development
March 2002	Basic design development
June 2007	SMART-PPS (Pre-Project Service)
July 2012	Technology verification, Standard Design Approval (SDA)
March 2012	First step of Post-Fukushima corrections and commercialization
September 2015	Pre-project engineering agreement signed between Republic of Korea and Kingdom of Saudi Arabia for the deployment of SMART in the Gulf country
November 2015	Pre-Project Engineering started.
February 2019	The Pre-Project Engineering completed.
January 2020	SMART100 Standard Design Approval Applied.

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

BASIC TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Afrikantov OKBM JSC, Rosatom, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Water / Water
Thermal/electrical capacity, MW(t)/MW(e)	190 / 55
Primary coolant system circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.7 / 3.83 (steam pressure)
Core inlet/outlet coolant temperature (°C)	283 / 321
Fuel type/assembly array	UO_2 pellet/hexagonal assembly
Number of fuel assemblies in the core	199
Fuel enrichment (%)	< 20
Refueling cycle (months)	60 – 72
Core discharge burnup (GW·d/ton)	–
Reactivity control mechanism	CRD mechanisms of CPS
Approach to safety system	Combined active and passive safety systems
Design service life (years)	60
NPP footprint (m^2)	~ 60 000, for NPP with 2 reactors ~ 90 000, for NPP with 4 reactors
RPV height/diameter (m)	7.5 / 3.4
RPV weight (metric ton)	164
Seismic design (SSE)	9 points on the MSK-64 scale
Requirements for or approach to the operating cycle	Refueling every 5 – 6 years
Distinguishing features	Modular design; integral reactor; in-vessel corium retention in severe accidents
Design status	Detailed design

1. Introduction

The RITM-200 series reactors are the Afrikantov OKBM JSC’s 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 JSC-developed RITM-200 reactor plant are used in the RITM-200N reactor plant design for a pilot land-based small-sized nuclear power plant.

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

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Coolant System

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

(e) Steam Generator

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

(f) Pressurizer

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

5. Safety Features

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

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

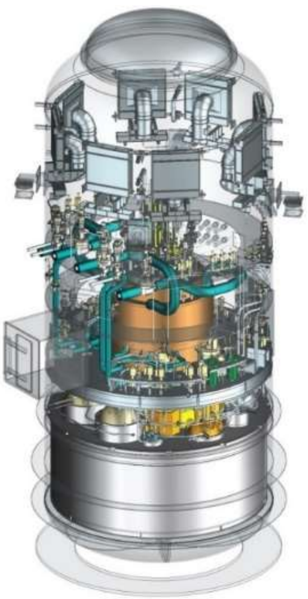
The high safety level of the RITM-series reactors is achieved through inherent safety features and through a combination of passive and active safety systems. In addition to that, it is provided that equipment and channels in the safety systems be redundant and functionally and/or physically separated to ensure high-level reliability. The control rods of the CPS fall into the core by gravity or driven by a spring when the power supply is disconnected from the electromagnetic couplings, which is followed by a reactor shutdown even in the case of a complete blackout.

(b) Residual Heat Removal System

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

(c) Emergency Core Cooling System (ECCS)

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



(d) Containment System

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

6. Plant Safety and Operational Characteristics

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

7. Instrumentation and Control System

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

8. General Layout of the Plant

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

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

9. Testing to Check and Validate the Design

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

10. Design and Licensing Status

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

11. Approach to the Operating Cycle

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

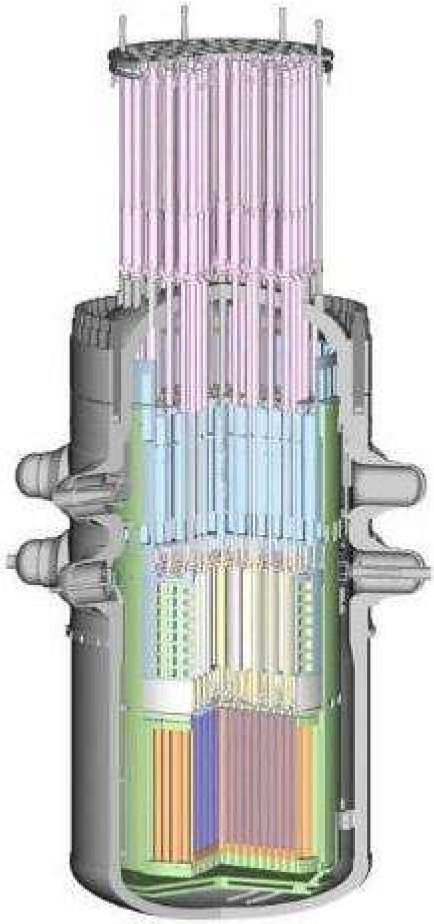
12. Waste Management System and Waste Disposal Plan

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

13. Development Milestones

2018	A conceptual design of the land-based small-sized NPP
2023	A site license for the small-sized NPP
2024	A construction license; start of the basic SMR period
2026	Operation license to be issued
2027	First electricity to be generated by the small-sized NPP

VK-300 (NIKIET, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Simplified passive BWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	750 / 250
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	6.9
Core Inlet/Outlet Coolant Temperature (°C)	190 / 285
Fuel type/assembly array	UO_2 pellet/hexahedron
Number of fuel assemblies in the core	313
Fuel enrichment (%)	4
Core Discharge Burnup (GWd/ton)	41.4
Refuelling Cycle (months)	72
Reactivity control mechanism	Rod insertions
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	40 000
RPV height/diameter (m)	13.1 / 4.535
RPV weight (metric ton)	325
Seismic Design (SSE)	Max 8 point of MSK-64
Fuel cycle requirements / Approach	Once through fuel cycle with UO_2
Distinguishing features	Innovative passive BWR based on operating prototype and well-developed equipment
Design status	Detailed design of reactor and cogeneration plant standard design

1. Introduction

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

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

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

(e) Reactor Coolant System

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

(f) Steam Generator

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

5. Safety Features

Innovative feature of the VK-300 project is the application of a metal lined primary containment (PC) of reinforced concrete. The PC helps to provide safety assurance, economically and reliably using structurally simple, passive safety systems.
The emergency cooldown tanks (ECTs) are located outside of the PC and are intended to function as accumulators and primary inventory make-up. If there is a line rupture and the pressure of the PC and reactor equalize, the ECTs actuate by gravity and fill the PC.
The residual heat is passively removed from the reactor by steam condensers located in the PC around the reactor that are normally flooded with the primary circuit water. When the level in the PC drops, the connecting pipelines to the condensers are opened, the reactor steam condenses and returns back to the reactor. The condensers are cooled with water from the ECTs.
At the same time the power unit design stipulates that the whole power unit will be within a leak-tight enclosure (the secondary containment). The containment accommodates the PC with the VK-300 reactor, emergency cooldown tanks, turbine, spent fuel storage pools, refuelling machine and central hall crane. The containment leak rate is 50% of the volume per day with the design pressure of not more than 0.15 MPa. Thanks to new

layout concepts for the main equipment of the VK-300 power unit, the containment dimensions do not exceed the dimensions of the VVER-1000 reactor containment.

(a) Engineered Safety System Approach and Configuration

The main technological solutions of VK-300:

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

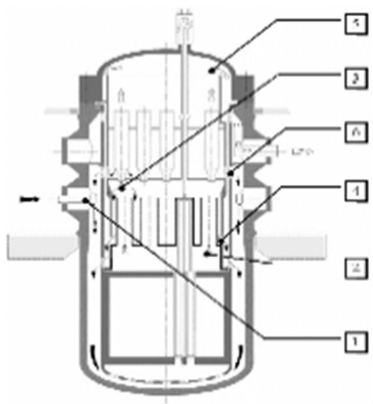
(b) Decay Heat Removal System

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

(c) Emergency Core Cooling System

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

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



(d) Containment System

The VK-300 reactor adopts a metal-lined primary containment system (PCS) of reinforced concrete. The PCS helps to solve the safety assurance problem economically and reliably using structurally simple passive safety systems. The PCS is rather small, with volume about 2000 m^3. The PCS of the VK-300 performs the functions of: (a) a safeguard reactor vessel; (b) a protective safety barrier limiting the release of radioactive substances during accidents with ruptures of steam, feedwater and other pipelines immediately near the reactor; and (c) providing the possibility of the emergency core cooling by the reactor cooling water making additional water inventory unnecessary.

6. Plant Safety and Operational Performances

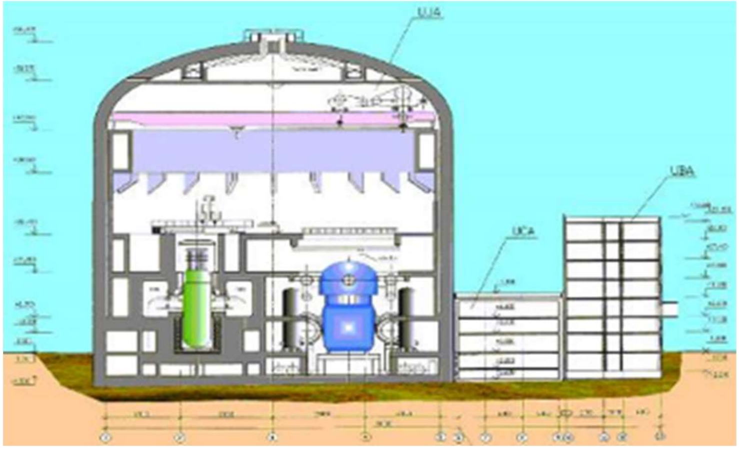
A set of reactor facility safety features and the concept of defense-in-depth against radioactivity escape allow plant location in the vicinity of a residential district limiting the control area around the VK-300 cogeneration plant by the dimensions of the cogeneration nuclear power plant (CNPP) site.

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement

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



(a) Reactor and Turbine Building

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

(b) Electric Power System

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

9. Design and Licensing Status

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

10. Fuel Cycle Approach

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

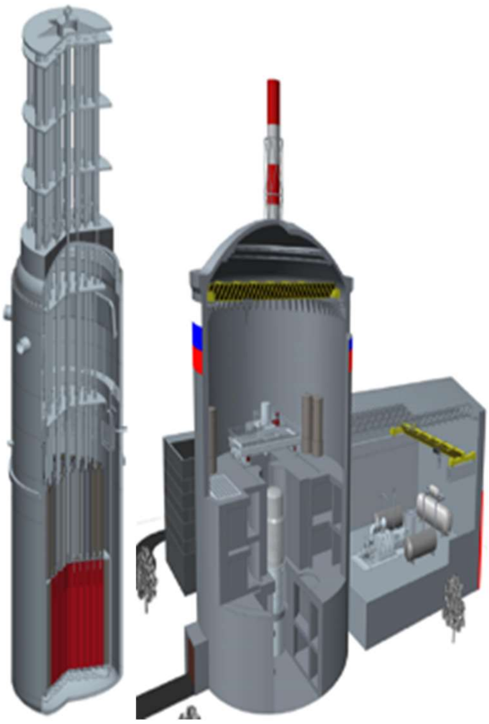
11. Waste Management and Disposal Plan

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

12. Development Milestones

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

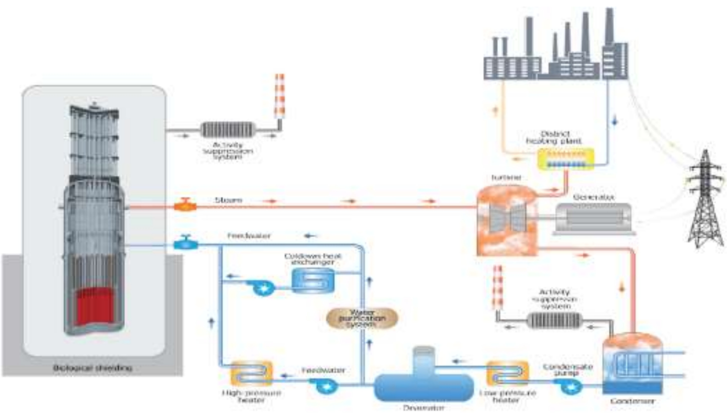
KARAT-45 (NIKIET, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	BWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	180 / 45-50
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	7.0 / –
Core Inlet/Outlet Coolant Temperature (°C)	180 / 286
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	109
Fuel enrichment (%)	4.5
Core Discharge Burnup (GWd/ton)	45.9
Refuelling Cycle (months)	84
Reactivity control mechanism	Control rods drive
Approach to safety systems	Passive
Design life (years)	80
Plant footprint (m^2)	9000
RPV height/diameter (m)	11.15 / 3.10
RPV weight (metric ton)	176
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Refueling (fuel shuffling) interval is up to 800 EFPD; Fuel assembly life cycle is about 6.6 years
Distinguishing features	Designed for extreme arctic and northern area conditions
Design status	Conceptual Design

1. Introduction

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



of Russian Federation.

2. Target Application

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

3. Design Philosophy

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

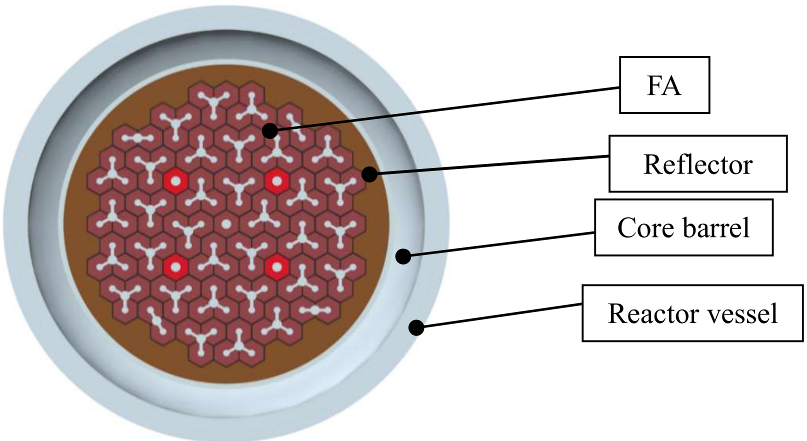
4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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



(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) accommodates the reactor core and the reactor internals, including two feedwater supply headers, two emergency heat exchangers and a block of louver-type separators. The reactor vessel has an elliptical bottom, cylindrical shells, and two nozzle and main connector flange shells welded one to the other. The RPV outer diameter is 3100 mm, wall thickness 100 mm, and height 11 150 mm.

(e) Reactor Coolant System

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

(f) Steam Generator

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

(g) Pressurizer

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

One of the major principles of safety systems design is the requirement that they should operate at any design-basis initiating event and during failure of any active or passive component with mechanical parts independently on the initiating event (single failure principle). The safety system design also meets the requirement for the systems to perform its functions automatically and reliably with the smallest possible number of active elements involved and using the passive protection principle.

(b) Decay Heat Removal System

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

(c) Emergency Core Cooling System

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

(d) Containment System

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

6. Plant Safety and Operational Performances

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

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

7. Instrumentation and Control Systems

In-core monitoring system is designed for monitoring of thermal-hydraulic and neutron properties of the

reactor core and in-core coolant natural circulation flow which are measured directly or indirectly in different operating modes of the reactor.
The following parameters are expected to be monitored in the KARAT-45 reactor:
- Continuous monitoring of neutron parameters defining the reactor period, neutron power and the control rod position;
- Continuous monitoring of thermal parameters defining the reactor's thermal power, the reactor water level, and temperature of water at FA inlet, in steam space and at the reactor vessel surface;
- Periodic water chemistry control;
- Periodic inspection of the thermal reliability of the reactor core operation. In-core power density field monitoring.

8. Plant Layout Arrangement

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

9. Design and Licensing Status

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

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

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

12. Development Milestones

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

KARAT-100 (NIKIET, Russian Federation)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	BWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	360 / 100
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	7.0
Core Inlet/Outlet Coolant Temperature (ºC)	104 / 286
Fuel type/assembly array	UO_2 pellet / hexagonal
Number of fuel assemblies in the core	199
Fuel enrichment (%)	4
Core Discharge Burnup (GWd/ton)	45.9
Refuelling Cycle (months)	90
Reactivity control mechanism	Control rods drive mechanism
Approach to safety systems	Passive
Design life (years)	80
Plant footprint (m^2)	22 500
RPV height/diameter (m)	13.25 / 4.00
RPV weight (metric ton)	348
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	Refueling (fuel shuffling) interval is up to 900 EFPD; Fuel assembly life cycle is about 7.5 years
Distinguishing features	Multi-purpose reactor: cogeneration of electricity and heat
Design status	Conceptual design

1. Introduction

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

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

The reactor uses a single-circuit heat removal system. Steam is generated by the coolant boiling in the reactor core. The steam discharge system is designed to remove steam from the reactor directly to the turbine plant. The steam pressure in the system is 7 MPa, at the steam temperature of 286 ^{0}C. The humidity of the steam fed to the turbine plant is 0.1%. KARAT-100 uses gravity-type steam separation with additional drying in louver-type separators without centrifugal axial separators, which improves the stability of the reactor operation.

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

(f) Pressurizer

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

KARAT-100's system of engineered and organizational measures forms five defence-in-depth levels:
- Conditions for KARAT-100 siting and prevention of anticipated operational occurrences;
- Prevention of design-basis accidents by normal operation systems;
- Prevention of beyond design-basis accidents by safety systems;
- Management of beyond design-basis accidents;
- Emergency planning.

(b) Decay Heat Removal System

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

(c) Emergency Core Cooling System

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

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

Additionally, the power unit with KARAT-100 reactor is equipped with the following safety systems:
- A steam localization system downstream of safety valves required to localize radioactive steam release when the safety valves actuate;
- The system for the return of boron solution into the reactor designed to feed the borated coolant back from the reactor cavity in the event of a reactor vessel or nozzle break;
- The reactor water accumulation system aimed at keeping the water inventory in the accumulators for making up the reactor in emergencies caused by a decrease in the reactor vessel coolant level;
- The emergency power supply system used in the event of a loss of power supply from the energy grid.

(d) Containment System

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

6. Plant Safety and Operational Performances

The major goal of the safety assurance arrangements is to limit KARAT-100 radioactive impacts on the personnel, the public and the environment during normal operation, and in cases of operational occurrences and emergency events.

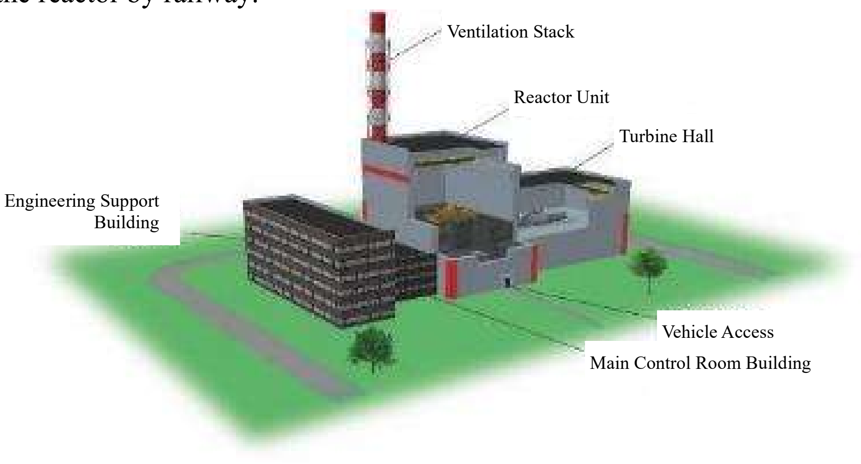
KARAT-100 safety is ensured through the specific transfer and distribution of radioactive substances due to water boiling. The key factors are:
- A high inter-phase barrier (water-steam) to the spreading of nongaseous radionuclides prevents these from entering the steam-condensate line;
- Continuous degassing of coolant and removal of gaseous fission products from the circuit limit their accumulation in the circuit.

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement

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



(a) Reactor Building

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

(b) Control Building

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

(c) Balance of Plant

(i).Turbine Generator Building
The turbine block houses the turbine generator, the steam condensate components and equipment, and a bridge crane for moving operations. The dimensions of the turbine block are 42 m x 28 m, and its height is 28.4 m.

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

9. Design and Licensing Status

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

10. Fuel Cycle Approach

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

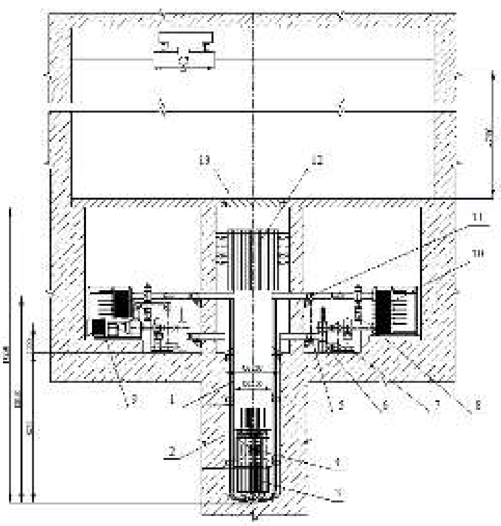
11. Waste Management and Disposal Plan

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

12. Development Milestones

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

RUTA-70 (NIKIET, Russian Federation)



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

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Pool-type
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	70 / NA
Primary circulation	Natural (below 30% of rated power)/forced (for 30-100% of rated power)
NSSS Operating Pressure (primary/secondary), MPa	Atmospheric pressure at reactor poll water surface
Core Inlet/Outlet Coolant Temperature (°C)	75 / 102
Fuel type/assembly array	Cermet (0.6 UO_2+ 0.4 Al alloy) / hexagonal
Number of fuel assemblies in the core	91
Fuel enrichment (%)	3.0
Core Discharge Burnup (GWd/ton)	25-30
Refuelling Cycle (months)	36
Reactivity control mechanism	Control rods
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	100 000
RPV height/diameter (m)	17.25 / 3.20
Seismic Design (SSE)	> 0.8g (automatic shutdown)
Fuel cycle requirements / Approach	Once-through fuel cycle with UO_2 fuel
Distinguishing features	Design capable for radioisotopes production for medical, neutron beams for neutron therapy and industrial purposes
Design status	Conceptual design

1. Introduction

RUTA-70 is a multi-purpose water-cooled water-moderated integral pool-type reactor serving as a Nuclear Heating Plant (NHP) of 70 MW(t) thermal capacity for district heating, desalination and radioisotopes production for medical and industrial purposes. It has no power conversion system. In the primary cooling circuit, the heat from the core is transferred to the primary heat exchanger (HX) by forced convection at full power and by natural convection at power below 30% of the rated power. Forced coolant circulation using pumps for operations at power levels of 30% to 100% rated power increases the coolant flow rate in the primary circuit and raises the down-comer temperature. RUTA-70 can perform continuous operation without any maintenance for about one year. RUTA-70 reactors can be located in the immediate vicinity of the heat users.

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

RUTA-70 has a two-circuit layout. The primary circuit is an in-pile reactor core cooling circuit and the secondary circuit is an intermediate one that removes heat from the reactor and transfers it to the third circuit, which is the consumer circuit, i.e., to the heating network. Most of the plant equipment, including the primary-to-secondary side heat exchangers (HX-1/2) resides at dry boxes outside the pool.

(b) Reactor Core

The reactor core is placed in the lower part of the reactor vessel, the vault, in the shell of the chimney section. The core is designed with the 'Cermet' fuel rods that contribute to the reactor safety due to a high thermal conductivity of the fuel matrix and its role as the additional barrier to the fission products release. The reactor core consists of 91 fuel assemblies (FA) of hexagonal geometry with 120 fuel rods per each FA. The height of reactor core is 1400 mm or 1530 mm depending on the fuel rod type. The core equivalent diameter is 1420 mm. In the radial direction, the design of the RUTA-70 fuel assembly is similar to that of the VVER-440 fuel assembly.

(c) Reactivity Control

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

(d) Reactor Coolant System

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

(e) Steam Generator

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

(f) Primary Cooling Mechanism

The heat from the core to the primary heat exchanger (HX) is transferred by forced convection of the primary water coolant at full power operation but by natural convection under operation conditions below 30% of the rated power. The application of forced coolant circulation using pumps for operations at power levels of 30% to 100% rated power increases the coolant flow rate in the primary circuit and raise the down-comer temperature by reducing the water thermal gradient in the reactor core. The distributing header is placed in the upper part of the shell of the chimney section. Pipelines of water supply to the primary HXs are connected to the header from both sides. Downstream of the HXs, coolant is directed via the suction header to the circulation pump that supplies water to a group of heat exchangers located at one side of the pool. Water is returned from the pump head via the supply header. Pumps are connected to the bypass line of the natural convection circuit and are placed in a special compartment in close vicinity to the reactor pool.

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

The RUTA-70 uses mostly passive systems to perform safety functions such as: air heat sink system for emergency cooldown (ASEC), gravity driven insertion of the control rods in the core as reactor safety control system, the secondary circuit overpressure protection system, the overpressure protection system for air space in the reactor pool and pre-stressed concrete external impacts protection system. In case of multiple failures in the reactivity control systems and devices, safety can be ensured by self-control of reactor power (boiling -

self-limitation of power), i.e. through the inherent safety features of the reactor. There is a stabilisation of reactivity feedbacks determined by negative fuel and coolant temperature reactivity coefficients and by the positive density reactivity coefficient.

(b) Decay Heat Removal System

Natural circulation in the secondary circuit provides for residual heat removal from the shutdown reactor and passive cooldown of the reactor facility in blackout emergency situation. The passively actuated ASEC provides residual heat removal to the ultimate heat sink (atmospheric air). ASEC is envisaged for reactor cooldown in case of loss of auxiliary power. Each loop of the secondary circuit has an ASEC subsystem (train); the ASEC is connected at the bypass line of the network heat exchangers. If all controlled trains of heat removal are lost, heat losses via the external surface of the reactor pool to the surrounding environment (ground) are considered as an additional safety train. Residual heat is accumulated in the pool water. The transient of pool water heat-up in the aqueous mode before the onset of boiling takes several days. As soon as boiling starts, steam goes to the reactor hall where it is condensed by passive condensing facilities. A reactor boil-off without makeup takes 18 to 20 days. Upon completion of this period residual heat is balanced by heat transfer to the ground. Core dry out is avoided. Moderate temperatures are not exceeding the design limits characterize fuel elements.

(c) Emergency Core Cooling System

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

(d) Reactor Pool

The reactor pool consists of reactor core and internals, control and protection system, distributing and collecting headers and a large amount of water. Big amount of water in the reactor pool ensures slow changing of coolant parameters and reliable heat transfer from the fuel, even if controlled heat transfer from the reactor is not available. Fuel temperatures are moderate.

(e) Containment System

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

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control Systems

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

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

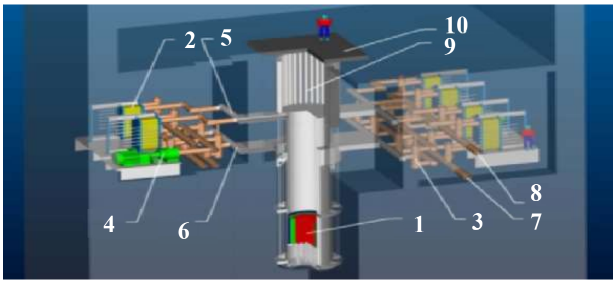
In the core there are 42 reactor control and protection system (RCP) rods composing two shutdown systems with diverse actuators. One of these systems intended specifically for core emergency protection (EP) includes 12 rods. The second shutdown system performing the concurrent functions of shutdown and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

8. Plant Layout Arrangement

(a) Reactor Building

The protective flooring composed of slabs is installed above the reactor pool to avoid possible damage to the primary components from external impacts. To prevent gas and vapour penetration to the reactor hall from the upper part of the reactor, joints of the protective slabs are gas-tight.

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



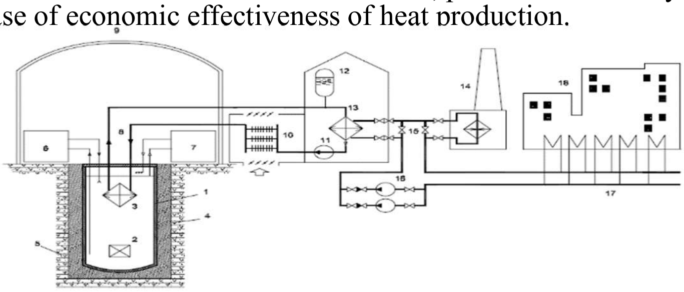
(b) Control Building

The smallest staffing of the operating shift is four persons. These are the NDHP shift supervisor, the chief reactor control engineer, a fitter-walker for normal operation systems and a duty electrician to attend to electrical devices and systems, instrumentation and control. A supervising physician and a refuelling operator are added to the regular shift staff for the core fuelling, first core critical mass attaining, power start up and refuelling periods. The total personnel number including regular engineers, technicians and administrative staff may reach up to 40 persons.

(c) Balance of Plant

The turbine and associated systems are not used in the NHP RUTA-70. Such scheme in spite of some reduction of autonomy of district heating system (in comparison to high-temperature reactors) possesses several major advantages: Increase in reliability of heat supplying due to diversification of heat sources, provide redundancy required by relatively cheap heat sources and increase of economic effectiveness of heat production.

1. Reactor pool, 2. Reactor Core
3. Primary heat exchanger, 4. Concrete vessel
5. Soil, 6. Purification system,
7. Ventilation system, 8. Secondary circuit
9. Containment, 10. Residual heat removal system
11. Secondary circuit circulation pump,
12. Secondary circuit pressurizer,
13. Secondary heat exchanger
14. Peak/backup heat source, 15. Control valves
16. Grid circulation pumps, 17. Grid water, 18. Consumers



9. Design and Licensing Status

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

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

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

12. Development Milestones

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

STAR (STAR ENERGY SA, Switzerland)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	STAR ENERGY SA Switzerland
Reactor type	Pressure tube light water reactor
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	30 / 10
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	12.5 / 4.4
Core inlet/outlet coolant temperature (°C)	270 / 300
Fuel type/assembly array	UO_2 pellet assemblies in individual pressure channels
Number of fuel assemblies in the core	138
Fuel enrichment (%)	19
Refuelling cycle (months)	120
Core discharge burnup (GWd/ton)	60
Reactivity control mechanism	Control rods and Er / Gd_2O_3 burnable absorber
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m²)	36 000
RPV height/diameter (m)	7.6 / 2.3
RPV weight (metric ton)	20
Seismic design (SSE)	Degree VIII (MSK-64)
Fuel cycle requirements/approach	Once-through HALEU; no on-site storage
Distinguishing features	Compact, low-weight design without a pressure vessel; flexibility of operation; intrinsic passive safety
Design status	Basic design

1. Introduction

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

2. Target Application

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

3. Design Philosophy

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

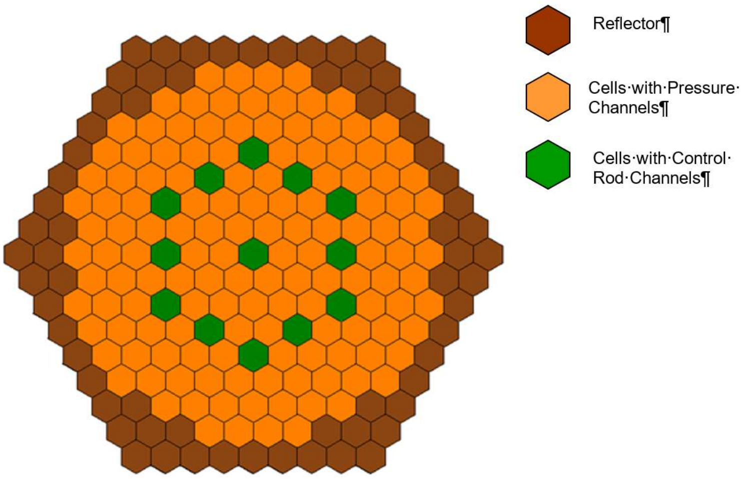
4. Main Design Features

(a) Power Conversion

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

(b) Reactor Core

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



Reactor core layout

(c) Fuel Characteristics

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

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

(d) Reactivity Control

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

(e) Reactor Pressure Vessel and Internals

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

(f) Reactor Coolant System

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

(g) Secondary System

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

(h) Steam Generator

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal System / Reactor Cooling Philosophy

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

(c) Spent Fuel Cooling Safety Approach / System

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

(d) Containment System

The design's containment system is based on the "defence in depth" philosophy and includes a set of technical and administrative measures aimed at preventing environmental exposure to radiation and radioactive substances from the reactor.
The set of barriers includes:
- Fuel pellet matrix;
- Fuel rod casings;
- Pipelines end equipment of primary coolant circuit;
- Reactor core pressure channels;
- Reactor core shielding;
- Airtight reactor vessel;
- Ferrocement reactor shaft.

(e) Chemical Control

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

6. Plant Safety and Operational Performances

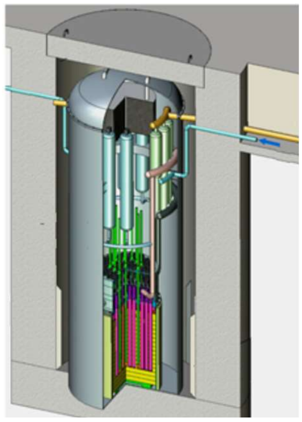
From the beginning of development of "STAR", reasonable safety requirements to be imposed for the reactor's design features are as follows:
- Core damage frequency – less than 10^{-6}/reactor year
- Large early radioactivity release frequency – less than 10^{-7}/reactor year

7. Instrumentation and Control System

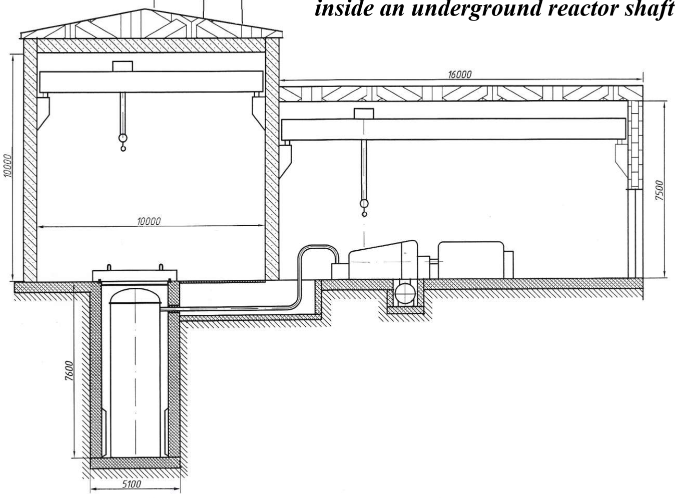
On the conceptual level, the I&C system is divided into the following main blocks:
- Primary and reserve control rooms;
- Reactor control and protection system, including remote controls and two independent emergency shutdown systems;
- Process parameters control system;
- Internal reactor control system;
- Normal operation control systems;
- Miscellaneous automatic systems.

8. Plant Layout Arrangement

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



"STAR" module placement option inside an underground reactor shaft



Example layout of a reactor building with an adjacent secondary island

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

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

11. Fuel Cycle Approach

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

12. Waste Management and Disposal Plan

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

13. Development Milestones

2015	Initial concept developed; patent priority date secured
2019	Concept submitted for peer review
2021	Conceptual design developed; preliminary feasibility study for a FOAK unit completed
Current (2022)	Conduction of feasibility study for FOAK unit and initiation of licensing procedures with local regulator and basic design
2024	Development of basic design and PSAR, conduction of site survey
2025	Launch of FOAK plant project, development of detailed design and extended SAR
2026-2027	Pilot unit installation and launch

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

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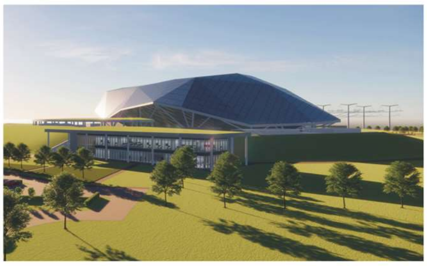
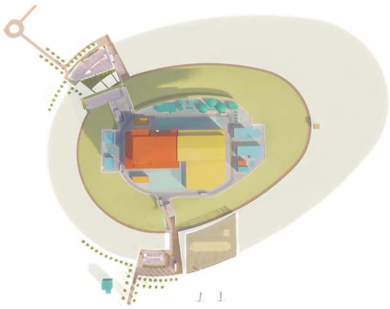




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Rolls-Royce SMR Ltd, United Kingdom
Reactor type	3-loop PWR
Coolant/moderator	Light water
Thermal/electrical capacity, MW(t)/MW(e)	1 358 / 470
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.5 / 7.8
Core inlet/outlet coolant temperature (°C)	295 / 325
Fuel type/assembly array	UO_2 / 17 × 17 Square
Number of fuel assemblies in the core	121
Fuel enrichment (%)	<4.95
Refuelling cycle (months)	18
Core discharge burnup (GWd/ton)	50 – 60
Reactivity control mechanism	Control rods
Approach to safety systems	Passive and Active
Design life (years)	60
Plant footprint (m^2)	40 000
RPV height/diameter (m)	7.9 / 4.2
RPV weight (metric ton)	150
Seismic design (SSE)	>0.3g
Fuel cycle requirements/approach	Open cycle
Distinguishing features	Modular approach facilitating rapid and cost-effective build.
Design status	Detailed design

1. Introduction

The Rolls-Royce SMR has been developed to deliver a market driven, affordable, low carbon, energy generation capability. The developed design is based on optimised and enhanced use of proven technologies that presents a class leading safety outlook and attractive market offering with minimum regulatory risk. Rapid, certain and repeatable build is enhanced through site layout optimisation and maximising modular build, standardisation and commoditisation.

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

The Rolls-Royce SMR is a close-coupled 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. All poison pins in a single fuel assembly have the same gadolinia loading.

(c) Reactivity Control

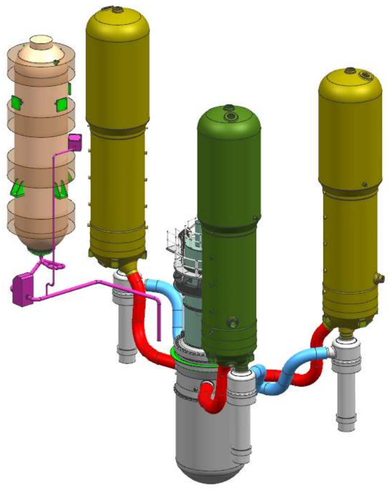
Duty reactivity control is provided through movement of control rods and use of the negative moderator temperature coefficient inherent to Pressurised Water Reactors. No concentration of soluble boron is maintained in the primary coolant for duty reactivity control, which facilitates a simplified plant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

(d) Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel comprises the upper shell containing the six Reactor Coolant Loop nozzles, three Direct Vessel Injection nozzles, the lower shell consisting of a plain cylindrical forging with no circumferential welds present within the high flux region of the core and the torispherical lower head.
The Integrated Head Package forms an assembly of the reactor head area components. It comprises the Reactor Pressure Vessel Closure Head with torispherical profile 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

The Reactor Coolant System is a three-loop close-coupled configuration. Steam Generators are located around the circumference of the Reactor Pressure Vessel, with short close-coupled pipework connections between them. The pressuriser is connected to the pipework hot leg. A centrifugal Reactor Coolant Pump is mounted directly to the outlet nozzle of the corresponding 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.



Reactor Coolant System

(f) Steam Generator

A vertical u-tube Steam Generator design has been selected as a mature and readily deployable technology. 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.

(g) Pressuriser

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

(h) Primary pumps

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The Rolls-Royce SMR design has been developed through a combined system engineering and safety assessment approach. The safety informed design supports the process by which risks are demonstrated to be acceptable and as low as reasonably practicable. Defence in depth is provided through multiple layers of fault prevention and protection in the form of independent and diverse active and passive systems, with multiple trains per system. Passive safety systems are designed to deliver their safety functionality autonomously for 72 hours, minimising the demand on human actions and electrical power.

(b) Decay Heat Removal System

The Rolls-Royce SMR employs both active and passive decay heat removal systems. Condenser Decay Heat Removal utilises the Steam Generators and the normal duty steam condenser to cool the primary plant. It utilises most of the same equipment used for steam condensing during power operation. Heat is eventually passed to the tertiary heat sink. Passive Decay Heat Removal is a dedicated safety system that utilises the Steam Generators and the Local Ultimate Heatsink system to cool the primary plant. Both means of heat rejection are 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.

(c) Emergency Core Cooling System

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 Vessel pipework associated with gravity drain from the Refuelling Pool to the Reactor Vessel and three containment sump screens. 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.

(d) 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.

(e) 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.

6. Plant Safety and Operational Performances

Plant conditions have been analysed using industry validated codes to demonstrate significant safety margin across the levels of defence in depth. The Probabilistic Safety Assessment calculates a core damage frequency from plant faults of <1E-07 per year of power operation, and that no single fault or class of faults makes a disproportionate contribution to risk, i.e. a balanced risk profile is achieved. Internal and external hazard assessments have defined the design basis and informed the plant layout to optimise segregation. Key equipment is protected by the hazard shield which is resilient against external hazards including aircraft impact and tsunami.

The Rolls-Royce SMR is designed for full compliance with the U.K. Grid Code, including frequent load following between 50% and 100% power at a rate of 3-5% per minute with a design target to allow operation at lower power e.g. house load operation (stable operation for at least two hours supplying only the power station's house load).

A 15-day outage period is targeted to support high availability (at least 90% availability over the 60-year life of the Rolls-Royce SMR), including a target period of 8 days for refuelling.

7. Instrumentation and Control System

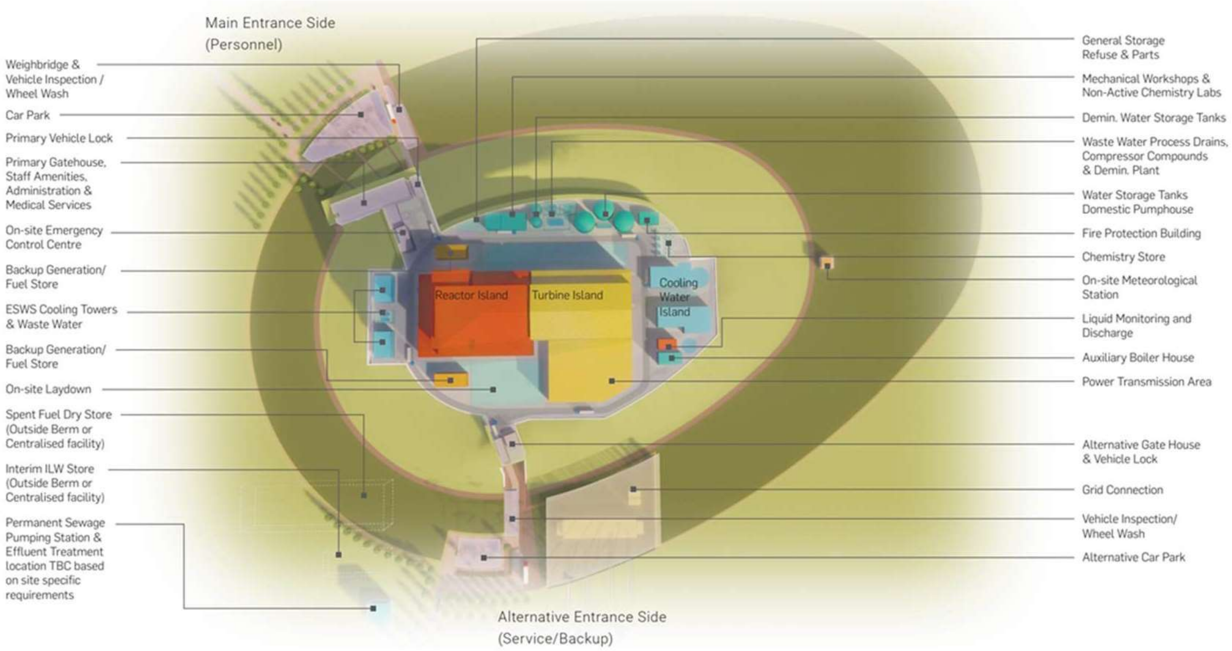
The Rolls-Royce SMR plant is controlled and protected by a number of control and instrumentation systems. The Reactor Plant Control System manages duty operations and uses an available in industry programmable logic controller or distributed control system. It uses mixed analogue and non-programmable digital sensors and communicates on hardwired multichannel digital electrical networks. Opportunities to use smart devices and wireless technologies are being pursued.

The Reactor Protection System provides shutdown in response to a fault. The Reactor Protection System contains priority logic, which from the range of input signals received determines whether to initiate reactor shutdown. The Hardwired Diverse Protection System uses non-programmable electronics and as such provides a diverse means to shut down the plant in response to fault conditions.

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

8. Plant Layout Arrangement

The power station is designed for installation on an extensive range of inland and coastal sites, across a wide range of soil and earth conditions, whilst maintaining a compact site footprint of approximately 40,000 m2. This flexibility is enabled through design features such as seismic isolation for key safety areas. The nuclear reactor is located in the Reactor Island, adjacent to Turbine Island with the Cooling Water Island following. Support buildings and auxiliary services are situated within a berm that sweeps around the site and provides a layer of protection from external hazards.



Plant Layout

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 targets completion in time for construction of the first of a kind power station to commence in 2026.

11. Fuel Cycle Approach

The Rolls-Royce SMR operates on an 18-month fuel cycle, with a three-batch equilibrium core. Refuelling is managed through the provision of an in-containment refuelling pool which temporarily stores both new and used fuel during a refuelling outage. Used fuel is subsequently transferred to an external spent fuel pool for storage prior to transfer to long term dry cask storage.

12. Waste Management and Disposal Plan

The Rolls-Royce SMR waste treatment systems are based on use of proven technologies and best available techniques. Industry lessons learned and good practices have been used in the development of systems to minimise active and non-active wastes and discharges, through both design and operational practices adopted. Standardised waste treatment system components and modules are used to achieve the flexibility required for the waste informed design. Operation without soluble boron in the primary coolant allows significant reduction in environmental discharges and concomitant simplification of the waste treatment systems.

13. Development Milestones

2015	Rolls-Royce development of initial reference design
2016	Formation of consortium for design of whole power station concept
2017	Mature design concept developed
2022	Formal regulation entered in the UK
2026	Projected start of first of a kind construction
2030	Planned first of a kind commercial operation

VOYGR™ (NuScale Power Corporation, United States of America)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NuScale Power Corporation, USA
Reactor type	Integral PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	250 / 77 (gross)
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	13.8 / 4.3
Core Inlet/Outlet Coolant Temperature (°C)	249 / 316
Fuel type/assembly array	UO_2 pellet / 17x17 square
Number of fuel assemblies in the core	37
Fuel enrichment (%)	≤ 4.95
Core Discharge Burnup (GWd/ton)	≥ 45
Refuelling Cycle (months)	Nominal 18
Reactivity control mechanism	Control rod drive, boron
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	140 000 (VOYGR™-12)
RPV height/diameter (m)	17.7 / 2.7
RPV weight (metric ton)	TBC
Seismic Design (SSE)	0.5g
Fuel cycle requirements / Approach	Nominal three-stage in-out refuelling scheme
Distinguishing features	Unlimited time for core cooling without AC or DC power, or water addition, or operator action
Design status	Equipment Manufacturing in Progress

1. Introduction

The NuScale Power Module™ (NPM) is a small, light-water-cooled pressurized-water reactor (PWR). NuScale VOYGR™ SMR plants are scalable and can be built to accommodate a varying number of NPMs to meet customer's energy demands. NuScale standard plants include the VOYGR-4 at 308 MW(e), the VOYGR-6 at 462 MW(e), and the VOYGR-12 at 924 MW(e). A six-module configuration is the reference plant size for the Standard Plant Design Approval design and licensing activities. Each NPM is a self-contained module that operates independently of the other modules in a multi-module configuration. All modules are managed from a single control room.

2. Target Application

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

3. Design Philosophy

The NuScale VOYGR SMR plant design adopts design simplification, proven light-water reactor technology, modular nuclear steam supply system, factory-fabricated power modules, and passive safety systems that allow for unlimited reactor cooling time after a beyond design basis accident without AC or DC power, operator action, or makeup water. There are no design basis accidents that uncover the core or require operator action.

The NPM is designed to operate efficiently at full-power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps.

4. Main Design Features

(a) Nuclear Steam Supply System

The nuclear steam supply system (NSSS) consists of a reactor core, helical coil steam generators, and a pressurizer within a reactor pressure vessel (RPV). The NSSS is enclosed in an approximately cylindrical containment vessel (CNV) that sits in the reactor pool structure. Each power module is connected to a dedicated turbine-generator unit and balance-of-plant systems.

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

The RPV consists of a cylindrical steel vessel with an inside diameter of 2.7 m, an overall height of approximately 17.7 m, and is designed for an operating pressure of 13.8 MPa. The upper and lower heads are torispherical and the lower portion of the vessel has flanges just above the core region to provide access for refuelling. The RPV upper head supports the control rod drive mechanisms. Nozzles on the upper head provide connections for the reactor safety valves, the reactor vent valves, and the secondary system steam piping.

(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 operational flows. 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.

(f) Pressurizer

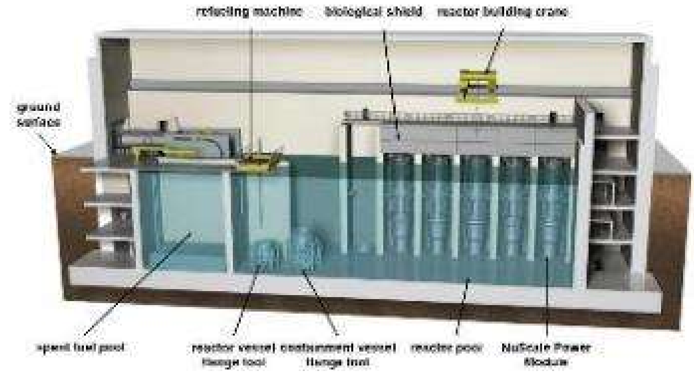
The internal pressurizer provides the primary means for controlling reactor coolant system pressure. It is designed to maintain a constant reactor coolant pressure during operation. Reactor coolant pressure is increased by applying power to a bank of heaters installed above the pressurizer baffle plate.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

5. Safety Features

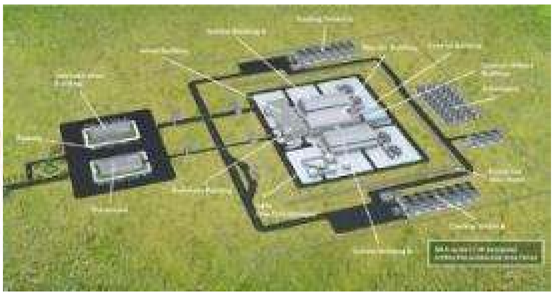
The NuScale VOYGR SMR plant adopts a set of engineered safety features designed to provide reliable long-term core cooling under all conditions, including severe accident mitigation. They include integral primary system configuration, a containment vessel, passive heat removal systems, and severe accident mitigation features. This fully passive safety design is rigorously proven by the NuScale Triple Crown for Nuclear Plant Safety™, which ensures that reactors will safely shut down and self-cool, indefinitely, and do so with no need for operator or computer action, AC or DC power, or the addition of water—a first for light water reactor technology.

(a) Engineered Safety System Approach and Configuration

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



Cut-away view of NuScale power plant


Site Layout of 12 module VOYGR-12 plant

(b) Safety Approach and Configuration to Manage DBC

The decay heat removal system (DHRS) provides secondary side reactor cooling for non-LOCA events when normal feedwater is not available. The system is a closed-loop, two-phase natural circulation cooling system. Two trains of decay heat removal equipment are provided, one attached to each steam generator loop. Each train is capable of removing 100% of the decay heat load and cooling the primary coolant system. Each train has a passive condenser immersed in the reactor pool.

(c) Safety Approach and Configuration to Manage DEC

Emergency core cooling system (ECCS) consists of three independent reactor vent valves (RVVs) and two independent reactor recirculation valves (RRVs). The ECCS provides decay heat removal in the unlikely event of a loss of feedwater flow, combined with the loss of both trains of the DHRS. The ECCS removes heat and limits containment pressure by steam condensation on, and convective heat transfer to, the inside surface of the CNV.

(d) Containment System

The functions of CNV are to contain the release of radioactivity following postulated accidents, protect the RPV from external hazards, and to provide heat rejection to the reactor pool following ECCS actuation. Each CNV consists of a steel cylinder with an outside diameter of 4.5 m and an overall height of 23.1 m. The CNV houses the RPV, control rod drive mechanisms, and associated piping and components of the NSSS. The CNV is immersed in the reactor pool, which provides an assured passive heat sink for containment heat removal under LOCA conditions.

(e) Spent Fuel Cooling Safety Approach

The used fuel pool provides storage for up to 10 years of used fuel storage, plus temporary storage for new fuel assemblies. The pool is connected to the ultimate heat sink, and hence, protected by the reactor building. The pool water volume provides approximately 150 days of passive cooling of the used fuel assemblies following a loss of all electrical power without the need for additional water. The VOYGR 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. After inspecting the module sections and refuelling the core, the module is reassembled and moved to its operations bay and reconnected to steam and feedwater lines. Other modules in the plant continue to operate while one module is refuelled. The nominal plant capacity factor is 95% with a refuelling outage time of 10 days.

7. Instrumentation and Control System

The NuScale design includes a fully digital control system based on the use of field programmable gate array (FPGA) technology. The highly integrated protection system (HIPS) platform, approved by the NRC, is based on the fundamental I&C design principles of independence, redundancy, predictability, repeatability, and diversity. The HIPS platform is comprised of four module types that can be interconnected to implement multiple configurations to support various types of reactor safety systems. The FPGA technology is not vulnerable to cyber-attacks.

8. Plant Layout Arrangement

(a) Reactor Building

The NuScale VOYGR SMR plant consists primarily of a reactor building, a control room building, two turbine-generator buildings, a radwaste treatment building, forced-draft cooling towers, a switchyard, and a dry-cast storage area for discharged fuel. The reactor building consists of 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.

(b) Control Building

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

(c) Balance of Plant

A NuScale VOYGR SMR plant has two separate turbine buildings, each housing up to six turbines and air-cooled generators. 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 has designed, built, and operates a one-third scale prototype testing facility that is used to test the NPM and gather data for thermal hydraulic codes, safety analysis code, and reactor design validation. NuScale has 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 has an average Technology Readiness Level (TRL) of 8. NuScale testing programs have been audited by the NRC.

10. Design and Licensing Status

In December 2016, NuScale Power submitted the Design Certification Application (DCA) to the NRC. 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. NuScale is applying for a power uprate to 250 MWt in 2022, with an expected NRC review completion in 2024. The first plant will be a 6-module VOYGR-6 plant located near Idaho Falls, Idaho with owner, Utah Associated Municipal Power Systems (UAMPS). The target commercial operation date for the first plant is 2029.

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 are customer driven optimization requirements.

12. Waste Management and Disposal Plan

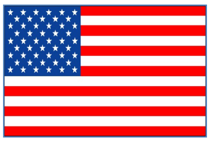
Removed assemblies are stored in the used fuel pool for initial cool-down and later moved to an on-site dry-cask storage. The plant design includes sufficient on-site storage space for all the spent fuel produced during the 60-year life of the plant. Final disposal is expected to be in a national fuel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

13. Staffing

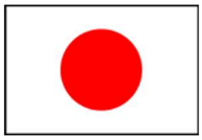
A VOYGR-12 plant will require a minimum of 3 licensed operators per shift in the control room and is estimated to require 270 plant employees for normal operation and maintenance. No Shift Technical Advisor is required in the U.S. for this design.

14. Development Milestones

2003	Initial concept developed and integral test facility operational
2007	NuScale Power, LLC was formed
2008	Initiation of NRC Pre-Application
2011	NuScale Power is acquired by Fluor
2013	U.S. DOE SMR Cooperative Agreement signed
2016	Design certification application was submitted to the U.S. NRC
2020	Standard Design Approval (SDA) received from the NRC in September 2020
2022	NPM production began with the production of forging dies for the Upper RPV
2023	Combined License Application (COLA) to be submitted to the NRC for first plant
2025	Construction targeted to start for first VOYGR-6 plant in Idaho Fall, Idaho, USA
2029	First commercial NuScale VOYGR-6 plant targeted to be operational in Idaho



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



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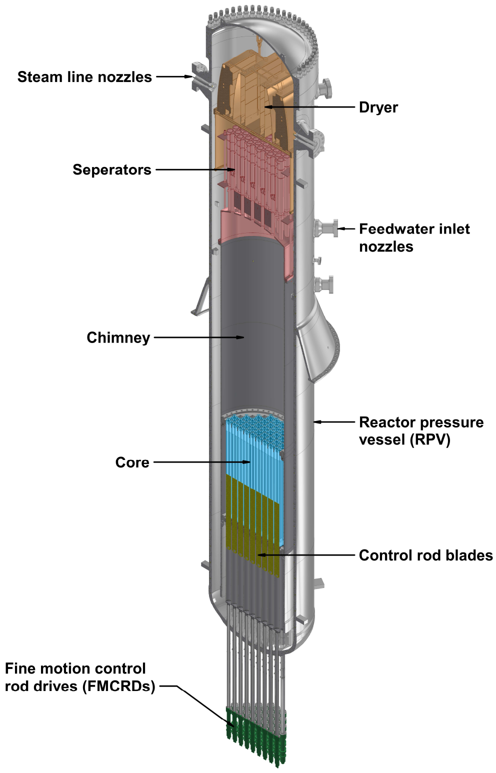


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	GE-Hitachi Nuclear Energy, United States and Hitachi-GE Nuclear Energy, Japan
Reactor type	Boiling water reactor
Coolant/moderator	Light-water / light-water
Thermal/electrical capacity, MW(t)/MW(e)	870 / 270 – 290
Primary circulation	Natural circulation
NSSS operating pressure (primary/secondary), MPa	7.2 / Direct cycle
Core inlet/outlet coolant temperature (ºC)	270 / 288
Fuel type/assembly array	UO_2 / 10×10 array
Number of fuel assemblies in the core	240
Fuel enrichment (%)	3.81 (avg) / 4.95 (max)
Refuelling cycle (months)	12 - 24
Core discharge burnup (GWd/ton)	49.6
Reactivity control mechanism	Rods and solid burnable absorber (B_4C, Hf, Gd_2O_3)
Approach to safety systems	Fully passive
Design life (years)	60
Plant footprint (m^2)	9 800
RPV height/diameter (m)	26 / 4
RPV weight (metric ton)	485
Seismic design (SSE)	0.3g
Fuel cycle requirements/approach	Open fuel cycle utilizing standard BWR fuel
Distinguishing features	Natural circulation BWR, integral RPV isolation valves, isolation condenser
Design status	Detailed design

1. Introduction

GE-Hitachi Nuclear Energy's (GEH's) BWRX-300 is a designed-to-cost 300 MWe water-cooled natural circulation small modular reactor (SMR) utilizing simple natural phenomena-driven safety systems. It is the tenth generation of the Boiling Water Reactor (BWR) and represents the simplest BWR design since GE began developing nuclear reactors in 1955. BWRX-300 is an evolution of the U.S. Nuclear Regulatory Commission (NRC) licensed 1,520 MWe Economic Simplified Boiling Water Reactor (ESBWR).

2. Target Application

Target applications include base load electricity generation, load following electrical generation within a range of 50 to 100% power, district heating, synthetic fuel production, hydrogen production and other process heat applications.

3. Design Philosophy

The BWRX-300 leverages the U.S. NRC approved ESBWR design, proven in-use materials, off-the-shelf components, and operational parameters within the range of the existing BWR experience base. It uses a well-defined, robust, procedure driven process to direct cost-effective design decisions. The BWRX-300 design is near term deployable with the lead unit planning to enter commercial operation in 2028. It utilizes a coolant conservation strategy to mitigate loss of coolant accidents which results in significantly simplified structures and systems with improved safety.

4. Main Design Features

(a) Nuclear Steam Supply System

The BWRX-300 Nuclear Steam Supply System (NSSS) leverages the proven supply chain of the ABWR and the natural phenomena driven safety features from the ESBWR. Components in this system include the Reactor Pressure Vessel (RPV) and fine motion control rod drives (FMCRDs). The NSSS delivers steam from the RPV to the turbine via the main steam system; and the condensate and feedwater system delivers feedwater to the RPV. It also provides overpressure protection of the reactor coolant pressure boundary (RCPB).

(b) Reactor Core

The reactor core of the BWRX-300 is arranged as an upright cylinder containing fuel assemblies located within the core shroud. During operation, the coolant/moderator enters the bottom of the core as subcooled water and exists the core as saturated steam. The BWRX-300 utilizes GNF2 fuel which is the same fuel as used by the majority of the BWR fleet today. GNF2 fuel design is a 10x10 array of 78 full-length fuel rods, 14 part-length rods and two large central water rods.

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

The BWRX-300's RPV assembly consists of a pressure vessel with removable head, internal components and appurtenances. The reactor internal include the core (control rods and nuclear instrumentation), core support and alignment structures (shroud, shroud support, top guide, core plate, control rod guide tube and orificed fuel support), chimney, chimney head and steam separator assembly and steam dryer assembly.

(e) Reactor Coolant System

The reactor coolant system (RCS) is natural circulation driven and provides cooling of the reactor core in all operational states and all postulated off normal conditions. The BWRX-300 leverages natural circulation modelling and operational information from the ESBWR and the Dodewaard Nuclear Power Plant that operated in the Netherlands.

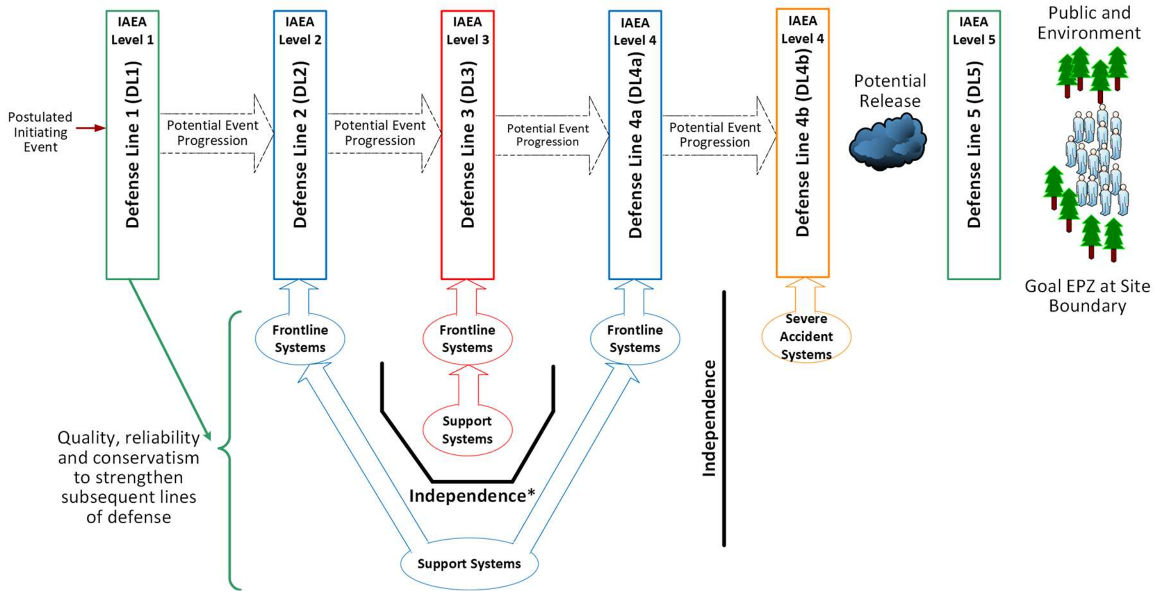
(f) Steam Generator

The BWRX-300 is a direct cycle plant with steam generated directly from the RPV. The RPV has internal steam separators and driers to provide high quality steam to the steam turbine. Steam generators are not required.

(g) Primary pumps

The BWRX-300 utilizes natural circulation and requires no primary pumps.

5. Safety Features



Defense in Depth

(a) Engineered Safety System Approach and Configuration

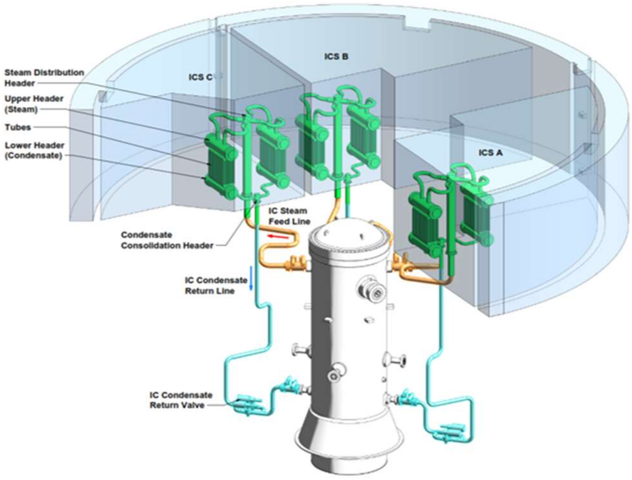
The BWRX-300 is designed using IAEA guidelines to streamline licensing in many countries. The basic BWRX-300 safety design philosophy is built on utilization of inherent margins (e.g., larger structure volumes and water inventory) to eliminate system challenges and can easily accommodate transients. Natural driven phenomena safety-related systems mitigate accidents without the need for AC power. The BWRX-300 defence in depth concept uses Fundamental Safety Functions (FSF) to define the interface between the defence lines and the physical barriers.

(b) Decay Heat Removal System

The normal, non-Safety Class 1, shutdown cooling (SDC) system for decay heat removal comprises two independent pump and heat exchanger trains. These trains provide redundant decay heat cooling capacity. The major components of each train are a pump and a HX, along with valves, piping, instrumentation and controls, and power inputs.

(c) Emergency Core Cooling System

The ECCS for the BWRX-300 is the Isolation Condenser System (ICS). The ICS removes decay heat after any reactor isolation and shutdown from power operations under normal or abnormal conditions. The ICS consists of three independent loops each containing a heat exchanger (HX) and isolation condenser (IC) with capacity of approximately 33 MW. The ICS tubes condense steam from the RPV on the inside surface and transfer heat on the outside surface to the IC pool which is at atmospheric pressure. This steam condensation and gravity allow BWRX-300 to cool itself for a minimum of seven days without power or operator action.



Isolation Condenser

(d) Spent Fuel Cooling Safety Approach / System

The Fuel Pool Cooling and Cleanup System (FPC) provides cooling, filtration, and demineralization of the water in the spent fuel pool (SFP), reactor well, and equipment pool. The FPC consists of two parallel trains of equipment. Each train includes a pump and heat exchanger. Water is passed from the pump through a filter element and mixed resin deep bed demineralizer, then passed through a heat exchanger to remove thermal energy before being returned to the pools.

(e) Containment System

The BWRX-300 Primary Containment Vessel (PCV) encloses the RPV, provides radiation shielding and acts as a boundary for radioactive released from the RPV. The PCV is a vertical cylinder approximately 16 meters diameter and 44 meters high. It is integral to and surrounded by the Reactor Building (RB). The PCV is dry and is located mostly below grade.

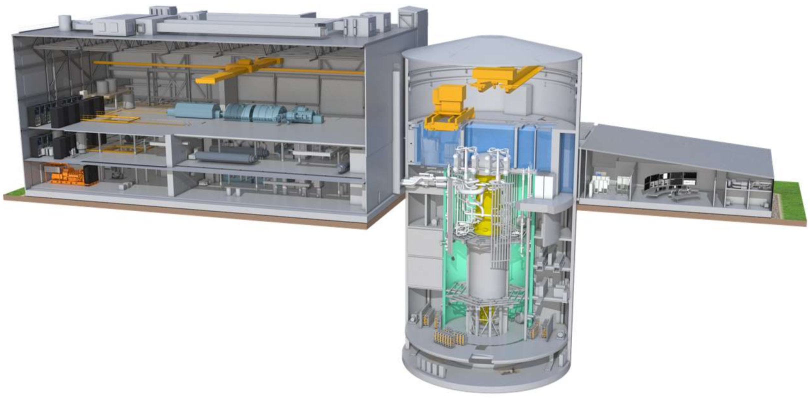
6. Plant Safety and Operational Performances

The defense in depth approach utilized in the BWRX-300 results in an internal event core damage frequency less than 10^{-7} per year and a large release frequency less than 10^{-7} per year. The lessons learned and best practices of the BWR fleet have been applied to the BWRX-300 with the expected availability factor greater than 95%. The BWRX-300 is capable to operate under the load following range of 50 to 100% with a ramp rate of ±0.5% per minute.

7. Instrumentation and Control System

The BWRX-300 I&C system (also referred to as the Distributive Control and Information System or "DCIS") is a completely integrated control and monitoring system for the power plant. The overall I&C system is divided into systems-Safety Class 1, Safety Class 2/3 and non-Safety.

8. Plant Layout Arrangement


Plant Layout Arrangement

The reference site for BWRX-300 is a 260 m by 332 m footprint. The power block is comprised of the Reactor Building (RB), Turbine Building (TB), Control Building (CB) and Radwaste Building (RwB). The footprint of the power block is approximately 140 m x 70 m.

The RB is the only Seismic Category 1 structure in the BWRX- 300. The PCV and RPV are situated at or below grade within the RB. A water pool sites above the PCV. The three ICS pools sit next to the pool above the PCV with one IC located in each pool. The CB houses the control room, electrical, control and instrumentation equipment. The TB encloses the turbine, generator, main condenser, condensate and feedwater systems, condensate purification system, and off gas system.

9. Testing Conducted for Design Verification and Validation

For most structures, systems and components (SSCs), the BWRX-300 uses commercial off-the-shelf equipment and proven construction techniques. Design verification is performed to ensure the design complies with customer requirements, technical requirements, regulatory requirements, and codes and standards. Engineering tests that provide product performance or design basis information, or impact the design, licensing, or operation of a product, product line, or environmental qualification, are also used. The test program also includes prototype qualification tests, production tests, proof tests before installation, and preoperational tests to demonstrate satisfactory performance of the system/components.

10. Design and Licensing Status

The BWRX-300 has undergone preapplication reviews with the United Kingdom Office of Nuclear Regulation (ONR), the United States Nuclear Regulatory Commission (U.S. NRC) and the Canadian Nuclear Safety Commission (CNSC). In the United Kingdom, the BWRX-300 has undergone an evaluation by UK Department for Business, Energy and Industrial Strategy funded Mature Technology. In the USA, five Licensing Topical Reports (LTRs) for design features and analysis methods considered to have higher regulatory risks have been submitted and approved. In Canada, the BWRX-300 is undergoing a combined Phase 1 and 2 Vendor Design Review process that is expected to be completed in the Fall of 2022. In late 2022, Ontario Power Generation (OPG) is planning to submit a License to Construct Application for a BWRX-300 at Darlington New Nuclear Project site, DNNP-1.

11. Fuel Cycle Approach

The BWRX-300 has the same open fuel cycle as operating BWRs. Refueling outages are 10-20 days based on the number of fuel assemblies to be replaced considering plant cycle duty. The cycle durations of 12 to 24 months can be accommodated to meet customer needs. Approximately 32 bundles are replaced following a 12-month cycle and 72 bundles following a 24-month cycle.

12. Waste Management and Disposal Plan

The BWRX-300 utilizes the lessons learned and best practices from the decades of operational experience of the BWR fleet in waste management and disposal. Waste that is generated will be segregated for optimal treatment, storage and final disposal. Gaseous wastes will be removed from the steam system by steam jet air ejectors and passed through charcoal beds for absorption.

13. Development Milestones

2017	Conceptual design started
2018	BEIS funded Mature Technology Evaluation the UK ONR
2019	Pre-application activities initiated with the US NRC
2020	Phase 1 and 2 Vendor Design Review initiated with CNSC in Canada
2021	BWRX-300 Downselected by OPG after extensive evaluation of all SMRs
2022	Planned OPG submittal of License to Construct for DNNP-1
2028	Planned Commercial operation of lead BWRX-300

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

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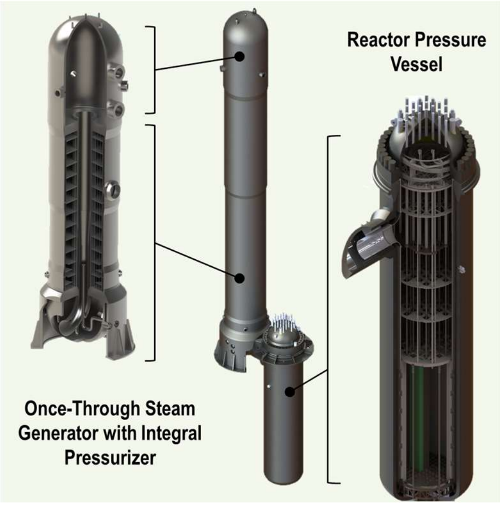
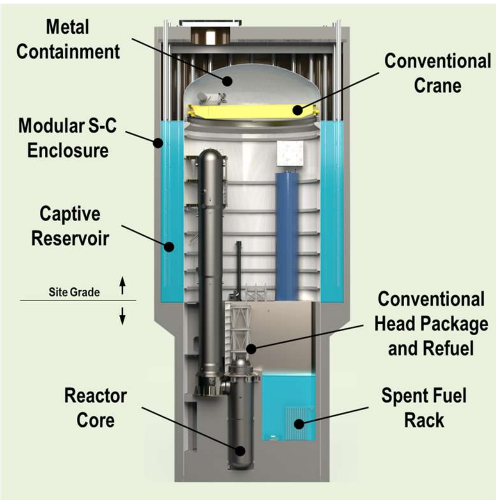


Reactor Coolant System



Containment Systems

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
Thermal/electrical capacity, MW(t)/MW(e)	525 MW(t) 174 gross / 160 net MW(e)
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	15.5 / 3.4
Core Inlet/Outlet Coolant Temperature (ºC)	243 / 321
Fuel type/assembly array	UO_2 pellet / square array
Number of fuel assemblies in the core	57
Fuel enrichment (%)	4.0 (average)
Core Discharge Burnup (GWd/ton)	45 (initial design)
Refuelling Cycle (months)	24 (equilibrium)
Reactivity control mechanism	Control rods and soluble boron
Approach to safety systems	Fully Passive; utilize varied phenomena and redundancy.
Design life (years)	80
Plant footprint (m^2)	28,000
RPV height/diameter (m)	15 / 3
RPV weight (metric ton)	295 (with fuel and internals)
Seismic Design (SSE)	0.5g, derived from the NRC Regulatory Guide 1.60
Fuel cycle requirements / Approach	Approximately 1/3 batch fraction removed each refuelling outage
Distinguishing features	Passive safety cooling systems and active non-safety systems; critical components below grade. Integrated dry spent fuel storage and transportation system
Design status	PSAR for commercial project by 2023, with detailed design to complete by 2025

1. Introduction

The SMR-160 has been developed by Holtec International as an advanced PWR small modular reactor producing 525 MW thermal power or 160 MW net electric power. The plant design incorporates robust passive safety systems to achieve a highly reliable design that protects owner's investment from all postulated accidents, sabotage, or inadvertent human actions. In accord with Holtec's *Design Philosophy*, the SMR-160 is designed as a 'walk-away safe' power plant – no operator actions are necessary to cope with design basis accidents and safely reject decay heat. The plant is greatly simplified relative to conventional plants to improve its fabricability, constructability, and maintainability, substantially facilitated by incorporating entirely passive safety systems with a natural circulation primary loop NSSS. A modular construction plan for the SMR-160 involves fabrication and assembly of the largest shippable components prior to arrival at a site. A 24-month construction period is envisaged for each nth-of-a-kind unit.

2. Target Application

The primary application of SMR-160 is electricity production with optional cogeneration equipment (i.e., hydrogen generation, thermal energy storage, district heating, seawater desalination). The design is readily configurable for siting in water-scarce locations using Holtec International's proprietary air-cooled condenser technology. The SMR-160 is capable of both "black-start" and isolated operation, rendering the plant ideal for destinations with unstable power grids or off-grid applications.

3. Design Philosophy

Defence-in-Depth is achieved by including passive safety cooling systems and active non-safety systems within the NSSS, with all critical components installed below grade and protected by a robust containment enclosure structure. The plant is designed to be walk away safe, with a small spent fuel pool within containment and a small source term, resulting in an orders of magnitude improvement in safety, e.g., CDF, as compared to current generation plants, with no dependence on operator actions.

4. Main Design Features

(a) Design Philosophy

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

(b) Nuclear Steam Supply System

The SMR-160 is a PWR with a naturally circulating reactor coolant system (RCS) primary loop. The RCS is comprised of the reactor pressure vessel (RPV) and a steam generator (SG) in an offset configuration with an integrated pressurizer welded to the top of the steam generator. The RPV and the SG are connected by a single connection which contains both the hot leg and the cold leg in concentric ducts. The offset configuration allows easy access to the core without moving the RPV or SG during refuelling. The entire primary circuit pressure boundary is qualified under the ASME Code as a combined vessel, without any intermediary loop piping.

(c) Reactor Core

The SMR-160 employs an efficient reactor core design that uses a traditional reload shuffle. The reactor core contains standard length 17x17 PWR fuel assemblies commercially available, along with typical magnetic-jack driven control rod assemblies. The reactor vessel internals support the reactor core, the control rod assemblies, and the control rod drive shafts within the RPV. The core is designed for a nominal two-year cycle with flexibility for shorter or longer cycles depending upon utility requirements.

(d) Reactivity Control

Long term reactivity control is provided by burnable absorbers integral to the fuel which are designed to optimize 3D power distributions, cold shutdown margin, and hot excess reactivity. Short term changes in reactivity are controlled by adjusting soluble boron and movements of control rod assemblies (CRAs). CRAs are positioned by control rod drive mechanisms (CRDM) based on existing electro-mechanical technology. The CRDMs are located outside the reactor coolant system on the RPV upper head.

(e) Reactor Pressure Vessel and Internals

The RPV is an ASME Section III, Class 1, thick-walled cylindrical pressure vessel with an integrally welded bottom head and a removable top head. The offset configuration of the SG and RPV enables the use of traditional external control rod drive mechanism and greatly simplifies refuelling operations relative to typical integral PWR designs. The reactor internal structures are designed to be supported from the bottom of the vessel and are completely replaceable.

(f) Reactor Coolant System

The SMR-160 RCS operates purely by natural circulation induced by the density difference in the primary water and the height of the RPV and steam generator. There are no reactor coolant pumps in the system. The RCS consists of three major components, a Reactor Pressure Vessel (RPV), a Steam Generator (SG), and an integral Pressurizer, qualified as a single combine vessel.

(g) Steam Generator

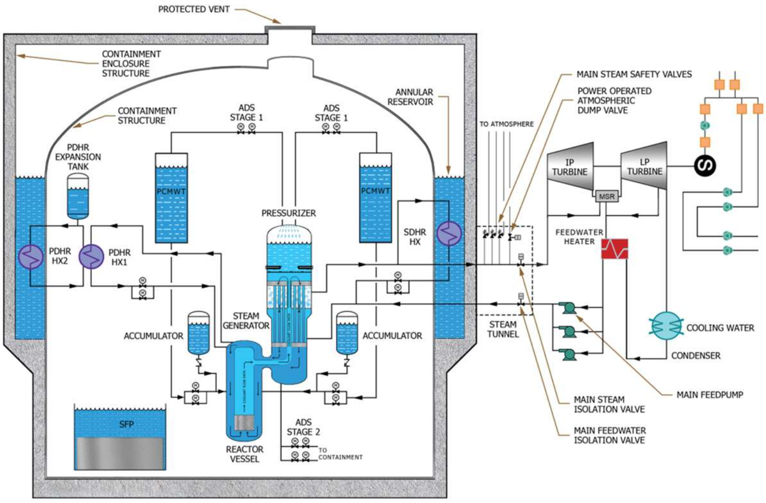
The SMR-160 includes a single, vertically oriented, once through straight tube SG with the reactor coolant flowing inside thermally treated Inconel 690 tubes. The use of straight tubes ensures easy access for in-service inspection. The SG uses sub-cooled feedwater to produce superheated steam on the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

(h) Pressurizer

The pressurizer is integral to the steam generator and uses heaters and cold-water spray nozzles to perform the functions of a typical pressurizer. Integrating the pressurizer with the steam generator eliminates significant primary piping. The large relative size of the Pressurizer eliminates any need for Power Operated Relief Valves (PORVs) and simplifies power manoeuvring.

5. Safety Features

The SMR-160 safety basis incorporates defence-in-depth via multiple and varied pathways for rejection of decay heat. All safety systems are located inside the robust containment enclosure structure, rendering them secure and safe from external threats. All makeup water needed for a postulated loss of coolant accident (LOCA) is inside containment. Another large inventory of water within a captive reservoir between the containment enclosure structure and the containment structure provides long-term post-accident coping and allows the decay heat removal function to transition to air cooling for an unlimited coping period after a design basis accident without any operator action or make-up.



PCCS and Power Conversion Diagram

(a) Engineered Safety System Approach and Configuration

SMR-160 relies on passive and redundant safety systems that also operate by natural circulation. The passive safety systems ensure safe shutdown can be maintained and decay heat removal occurs for an unlimited period without the need for power, make-up water, or operator actions.

(b) Passive Core Cooling System (PCCS) for Decay Heat Removal and Emergency Core Cooling

The PCCS is designed to provide emergency core cooling and makeup to the RCS during postulated accidents. The system uses passive means such as natural circulation for core cooling and compressed gas expansion and gravity injection for core makeup without the use of active components such as pumps. The PCCS is comprised of four major subsystems: (i) Primary decay heat removal system (PDHR); (ii) Secondary decay heat removal system (SDHR); (iii) Automatic depressurization system (ADS) and (iv) Passive core make-up water system (PCMWS). The PDHR directly cools the primary coolant by re-routing the reactor coolant through a heat exchanger and rejecting the heat to a second loop full of water.

(c) Spent Fuel Cooling Safety Approach

The spent fuel pool of the SMR-160 is located within the containment. The SMR-160 provides emergency cooling to spent fuel during postulated events by integrating the cooling and make-up to the spent fuel pool to a common reactor protection strategy. For long term cooling and make-up, the spent fuel pool acts as a combined containment sump for the recirculation of coolant within the containment vessel to the reactor..

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

The SMR-160 containment system consists of a steel containment structure (CS), enclosed within a reinforced concrete containment enclosure structure (CES). The CES provides shielding and protection from external events. In addition to preventing the release of radioactive fission products to the environment, the containment system acts as a large passive heat exchanger. The PCHR passively cools the containment volume, without any required actuations. The large heat transfer area and high conductance of the metal containment wall results in near-instantaneous heat rejection to the AR. The AR then rejects heat to the environment.

6. Plant Safety and Operational Performances

The SMR-160 natural convection-driven reactor coolant loop is coupled with an optimized simple steam cycle and is well adapted to load following. The SMR-160 is designed to achieve a high capacity factor of > 95% with an estimated refuelling outage of 10 days every two-year fuel cycle. The reliance on passive safety features and elimination of operator actions for response to postulated events yields a safer plant. The core damage frequency is calculated to be on the order of 10^{-8}/year and importantly on the order of 10^{-7}/year when considering only plant safety systems.

7. Instrumentation and Control Systems

SMR-160 utilizes the Mitsubishi Electric Total Advanced Controller (MELTAC) platform for the plant I&C/HSI design. MELTAC provides unique nuclear specific I/O and configuration flexibility to perform all nuclear safety and non-safety functions using the same digital platform..

8. Plant Layout Arrangement

(a) Containment Enclosure Structure

The SMR-160 reactor is housed within a containment structure (CS) protected by a containment enclosure structure (CES) comprised of steel-concrete modules. The CES is missile-hardened and protects the CS and safety systems from the most severe environmental hazards or sabotage. Nearly half of the CS and CES is embedded underground. These structures house all safety systems and the spent fuel pool.

SMR-160 Plant Layout

(b) Reactor Auxiliary Building

The reactor auxiliary building houses many of the plant auxiliary systems. This building is designed to process spent fuel for dry interim on-site storage within Holtec International's proprietary HI-STORM UMAX modules (an underground dry cask storage technology), without any modification to the standard plant design. The integrated HI-STORM UMAX canister system can also be used for off-site transport to a centralized repository without repackaging.

(c) Balance of Plant

The steam turbine and associated systems are housed within the turbine building structure at grade level. The SMR-160 features a side exhaust steam turbine, optionally configured for air cooled condensation. The electric power system consists of the main generator, main transformer, auxiliary transformers, non-safety diesel generators and Class 1E batteries. The electrical system is designed to permit isolated operation in "island-mode" as well as start-up operations independent of the grid or "black-start."

9. Testing Conducted for Design Verification and Validation

A thermal hydraulic test facility has been engineered to yield data and information for benchmarking the SMR-160 following the process defined by the U.S. Nuclear Regulatory Commission. The facility provide data for system performance characteristics and important thermal hydraulic phenomena to support verification and validation. Facility implementation is underway in 2022, with experiments at an Integral and Separate Effects Test (ISET) facility at Idaho National Lab (INL) in the U.S planned to begin in late 2023.

10. Design and Licensing Status

Pre-application activities for the SMR-160 have commenced with multiple international regulators in parallel with development of commercial project opportunities. Formal pre-application activities, submittals and reviews are conducted with the USNRC in accord with a cooperatively agreed Regulatory Engagement Plan.

11. Fuel Cycle Approach

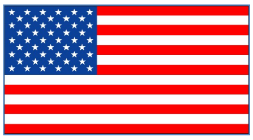
The SMR-160 fuel cycle is designed to discharge approximately one third of the fuel assemblies in the core each refuelling cycle, along with shuffling of a portion of the remaining fuel assemblies. The spent fuel is stored briefly in the spent fuel pool, which is uniquely protected within the same containment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

12. Waste Management and Disposal Plan

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

13. Development Milestones

2012	Conceptual design of SMR-160 commencement
2015	Conceptual design completed for SMR-160
2020	Preliminary design completed for SMR-160
2023	Ready for commercialization using a construction permit based process



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

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse Electric Company LLC, USA
Reactor type	Integral PWR
Coolant/moderator	Light water
Thermal/electrical capacity, MW(t)/MW(e)	800 / >225
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	15.5
Core Inlet/Outlet Coolant Temperature (ºC)	294 / 324
Fuel type/assembly array	UO_2 pellet/17x17 square
Number of fuel assemblies in the core	89
Fuel enrichment (%)	< 5
Core Discharge Burnup (GWd/ton)	> 62
Refuelling Cycle (months)	24
Reactivity control mechanism	CRDM, boron
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	65 000
RPV height/diameter (m)	28 / 3.7
Seismic Design (SSE)	Based on CEUS sites
Distinguishing features	Incorporates passive safety systems and proven components of the AP1000 plant and earlier Westinghouse designs
Design status	Conceptual design completed

1. Introduction

The Westinghouse small modular reactor (SMR) is an integral pressurized water reactor (PWR) design that builds upon the concepts of simplicity and advanced passive safety demonstrated in the AP1000® plant. The power station delivers a thermal output of 800 MW(t) and a net electrical output of greater than 225 MW(e) as a standalone unit, completely self-contained within a compact plant site. The entire plant is designed for modular construction with all components shippable by rail, truck, or barge.

2. Target Application

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

3. Main Design Features

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

The Westinghouse SMR reactor core is based on the licensed Westinghouse robust fuel assembly (RFA) design and uses 89 standard 17×17 fuel assemblies with a 2.4 m active fuel height and Optimized ZIRLO® cladding for corrosion resistance. A metallic radial reflector is used to achieve better neutron economy in the core while reducing enrichment requirements to less than the existing statutory limit of 5.0 weight percent ^{235}U. Approximately 40% of the core is replaced every 2 years with the objective to achieve efficient and economical operating cycle of 700 effective full power days, which coincides with existing regulatory surveillance intervals.

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

5. Safety Features

The Westinghouse SMR is an advanced passive plant where the safety systems are designed to mitigate accidents by natural driving forces such as gravity flow and natural circulation flow. The plant is not reliant on alternating current (ac) power or other support systems to perform its safety functions. The 7-day minimum coping time following loss of offsite power is a fundamental advancement over the 3-day coping time applied in the operating plants. The integral reactor design eliminates large loop piping and potential large break LOCA and reduces the potential flow area of postulated small-break LOCAs. The below grade locations of the reactor vessel, containment vessel, and spent fuel pool provide protection against external threats and natural phenomena hazards. The small size and low power density of the reactor limits the potential consequences of an accident relative to a large plant. The plant is designed to be standalone, with no shared systems, thus eliminating susceptibility to failures that cascade from one unit to another in a multi-unit station. The result is a plant capable of withstanding natural phenomena hazards and beyond-design-basis accident scenarios, including long-term station blackout.

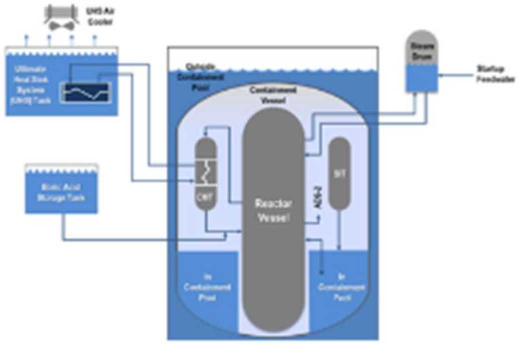
(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal System

Three diverse decay heat removal methods are provided in the Westinghouse SMR. The first method of decay heat removal uses gravity feed from the steam drum through the steam generator for approximately 80 minutes of natural circulation cooling. In this scenario, steam is released to the atmosphere through two redundant power-operated relief valves. The second decay heat removal method can be achieved by cooling the RCS with a passive decay heat removal heat exchanger, one of which is located in each of four core makeup tanks (CMTs). Heat from the CMTs is then rejected to four heat exchangers located in two ultimate heat sink (UHS) system tanks. The UHS tanks are sized to provide a minimum of 7 days of decay heat removal, with additional options to replenish lost inventory and cool the plant indefinitely. A third diverse method of decay heat removal capability is available by cooling the RCS with diverse bleed-and-feed methods, including a two-stage automatic depressurization system that vents the RCS to the containment through direct vessel injection (DVI) pathways, water injection from the four CMTs and in-containment pool (ICP) tank paths, and gravity-fed boric acid tank water makeup to the DVI paths. The steam vented from the RCS to the containment is cooled and condensed by the containment shell. The containment shell is cooled by the water in the outside containment pool (OCP) that completely surrounds the containment. When the OCP water eventually boils, makeup water is provided by gravity from each of the two redundant UHS tanks that maintain the OCP full of water. The water condensed on the containment shell flows back into the RCS through two sump injection flow paths.

Below-Grade Location of Westinghouse SMR Reactor and Containment Vessels.



Three Diverse Decay Heat Removal Methods of the Westinghouse SMR.

(c) Containment System

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

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control Systems

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

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

8. Plant Layout Arrangement

(a) Reactor Building

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

(b) Balance of Plant

The balance of plant design (BOP) consists of a conventional power train using a steam cycle and a water-cooled (or optional air-cooled) condenser. The BOP operation is not credited for design basis accidents. The power conversion system is comprised of the turbine-generator, main steam system and condenser. Each reactor drives a separate turbine generator and no sharing of reactor safety systems. The design includes “Island Mode” capability to handle grid disconnection and is capable of 100% steam bypass capability to handle turbine trip, both preventing a need for a reactor scram.

Plant Layout of Westinghouse SMR

(c) Turbine Generator Building

The electrical generator is designed for air cooling, which eliminates the potential for explosions that can occur with hydrogen-cooled options. The Westinghouse SMR condenser design includes the capability to use air cooling. The turbine is designed to accommodate a wide variety of backpressures with different blade configurations optimized for narrow-range, high-performance power. The water intake requirements will be comparable to existing plants on a per-power basis, but significantly less on a plant basis because of the lower power rating. This low water usage enables the reactor to be sited in places previously not available for nuclear construction.

(d) Electric Power Systems

The Westinghouse SMR onsite power system consists of a main ac power system and a direct current (DC) power system. The main ac power system is a non-Class 1E system and does not perform any safety-related functions. The plant dc power system is composed of the independent Class 1E and non-Class 1E DC power systems. Safety-related dc power is provided to support reactor trip and engineered safeguards actuation. Batteries are sized to provide the necessary dc power and uninterruptible ac power for items such as protection and safety monitoring system actuation; control room functions, including habitability; DC-powered valves in the passive safety-related systems; and containment isolation. Two diverse, non-safety AC power backup systems are provided: 1) diesel-driven generators to provide power for defence-in-depth electrical loads, and 2) a decay heat-driven generator. The decay heat-driven generator provides ac power to the plant using the heat generated by the core following reactor trip.

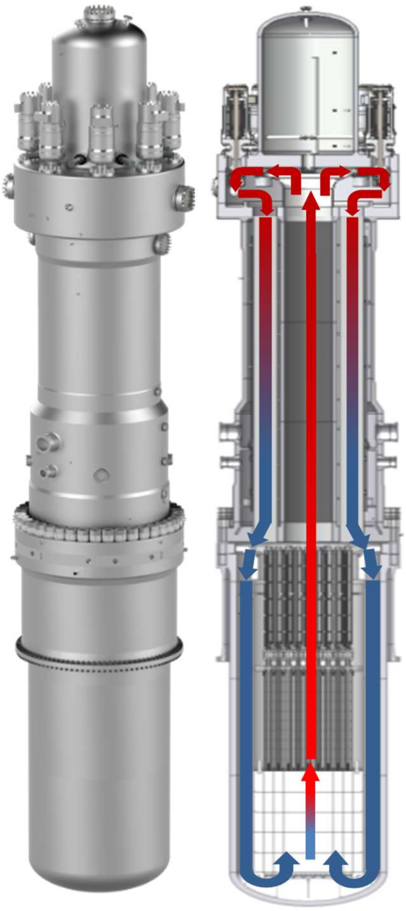
9. Design and Licensing Status

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

10. Development Milestones

2015	Conceptual design completed

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



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

1. Introduction

The mPower™ plant consists of an integral PWR small modular reactor and related balance of plant, designed by Generation mPower LLC to generate a nominal output of 195 MW(e) per module. In its standard plant design, each mPower plant comprises a ‘twin-pack’ set, or two mPower reactor modules, generating a nominal 390 MW(e). The design adopts internal steam supply system components, once-through steam generators, pressurizer, in-vessel control rod drive mechanisms (CRDMs), and vertically mounted canned motor pumps for its primary cooling circuit and passive safety systems. The plant is composed of reactor modules that are fully shop-manufactured, rail-shippable to a site and installed into the facility.

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

(f) Steam Generator

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

(g) Pressurizer

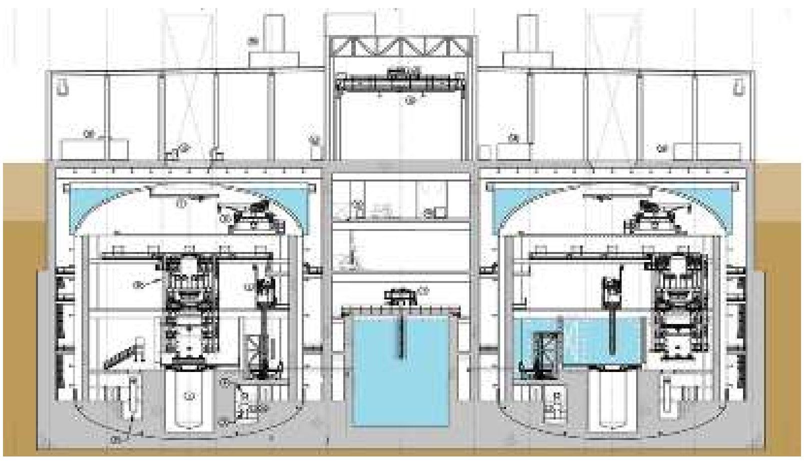
The integrated electrically heated pressurizer located at the top of the RPV maintains a nominal 14.8 MPa. Reactor coolant pressure is controlled by the heaters and steam space spray.

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

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



(b) Decay Heat Removal System

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

(c) Emergency Core Cooling System

The mPower ECCS is a safety system that provides three basic functions: (1) depressurization of the RCS, (2) reactor coolant inventory control during the event, and (3) core decay heat removal. With a system of automatic depressurization valves (ADVs) and the large coolant reserve provided by accumulators referred to as intermediate pressure injection tanks (IPIT) and the in-containment refuelling water storage tank (RWST), the reactor core remains covered following a design-basis event. The IPITs are maintained with pressurized nitrogen over the water. Should the automatic depressurization valves open, the reactor pressure vessel will depressurize until in equilibrium with the containment atmosphere. During and following that pressure equalization, check valves between the reactor pressure vessel and IPITs (early) and the RWST (late) open, injecting gravity-driven cooling water from the RWST. The RWST provides a minimum of 7 days to as much as 14 days of cooling without the need for external intervention or AC power to maintain reactor core cooling and safe shutdown.

To address the single-failure general design criteria, the ECCS is designed for n+1 components, with all components located inside containment. There are four high-pressure and four low pressure depressurization paths arranged in pairs connected to four pressurizer connections. The ADVs are the isolation valves for the eight lines. There are two injection paths to the RCS. Each path to the RCS is connected to an IPIT and to the RWST.

(d) Containment System

The steel containment vessel and interfacing safety systems work in concert to protect the core, provide long-term core cooling, and prevent the release of radioactive materials to the environment without reliance on AC power or operator action well beyond the regulatory expectation of 72 hours following an accident. This seismic Category I structure is designed to withstand the maximum internal pressure from design basis accidents, including LOCA and steam line break. Normal access to the steel containment vessel is via personnel hatches and a removable equipment hatch provides access for large component replacement. Internal containment atmosphere pressure and temperature is maintained by passive containment cooling (PCC) via an integral water tank situated in direct contact with the containment dome, providing passive cooling under accident conditions. Heat is removed from the hot steam and air inside containment via heat transfer through the containment dome structure to the water in the PCC tank located on the outside surface of the dome. Containment atmosphere response to a breach in the RCS may be characterized by two distinct phases. The first phase (blowdown) is an injection of hot steam corresponding to RCS depressurization. The high rate of steam injection during this period increases containment atmospheric temperature and pressure and quickly disperses a steam and air mixture throughout the containment volume. With cold walls and other structure surfaces, condensation is the most important heat transfer mechanism. Natural convection and conduction-limited heat transfer to the PCC tank distinguishes the second phase. There is sufficient water available to passively remove a minimum of seven days of core decay heat by evaporation.

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement

(a) Reactor Building

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

(b) Balance of Plant

The balance of plant design (BOP) consists of a conventional power train using a steam cycle and a water-cooled (or optional air-cooled) condenser. The BOP operation is not credited for design basis accidents. The steam and power conversion system is comprised of the turbine-generator, main steam system and condenser. Each reactor drives a separate turbine generator and no sharing of reactor safety systems. The design includes "Island Mode" capability to handle grid disconnection and is capable of 100% steam bypass capability to handle turbine trip, both preventing a need for a reactor scram.

i. Turbine Generator Building:

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

ii. Electric Power Systems:

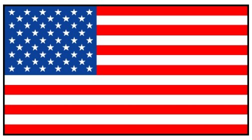
The main generator supplies power to plant auxiliaries during normal plant operation through an isolated phase bus duct and the unit auxiliary transformer. Offsite power to plant auxiliaries during startup, shutdown, and outage conditions is supplied via back-feed from the main transformer and the unit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

9. Design and Licensing Status

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

10. Development Milestones

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



OPEN20 (Last Energy Inc., United States of America)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Last Energy Inc., United States of America
Reactor type	1-loop PWR
Coolant/moderator	Light water / Light water
Thermal/electrical capacity, MW(t)/MW(e)	73 / 22
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.5 / 2.8
Core inlet/outlet coolant temperature (^oC)	286 / 312
Fuel type/assembly array	UO_2 pellet / 17x17 array
Number of fuel assemblies in the core	Up to 24
Fuel enrichment (%)	<4.95
Refuelling cycle (months)	72
Core discharge burnup (GWd/ton)	>30
Reactivity control mechanism	Control rods
Approach to safety systems	Passive, Defense-in-depth, Diverse, and Redundant
Design life (years)	40+ years
Plant footprint (m^2)	2023
RPV height/diameter (m)	9.1 / 2.3
RPV weight (metric ton)	89.3
Seismic design (SSE)	0.5g horizontal and 0.4g vertical peak ground accelerations
Fuel cycle requirements/approach	Open cycle; wet and dry storage on-site. Provisions for closed cycle are under consideration.
Distinguishing features	Fully-modular construction, underground nuclear island, digital controls, closed tertiary loop using air cooling
Design status	Detailed design

1. Introduction

Last Energy has developed a production ready, fully-modular micro-PWR with nominal electric output of 20MWe for distributed baseload applications including industrial siting and combined heat and power. Last Energy's technology is a scaled-down one-loop pressurized water reactor (PWR) that specifically avoids innovations around fuel, reactor physics, or material science. Despite the small power output, the fuel is still standard PWR fuel, <5% enriched, found in over 300 currently operating PWRs, and the entire power plant follows a design requirement of incorporating only "off-the-shelf" equipment. The primary technical innovations are specific to productization, modularization, and constructability of the power plant. The values listed in this book are generally for OPEN20, an academic version of Last Energy’s PWR-20 commercial plant design.

2. Target Application

The plant is designed for both electricity production as well as non-electrical process heat applications. Each plant comes standard with combined heat, power and cooling capabilities, and can deliver up to 300°C process steam for broad industry applications.

3. Design Philosophy

The plant design has been developed to target four key outcomes:
(1) Minimize technology risks by utilizing existing and proven technology (e.g., PWR & steam)
(2) Straightforward demonstration of risk reduction using conservative assumptions and fewer critical systems
(3) Modular design and factory construction accelerates delivery time by reducing required on-site activities
(4) Minimize siting constraints and environmental impact (e.g., air-cooled, small footprint)

4. Main Design Features

(a) Nuclear Steam Supply System

The nuclear steam supply system (NSSS) is based on the PWR technology found in over 300 currently operating reactors. The NSSS includes the reactor system, reactor coolant pumps, a pressurizer, a steam generator, misc. equipment, and piping required to generate steam for the turbine. The system uses forced convection and operates in a single closed primary loop.

(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 up to 24 fuel assemblies delivering a thermal power of approximately 70 MW(t). Each fuel assembly contains poisoned fuel pins using distributed Gd2O3 as well as 24 guide tube locations for control rod fingers and a central instrument tube. The operational characteristics are designed to ensure large margin to thermal-mechanical fuel limits including DNBR. The core is designed for a nominal six-year cycle with flexibility for shorter cycles depending upon utility requirements.

(c) Reactivity Control

Core reactivity is controlled by means of control rods, solid burnable poison, and soluble boron dispersed in the primary coolant. Burnable poison rods flatten the radial and axial power profile, which results in an increased thermal margin of the core. The number and concentration of the burnable absorber rods in each fuel type are selected so that reactivity of each assembly can be as flat as possible. The control rods use boron carbide as the absorber material.

(d) Reactor Pressure Vessel and Internals

The base of the reactor pressure vessel (RPV) is forged carbon steel. It is a thick-walled cylindrical pressure vessel with an integrally welded bottom head and a removable top head. The RPV has a relatively small width to height ratio. The internals are the same as a conventional PWR scaled to fit a reduced fuel bundle count. Notably this is not an integral reactor system.

(e) Reactor Coolant System

The Reactor Coolant System (RCS) is of the same basic design of light water PWRs in use throughout the world today and includes the equipment and piping required to transfer energy produced from nuclear fuel assemblies to convert high pressure water into steam. The system utilizes a single closed loop forced circulation to transfer high pressure (approximately 15.5 MPa), high temperature (approximately 330°C) water between the reactor vessel and the steam generator equipment where it passes through tubes in shell and tube heat exchangers to transfer energy and generate steam (approximately 315°C) from lower pressure feedwater.

(f) Steam Generator

A vertically oriented shell and tube steam generator uses heat created by the reactor to produce steam for the turbine. This style of heat exchanger isolates the reactor coolant loop (flow through the tubes) from the turbine steam (flow through the shell).

(g) Pressurizer

The reactor coolant pressure is established and kept nearly constant by a pressurizer. Electric heaters create and control the size of the steam space inside the pressurizer vessel. This imposes backpressure on the reactor coolant to prevent it from boiling at other locations. The pressurizer is sized to greatly dampen the effects of transient pressure events.

(h) Primary pumps

After exiting the steam generator, coolant is pumped to the reactor by vertical, canned, single stage centrifugal pumps that operate with variable frequency drives. To improve system reliability there are three reactor coolant pumps; however, only one pump is required to deliver full coolant flow to the reactor for normal operations.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Last Energy's design has been developed to incorporate the best practices learned over five decades of nuclear plant engineering. Defense-in-depth is provided through multiple layers of fault prevention (e.g., IAEA 5 level standards) as well as independent and diverse active and passive systems. Passive safety systems are designed to perform without human input or electrical power.

(b) Decay Heat Removal System

A passive decay heat removal system is included to protect the system integrity during unlikely accident scenarios.

(c) Emergency Core Cooling System

Last Energy's design has been developed to incorporate the best practices learned over five decades of nuclear plant engineering. Defense-in-depth is provided through multiple layers of fault prevention (e.g., IAEA 5 level standards) as well as independent and diverse active and passive systems. Passive safety systems are designed to perform without human input or electrical power.

(d) Spent Fuel Cooling Safety Approach / System

The plant design includes provisions for interim on-site wet and dry spent fuel storage in preparation for final offsite disposal or reprocessing.

(e) Containment System

The nuclear island is contained inside a modular, leak tight, steel enclosure that prevents any inadvertent release of radionuclides during either normal operating or accident conditions. Additionally, the subterranean containment provides protection from external events.

6. Plant Safety and Operational Performances

The behavior of the plant during normal and faulted conditions has been analyzed and assessed using industry validated codes to demonstrate significant safety margins across the levels of defense-in-depth. The Probabilistic Safety Assessment calculates a LERF < 1E-7. Additionally, our capacity factor is greater than 95% and refueling outage time is less than 3 months every 72 months.

7. Instrumentation and Control System

A distributed control system collects measurements and issues commands via input/output devices in the process areas. Digitized data is communicated over a secure Ethernet-based network. The on-site control room has only interfaces for critical process information and alarms in keeping with the largely autonomous operational concept of the plant. Extensive plant data from multiple installations is transmitted via one-way communication links to a remote monitoring center where off-site staff can observe trends and instruct field personnel to investigate abnormalities or perform corrective actions. Remote control of plant equipment is not possible.

8. Plant Layout Arrangement

Two primary structures are located on-site. The nuclear island is located below grade and houses the NSSS. The balance of plant structure is located above grade and houses all remaining systems including the steam turbine generator, feedwater system, auxiliary systems, and electrical systems.

Artistic rendering of OPEN20 containment design

9. Testing Conducted for Design Verification and Validation

Last Energy's power plant does not include any experimental systems. Its NSSS is similar in configuration to several historical micro/modular, land-based, single-loop PWRs such as the BR-3 and MHA-1 reactors. Where it deviates from traditional nuclear systems (e.g., digital controls) it utilizes only equipment that can be found in thousands of comparable fossil-based thermal power plants.

10. Design and Licensing Status

The PWR-20 has completed detailed design and is producing construction ready drawings. Pre-licensing activities have commenced with multiple international nuclear regulators in parallel alongside the fabrication of non-nuclear systems. The project execution plan projects initial operation of the first deployed reactors by the mid-2020s. Site control has already been secured at multiple locations.

11. Fuel Cycle Approach

The reactor is operated on a 72-month fuel cycle. The plant design includes provisions for interim on-site wet and dry spent fuel storage in preparation for final off-site disposal or reprocessing.

12. Waste Management and Disposal Plan

All spent fuel is stored onsite until final decommissioning. Spent fuel assemblies and waste can be handled and transported in manner similar to that of conventional PWRs.

13. Development Milestones

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

PART I.2. WATER COOLED SMALL MODULAR REACTORS (MARINE BASED)

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

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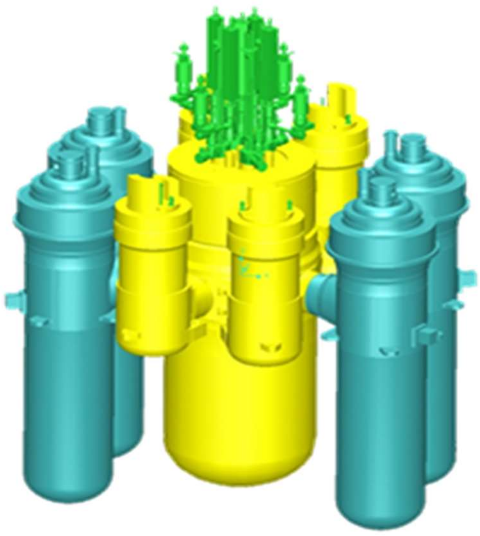


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", Rosatom, Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	150 / 35
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	12.7
Core Inlet/Outlet Coolant Temperature (ºC)	280 / 316
Fuel type/assembly array	UO_2 pellet in silumin matrix
Number of fuel assemblies in the core	121
Fuel enrichment (%)	18.6
Core Discharge Burnup (GWd/ton)	45.4
Refuelling Cycle (months)	30-36
Reactivity control mechanism	Control rod driving mechanism
Approach to safety systems	Active (partially passive)
Design life (years)	40
Plant footprint (m^2)	4320 (Floating NPP)
RPV height/diameter (m)	4.8 / 2.0
RPV weight (metric ton)	N/A
Seismic Design (SSE)	9 point on the MSK scale
Distinguishing features	Floating power unit for cogeneration of heat and electricity; no onsite refuelling; spent fuel take back
Design status	Connected to the grid in Pevek in December 2019. Entered full commercial operation in May 2020.

1. Introduction

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

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

The steam lines while exiting from the SGs are routed through containment to a set of steam inlet valves, and finally into the turbine building for electricity conversion. Cogeneration equipment could be modified into the medium-low temperature heat process concept if one or multiple separation heat exchangers are positioned between the primary and secondary loops.

(b) Reactor Core

Fuel utilization efficiency is achieved by using dispersion fuel elements. One of the advantages foreseen by the FNPP under construction is long term independent operation in remote regions with decentralized power supply. The design requires refuelling of reactor after every 2.5–3 years of operation. Refuelling is performed 14 days after reactor shutdown when the levels of residual heat releases from spent FAs have reached the required level. The spent nuclear fuel is initially stored on board at the FNPP and then returned to Russian federation. No special maintenance or refuelling ships are necessary. Single fuel loading is done in order to provide maximum operation period between refuelling. The fuel is loaded in the core all at once with all fuel assemblies being replaced at the same time.

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

5. Safety Features

The KLT-40S is designed with proven safety aspects such as a compact structure of the SG unit with short nozzles connecting the main equipment, primary circuit pipelines with smaller diameter, and with proven reactor emergency shutdown actuators based on different operation principles, emergency heat removal systems connected to the primary and secondary circuits. Additional barriers are provided to prevent the release of radioactivity from the FNPP caused by severe accidents. Among them are passive and active physically separated and independent safety systems, I&C systems, diagnostic systems, active cooling train through primary circuit purification system's heat exchanger thermally coupled with a 'third' independent circuit exchanging heat energy with ambient sea or lake water, active cooling train through the SGs heat exchangers with decay heat removal accomplished through the condenser which in turn is cooled down by ambient sea or lake water, 2 passive cooling trains through the SGs with decay heat removal via emergency water tank heat exchangers, and venting to atmosphere by evaporation from said tanks. Both active and passive safety systems are to perform the reactor emergency shutdown, emergency heat removal from the primary circuit, emergency core cooling and radioactive products confinement. The KLT-40S safety concept encompasses accident prevention and mitigation system, a physical barriers system, and a system of technical and organizational measures on protection of the barriers and retaining their effectiveness, in conjunction with measures on protection of the personnel, population and environment. The KLT-40S safety systems installed on FNPPs are distinctive from those applied to land-based installations in security of the water areas surrounding the FNPP, anti-flooding features, anti-collision protection, etc. Passive cooling channels with water tanks and in-built heat exchangers ensure reliable cooling to 24 hours.

(a) Engineered Safety System Approach and Configuration

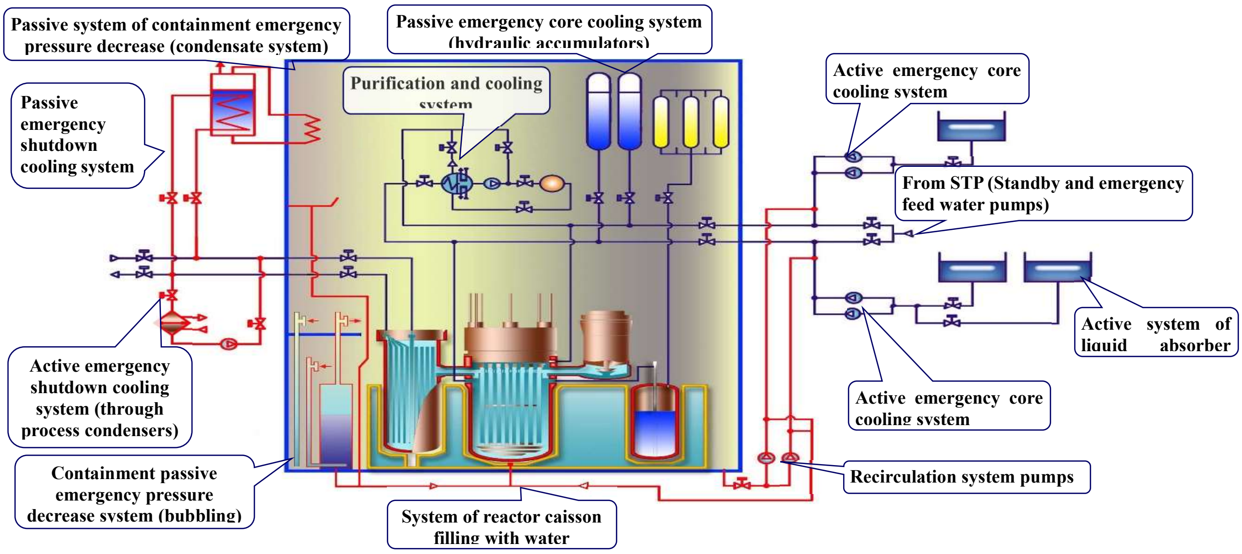
The active components of the protection system are scram actuators for six (6) groups of the control rods.

(b) Decay Heat Removal System

The decay heat removal system is intended to remove core residual heat upon actuation of reactor emergency protection in case of abnormal operation including accidents, as well as to remove residual heat at normal RP decommissioning. The decay heat removal system includes two secondary passive cooling channels via steam generators, one active secondary cooling channel via steam generators and one active cooling channel via the primary/third heat exchanger.

(c) Emergency Core Cooling System

The ECCS is intended to supply water to the reactor for core cooling in accidents associated with primary coolant loss, makeup of primary coolant during process operations, supply of liquid coolant to the reactor at failure of the electromechanical reactor shutdown system, adjustment of water chemistry and hydraulic testing of the primary circuit and associated systems, secondary and third loop sections disconnected at inter-circuit leaks and designed for primary pressure. The ECCS includes high-pressure ECCS subsystem with makeup, high-pressure ECCS subsystem with hydraulic accumulators and Low-pressure ECCS subsystem with recirculation pumps.



(d) Containment System

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

6. Plant Safety and Operational Performances

The KLT-40S NPP ensures electricity and heat generation within the power range of 10% to 100% for a continuous operation of 26 000 hours. The NPP is designed for manoeuvring speed of up to 0.1 %/s. As a countermeasure against the external impact, the NPP is fitted with both ground safety and floating physical protection means. Structures are designed to be placed in the Arctic zone at the depth of 2 m at freezing temperatures. The FPU and NPP buildings are designed to withstand the crash of an aircraft of 10 tons. Based on analysis, the radiation emission limits are satisfied for all conditions.

7. Electric Power Systems

The electric power system in the FPU is comprised of the following: main electric system; and emergency electric system. The main electric system of the FPU is intended to generate electricity and transmit it to the power system of the region, as well as to transmit electricity to internal consumers. The system includes two main three-phase AC generators of 35 MW each and eight back-up diesel generators of 992 kW each. The emergency electric system supplies electricity to safety system loads in all operation modes, including loss of operating and back-up electric power sources. The FPU has independent emergency electric systems for each reactor plant. Each emergency electric system has two channels with an emergency diesel generator of 200 kW.

8. Plant Layout Arrangement

The coastline line of the FCNPP has the complex engineering building with equipment to distribute and transmit electricity to loads and to prepare and transfer heating water to loads and auxiliary buildings, including: two hot water storage tanks; partially in ground tank with slime water; wet storage bunker; two cooling towers; access control point; site enclosure; lighting towers. The coastal line of the FCNPP does not provide for handling nuclear materials and radiation hazardous media.

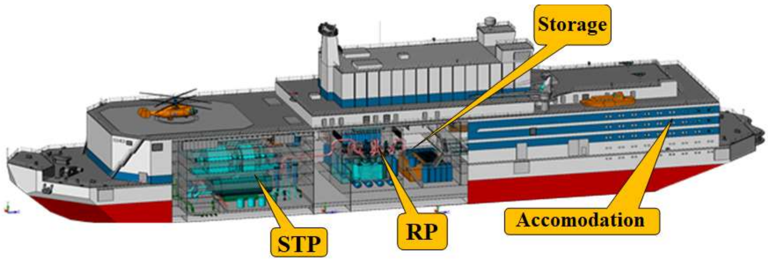
(a) Reactor Building

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

(b) Control Building

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

General cross-section view of the FPU



(c) Turbine Generator Building

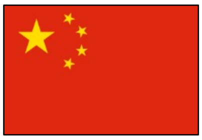
The steam turbine plant (STP) is intended to convert the thermal power from steam obtained in the KLT-40S reactor to the electric and thermal one to heat water in the intermediate circuit of the cogeneration heating system. The FNPP structure includes two steam turbine plants. Each STP is independent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

9. Design and Licensing Status

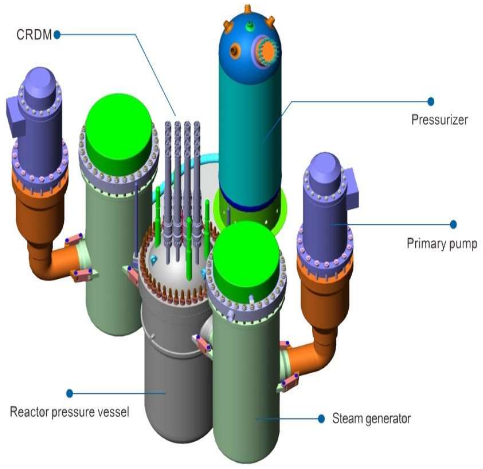
KLT-40S is the closest to commercialization of all available FNPP designs, and expects deployment through the Akademik Lomonosov FNPP. The KLT-40S is a modified version of the commercial KLT-40 propulsion plants employed by the Russian icebreaker fleet. The environmental impact assessment for KLT-40S reactor systems was approved by the Russian Federation Ministry of Natural Resources in 2002. In 2003, the first floating plant using the KLT-40S reactor system received the nuclear site and construction licenses from Rostechnadzor. The keel of the FNPP carrying the KLT-40S, the Akademik Lomonosov in the Chukotka region, was laid in 2007. The construction of Akademik Lomonosov was completed in 2017. The Akademik Lomonosov has started commercial operation in December 2019 in the town of Pevek in Chukotka region.

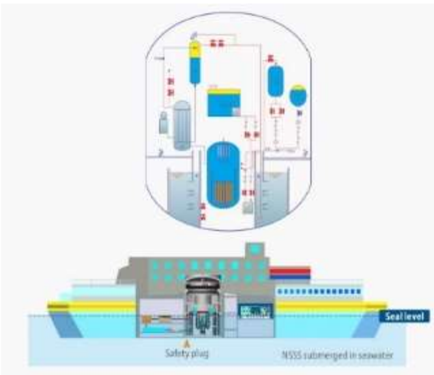
10. Development Milestones

1998	The first project to build a floating nuclear power plant was established
2002	The environmental impact assessment was approved by the Russian Federation Ministry of Natural Resources
2006	After several delays the project was revived by Minatom (Russian Federation Ministry of Nuclear Energy)
2012	Pevek was selected as the site for the installation of NPPs. JSC "Baltiysky Zavod" undertook charge of construction, installation, testing and commissioning the first FPU
2017	Completion of construction and testing of the floating power unit at the Baltic shipyard
2018	Dock-side trials, fuelling, final tests completion with reactor core, attainment of reactor's first criticality
Summer 2019	Transportation of FPU to the town of Pevek
December 2019	Connected to grid on 19th of December in Pevek
May 2020	Fully commissioned in Pevek on 22nd of May



ACPR50S (CGNPC, China)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	China General Nuclear Power Group (CGNPC), China
Reactor type	Loop type PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	200 / 50
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.5
Core inlet/outlet coolant temperature (°C)	299.3 / 321.8
Fuel type/assembly array	UO_2 pellet / 17 × 17 square
Number of fuel assemblies in the core	37
Fuel enrichment (%)	< 5
Refuelling cycle (months)	30
Core discharge burnup (GWd/ton)	< 52
Reactivity control mechanism	Control rod driving mechanism (CRDM), solid burnable poison
Approach to safety systems	Passive
Design life (years)	40
RPV height/diameter (m)	7.2 / 2.2
Fuel cycle requirements/approach	60 months
Distinguishing features	Floating power boat, once-through steam generator, passive safety
Design status	Detailed design

1. Introduction

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

2. Target Application

As an offshore floating SMR, the ACPR50S is designed for the following multipurpose applications: combined energy supply for offshore oil drilling platform; offshore combined energy supply; coastland and island combined energy supply; energy supply for offshore mining, nuclear power ship; and distributed clean energy for islands together with solar energy and wind power.

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

The compact loop-type PWR nuclear steam supply system (NSSS) design of the ACPR50S consists of the reactor pressure vessel (RPV) that houses the core, two once-through steam generators (OTSG), two main reactor coolant pumps (RCP) and a pressurizer (PZR), all of which are interconnected by short reactor coolant system legs. The primary cooling system is based on forced circulation during normal operation. The system has natural circulation capability and heat removal capacity up to 10% thermal power.

(b) Reactor Core

The low power density design with a low enriched UO_2 fuelled core ensures a thermal margin of greater than 15% which can accommodate any anticipated transient event. This feature ensures the core thermal reliability under normal and accident conditions. The 37 fuel assemblies (FAs) of ACPR50S core, with an axial length of 2.2m, have a square 17 × 17 configuration. The expected average fuel enrichment is less than 5%, similar to standard PWR fuel. The reactor will be able to operate 30 months per fuel cycle.

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

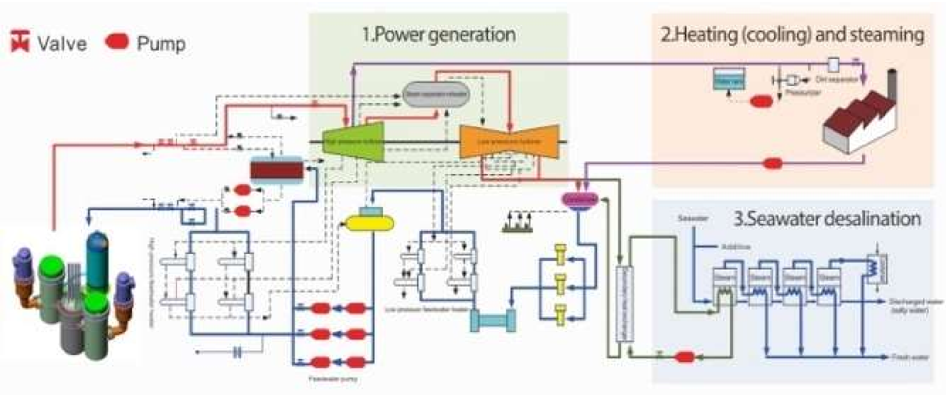
ACPR50S reactor pressure vessel is 2.2m in diameter and 7.2m high. It envelopes and fixes the core and the RPV internals, so that the fission reaction of the nuclear fuel is limited in one space.

(e) Reactor Coolant System

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

(f) Steam Generator

The ACPR50S has two OTSGs with helically coiled tubes to produce superheated steam under normal operating conditions. The OTSGs are located on both sides of the reactor vessel. The small inventory of the secondary side (tube side) water in each OTSG prohibits a return to power following a steam line break. In the case of accidents, the OTSG can be used as the heat exchanger for active and passive secondary residual heat removal system (ASHR & PSHR), which remove the decay heat from the primary system.



Energy Cascade Supply

(g) Pressurizer

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

(h) Primary pumps

The two ACPR50S Reactor Coolant Pumps (RCP) are intended to circulate the reactor coolant through the reactor core. They are mounted directly to Reactor Pressure Vessel by short nozzles. The reactor coolant pumps are designed with no seals, eliminating the potential risks of seal failure LOCA, which significantly enhances safety and reduces pump maintenance.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal System

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

(c) Emergency Core Cooling System

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

(d) Containment System

The containment system is to contain radioactive material and protect the environment against primary coolant leakage. The containment system of ACPR50S is called reactor cabin, a square containment. The containment has a volume of 870 m^3, a design internal pressure of 1.4 MPa and a design external pressure of 0.3 MPa. The containment isolation system (CIS) is to provide isolation for the containment and to prevent and restrict the escape of radioactive fission products in the event of an accident. The CHE is used to reduce the concentration of hydrogen in the containment to the safety limit under DBA and BDBA and to continuously monitor the hydrogen concentration at the top of the containment. The CPS is developed to cope with the DBA that can lead to a pressure rise inside containment and suppress the containment pressure peak to ensure the integrity of the containment. The CPS is composed of the suppression pool system and suppression pool cleaning and cooling system. The CHR is used to prevent the containment slow over temperature and overpressure. The CHR is a passive natural circulation system and can provided 7 days of cooling for the containment in the case of no external water apply.

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control System

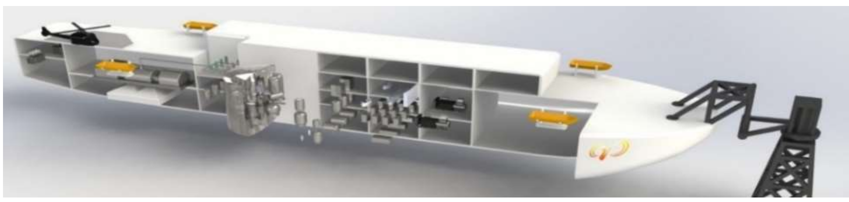
The I&C system design for ACPR50S is based on defence in depth concept, compliance with the single failure criterion and diversity. The I&C system design for ACPR50S is mainly used in the steady-state and transient power during operation and provides automatic protection against unsafe reactors and abnormal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

8. Plant Layout Arrangement

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

(a) Floating Platform

The reactor building and fuel storage area are equipped with a full monitoring system with closed circuit monitoring system to oversee and prevent unauthorized access to the fuel. Reactor building is a pre-stressed concrete shell structure composed of a right



General Plant Arrangement in the float platform cylinder with a hemispherical dome, with steel plate lining to act as a leak tight membrane. Reactor building is founded on a common base-mat together with the auxiliary building in which the main control room and fuel storage area are located. The figure above shows the overview of the reactor cabin, auxiliary cabin, turbine-generator cabin, the emergency diesel generator cabin and the main control room arrangement of ACPR50S.

(b) Onshore Basement

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

(c) Control Room

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

9. Testing Conducted for Design Verification and Validation

The CGN's experiment centre has safety test platforms, equipment and key technology test platform, SMR wave condition test platform, etc. Various experiments are performed for design validation.

10. Design and Licensing Status

The ACPR50S has completed the preliminary design work and is currently carrying out detailed design. An industrial demonstration plant of ACPR50S is planned to be constructed in China.

11. Fuel Cycle Approach

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

12. Waste Management and Disposal Plan

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

13. Development Milestones

2012	Starting Conceptual Design and formulating plan of theoretical tests
2014	Completion of Overall Design
2020	Completion of Preliminary Design
2022	Detailed design stage

ACP100S (CNNC/NPIC, China)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CNNC/NPIC, China
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	385 / 125
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15 / 4.6
Core inlet/outlet coolant temperature (°C)	286.5 / 319.5
Fuel type/assembly array	UO_2 / 17×17 square pitch arrangement arrangement
Number of fuel assemblies in the core	57
Fuel enrichment (%)	< 4.95
Refuelling cycle (months)	24
Core discharge burnup (GWd/ton)	< 52
Reactivity control mechanism	Control rod drive mechanism (CRDM), Gd_2O_3 solid burnable poison and soluble boron acid
Approach to safety systems	Passive + Active
Design life (years)	40
Plant footprint (m^2)	8 000 (Floating NPP)
RPV height/diameter (m)	10 / 3.35
RPV weight (metric ton)	300
Seismic design (SSE)	Not applicable
Fuel cycle requirements/approach	Shore refuelling, temporally stored in the spent fuel pool
Distinguishing features	Integrated reactor with once through steam generator, towed to sites along coast
Design status	Basic design

1. Introduction

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

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

The integrated nuclear steam supply system (NSSS) design consists of the reactor core, and sixteen (16) once-through steam generators (OTSG). The four (4) canned motor pumps are installed nozzle to nozzle to the RPV. All these technologies provide high inherent safety and avoid large LOCA in large-size nuclear power plant.

(b) Reactor Core

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

(c) Reactivity Control

The reactivity is controlled by means of control rods, solid burnable poison and soluble boron dissolved in the primary coolant. There are 20 control rods, with a magnetic force type control rod driving mechanism (CRDM).

(d) Reactor Pressure Vessel and Internals

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

The internals not only support and fasten the core but also form the flow path of coolant inside RPV.

(e) Reactor Coolant System

The ACP100S primary cooling mechanism under normal operating condition and shutdown condition is done by forced circulation. The RCS has been designed to ensure adequate cooling of reactor core under all operational states, during and following all postulated off normal conditions. The integral design of RCS significantly reduces the flow area of postulated small break LOCA.

(f) Steam Generator

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

(g) Pressurizer

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

(h) Primary pumps

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal System

The PDHRS prevents core meltdown in the case of design basis accident (DBA) and beyond DBA, such as station black out, complete loss of feedwater, small-break LOCA (i.e., to prevent the change of beyond DBA to severe phase). The PDHRS of ACP100S consists of two residual remove exchangers and associated valves, piping, and instrumentation. The residual heat remove exchanger is located in the passive residual heat remove water storage tank mounted on left and right side of shipboard, which provides the heat sink for the emergency cooler. The decay heat is removed from the core by natural circulation. The PDHRS provides core cooling for 72 hours without operator intervention.

(c) Emergency Core Cooling System

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

(d) Spent Fuel Cooling Safety Approach / System

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

(e) Containment System

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

6. *Plant Safety and Operational Performances*

Nuclear safety is always the first priority. The ultimate goal of nuclear safety is to establish and maintain an effective defence that can effectively protect people, community, and environment from radioactive disaster. To be specific, the design and operation of ACP100S ensures that radiation dose to the workers and to the members of the public do not exceed the dose limits and kept it as low as reasonably achievable. Accident prevention measures ensure that radioactive consequences are lower than limited dose in terms of all the considered accident sequences and even in the unlikely severe accidents, mitigation of accidents induced influences can be ensured by implementing emergency plan. The design of ACP100S incorporates operational experience of the state-of-the-art design. Proven technology and equipment are adopted as much as reasonably possible. It's estimated that ACP100S has the core damage frequency (CDF) of less than 1×10^{-5} per reactor year and the large release frequency (LRF) of 1×10^{-6} per reactor year. The load factor is no less than 0.9.

7. Instrumentation and Control System

The Instrumentation and Control (I&C) system designed for ACP100S is based on defence in depth concept, compliance with the single failure criterion and diversity. The diversity in the design of I&C system is achieved through: (1) different hardware and software platforms for 1E and N1E I&C, (2) reactor protection system (RPS) with functional diversity, and (3) diverse protection systems to cope with the common mode failure of the RPS. I&C systems of the NSSS include reactor nuclear instrumentation system, RPS, diverse actuation system, reactor control system, rod control and rod position monitoring system, reactor in–core instrumentation system, loose parts and vibration monitoring system and other process control systems.

8. Plant Layout Arrangement

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

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

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

11. Fuel Cycle Approach

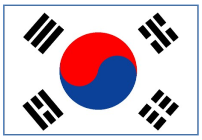
Spent fuel processing is conducted on the shore, using the service facility prepared at the base harbour, which is similar to the land-based reactor.

12. Waste Management and Disposal Plan

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

13. Development Milestones

2012	The concept design of ACP100S finished
2018	The basic design of ACP100S finished
2019	Preliminary feasibility study for one site finished
2021	Preliminary design of ACP100S finished
2023	PSAR planned to submit to National Nuclear Safety Authority
2024	Target to start construction



BANDI-60 (KEPCO E&C, Republic of Korea)

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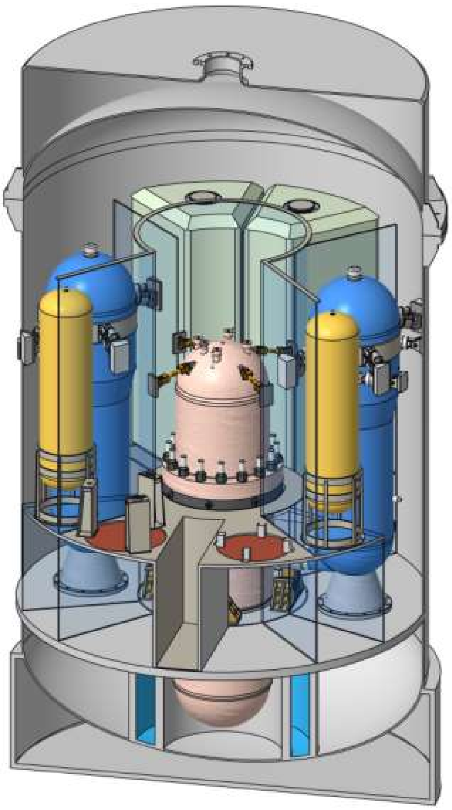


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	KEPCO E&C, Republic of Korea
Reactor type	PWR
Coolant/moderator	H_2O (Light Water)
Thermal/electrical capacity, MW(t)/MW(e)	200 / 60
Primary circulation	Forced
NSSS operating pressure (primary/secondary), MPa	15.5 / 6
Core inlet/outlet coolant temperature (°C)	290 / 325
Fuel type/assembly array	Oxide / 17×17
Number of fuel assemblies in the core	52
Fuel enrichment (%)	4.95
Refuelling cycle (months)	48~60
Core discharge burnup (GWd/ton)	29.4
Reactivity control mechanism	Rods
Approach to safety systems	Passive safety system
Design life (years)	60
Plant footprint (m^2)	6 500
RPV height/diameter (m)	11.2 / 2.8
RPV weight (metric ton)	110
Seismic design (SSE)	N/A
Fuel cycle requirements/approach	4~5 years / Single batch 4.95%
Distinguishing features	Passive Safety System
Design status	Conceptual design

1. Introduction

BANDI-60 is a compact two-loop PWR with two U-tube type steam generators. KEPCO E&C focuses on deploying a floating nuclear power plant in the sea. BANDI-60 is designed for niche markets, for instance, distributed power and heat supply to remote communities, desalination, and hybrid energy systems with renewables, through a collaboration with marine and shipbuilding industries.

Typical Plant Layout

2. Target Application

BANDI-60 is designed for off-gid and remote areas such as islands, Arctic area and isolated grid.

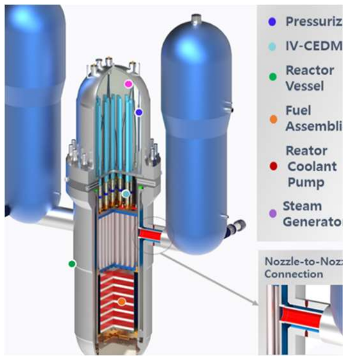
3. Design Philosophy

The BANDI-60 design adopts technology innovations developed in-house from 2013, such as the in-vessel control element drive mechanism (IV-CEDM), soluble boron-free (SBF) design and operation, and the Top-Mounted In-Core Instrumentations (TM-ICI). The CEDM inside of reactor pressure vessel will eliminate control element ejection accident. BANDI-60 is designed to operate at soluble boron-free conditions to simplify the design, operation and maintenance of nuclear reactor.

4. Main Design Features

(a) Nuclear Steam Supply System

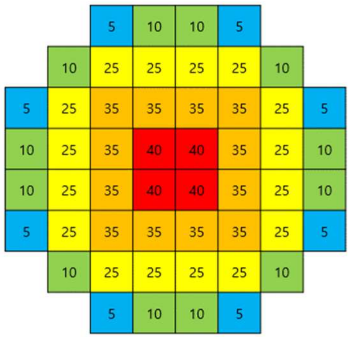
BANDI-60 is a block-type PWR plant in which the main components are directly connected, nozzle-to-nozzle, thus eliminating the large break loss of coolant accident (LBLOCA) from the design basis accident. This provides also improved operational surveillance and maintenance as compared to integral type designs. For steam generator, the U-tube recirculation type is adopted considering its performance has been proven with plenty of operational experience in commercial nuclear power plants.



Reactor Cross-section View

(b) Reactor Core

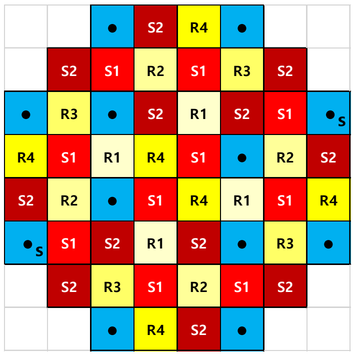
The BANDI-60 reactor uses a conventional UO_2 ceramic fuel enriched up to 4.95 % in a 17×17 square array fuel assembly. To produce 200MW thermal power over a 4 – 5 years, the reactor core is composed of 52 fuel assemblies. BANDI-60 does not use soluble boron. The Pyrex (B_2O_3) rods as burnable absorber are loaded in the core to control the excess reactivity over core life by replacing 24 fuel rods with different boron concentrations. The numbers shown in the figure represents boron concentration. The centre fuel assemblies have a higher boron concentration up to 40% while as low as 5% for the outer fuel assemblies to flatten radial power distribution. The burnup calculation showed that the core can operate for 57.5 months with full power.



Reactor Core with Burnable Poison

(c) Reactivity Control

The BANDI-60 reactor core uses 40 control rod assemblies with 24 B_4C control rods per assembly. In conjunction with the burnable poison, the control rod assemblies can provide enough negative reactivity to maintain shutdown with margin under worst condition for the soluble boron free BANDI-60 reactor core. There are two (2) control rod groups, 18 for regulating group and 22 for shutdown group. The control group is further grouped in 4 banks, R1 through R4, with 4 control rod assemblies per each bank except R4 which has 6 assemblies. The shutdown group consists of 2 banks, 10 assemblies for S1 and 12 assemblies for S2. The remaining 12 fuel assemblies without control rod are reserved for the TM-ICI.



Control Rod Arrangement

(d) Reactor Pressure Vessel and Internals

BANDI-60 Reactor Pressure Vessel (RPV) is similar to a conventional PWR vessel. The RPV is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical Reactor Vessel Closure Head (RVCH). The RPV contains internal structures, core support structures, fuel assemblies, the IV-CEDMs, and control and instrumentation components. The inlet nozzle of RPV is directly connected to the outlet nozzle of Steam Generator (SG). The Direct Vessel Injection (DVI) nozzles are attached to the RPV for a direct emergency coolant injection as a part of the safety injection system. The lifetime of the RPV is extended to 60 years by the use of low carbon steel. The reactor internals consist of the core support structures, which include the core support barrel, upper guide structure assembly, IV-CEDM support structure, lower support structure, and the internal structures.

(e) Reactor Coolant System

The primary coolant flows in through the RPV inlet nozzles from the reactor coolant pump, passes through the annulus between RPV and core support barrel, through RPV bottom plenum and core, and finally flows out through the outlet nozzles of RPV connected to the inlet nozzle of SG.

(f) Steam Generator

The steam generator, a U-tube recirculation type, is employed as the basic design since its performance has been proven with plenty of operational experience in commercial nuclear power plants. new technologies will be examined for their feasibility such as compact plate-shell steam generator.

(g) Pressurizer

The pressurizer is integrated into the upper head of the reactor pressure vessel where a relatively large water and steam volume is provided as compared to that of conventional nuclear power plants. The pressurizer pressure is controlled by heaters and sprays. The CEDM is installed inside the reactor vessel.

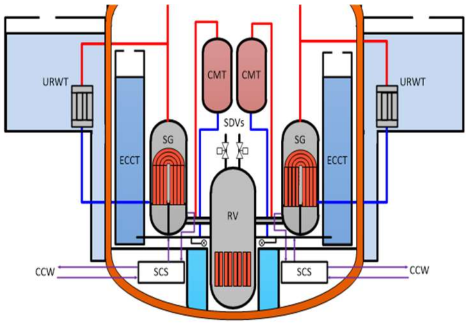
(h) Primary pumps

The Reactor Coolant Pump (RCP) is equipped with a leak-tight canned motor.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The BANDI-60 passive safety system consists of the PSIS, PRHRS, and PCCS. The PSIS performs gravity driven safety injection and system depressurization. The PRHRS performs decay heat removal by means of natural circulation and steam condensation. The PCCS performs condensation and convection through the containment vessel to the ultimate heat sink and refueling water tank (URWT) that has enough inventory of water for more than a month.



Safety Systems

(b) Decay Heat Removal System

The PRHRS provides decay heat removal when the normal cooling of RCS through SGs and condenser is not available after reactor trip. The steam from the SG condenses in the PRHRS heat exchanger, which is located inside the URWT, and the condensate returns by gravity to the SG via the feedwater nozzle. It cools down the RCS to the safe shutdown condition and maintains it for an extended time without refilling the URWT. The energy released to the containment atmosphere is continuously removed to the water stored in the URWT and CVH Cooling Jacket. This passive containment cooling will decrease the core damage frequency (CDF).

(c) Emergency Core Cooling System

The PSIS provides an RCS inventory makeup for any loss of coolant events other than LBLOCA by a gravity driven flow. The Core Makeup Tank (CMT) is pressurized to the RCS pressure by the Pressure Balance Line (PBL). Isolation valves between the CMT and DVI nozzle remain closed during normal operation, and are opened when the RCS depressurizes, the pressurizer level decreases, or the containment pressure increases beyond their set points. When the RCS pressure further decreases, the coolant stored in the Emergency Core Cooling Tank (ECCT) is injected by gravity. To facilitate ECCT injection into RCS, the RCS pressure has to be reduced and balanced with the containment pressure. For this rapid depressurization of RCS, the Safety Depressurization Valves (SDV) on the pressurizer are automatically opened before ECCT injection. Spilled water from the break is collected at the bottom of the containment and eventually the reactor vessel becomes submerged in the water and is continuously cooled by convection and natural circulation.

(d) Spent Fuel Cooling Safety Approach / System

The storage and handling facilities for new and spent fuel are located in the right side auxiliary building. Seismic category I tanks and pumps, equipment for the component cooling water system as well as equipment of plant cooling water system are arranged in the left side auxiliary building.

(e) Containment System

The containment vessel of BANDI-60 is a steel cylindrical type with 11 m inner diameter and 18.5 m in height. The reactor building, which surrounds the containment vessel, incorporates features to minimize and mitigate postulated severe accident phenomena. The steel containment vessel improves construction schedule compared to conventional pre-stressed concrete design. The vessel is designed to enable modularity to further reduce complexity and construction time. Heat released from the RCS to the containment is continuously removed to the water stored in the URWT through metal wall during a postulated accident.

6. Plant Safety and Operational Performances

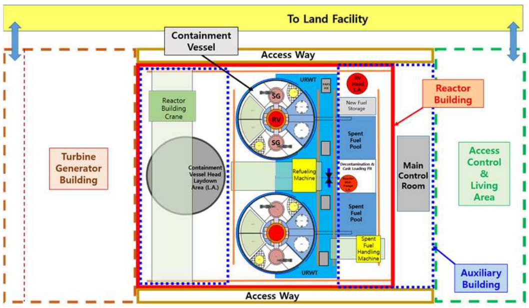
BANDI-60 is under conceptual design with CDF target of 10^{-8}/RY and 10^{-9}/RY of LERF.

7. Instrumentation and Control System

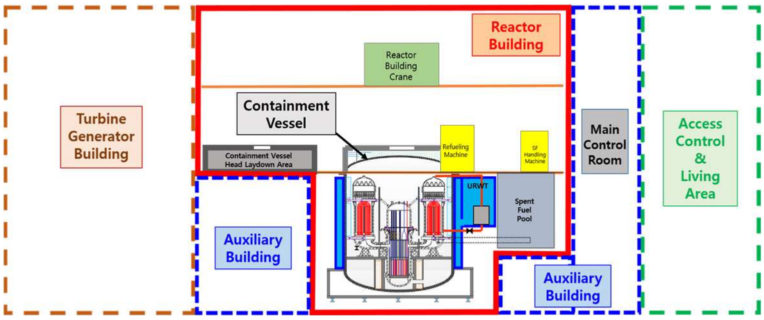
The instrumentation and control system (I&C) is composed of a fully digitalized system with data communication network and adopts a simple and compact structure suitable for BANDI-60 as floating SMR plant. This system can prevent unwanted reactor trip due to equipment failure or external disturbance by means of a Reactor Trip Prevention System (RTPS). The I&C design prevents the possibility of software common cause failure by implementing its own diversity design in the safety system. Therefore, a diverse actuation system is not required resulting in a more compact I&C system.

8. Plant Layout Arrangement

All safety systems are located in the reactor building positioned at the center of the floating site (ship). Auxiliary buildings are located on the left and right hand sides of the reactor building.



Sectional View of Plant Layout



Plan View of Plant Layout

a) Reactor Building

The reactor building contains two containment vessels with reactor coolant system, part of engineered safety systems, and related auxiliary systems. The reactor building is located in the middle of the ship, and the left and right sides of the reactor building are surrounded by auxiliary buildings and the north and south sides are equipped with passages for equipment transfer or operator movement.

b) Auxiliary Building

The auxiliary building houses facilities necessary to perform the main functions of the reactor coolant system and the engineered safety features related to the safety shutdown of the plant.

c) Turbine Building

Turbine building is adjacent to the left side auxiliary building and includes turbine, condenser, electrical system and plant chilled water system. It is designed to secure a space for carrying in and out of equipment, connection to external electric line and onshore facilities connected the secondary system.

d) Access Control

The access control area is in the access control and living area adjacent to the right side auxiliary building with a space for land facility connection. It also contains access controls to facilities and radiation areas.

9. Testing Conducted for Design Verification and Validation

To be developed.

10. Design and Licensing Status

To be launched. Design and Licensing status will be launched and site permit for FOAK plant will be developed. Design status of the BANDI-60 is under conceptual design.

11. Fuel Cycle Approach

The initial core design is based on a single batch operation having 4 – 5 years of cycle length to minimize refueling outage. As a floating power plant, the new and spent fuel are transported from and to an inland fuel fabrication and storage facility using a dedicated fuel transport ship. The spent fuel is shipped to the inland storage facility after about 10 years (2 cycle lengths) in the on-board spent fuel storage. To optimize the fuel cycle cost, a multi-batch core can be adopted with a shorter cycle length.

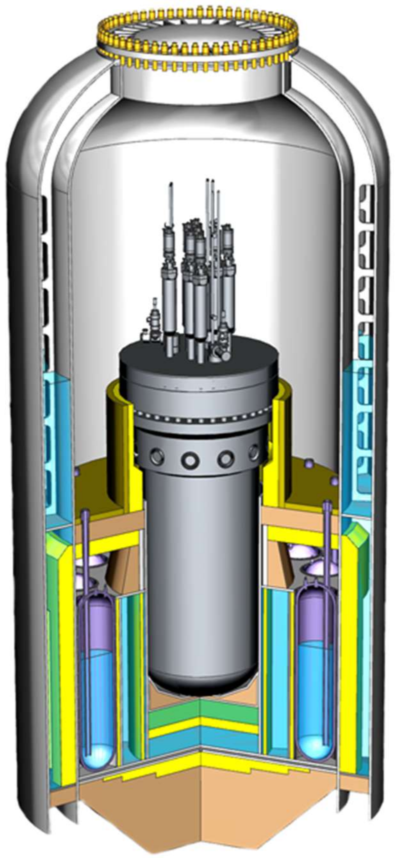
12. Waste Management and Disposal Plan

As a floating power plant, the spent fuels are transported from and to an inland fuel fabrication and storage facility using a dedicated fuel transport ship. The back-end fuel cycle option for spent fuel is dependent on the plant owner’s policy and requirements.

13. Development Milestones

2012	Preliminary and basic studies
2013	R&D projects for technology innovation started
2016	NSSS conceptual design started
2018	R&D projects for technology innovation completed
2019	NSSS conceptual design completed BOP and floating system conceptual design
2024	Complete BOP and floating system conceptual design
2025	Complete basic design and licensing review
2030	Projected construction

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



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, Rosatom, Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	38 / 6 – 9
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	16.2
Core Inlet/Outlet Coolant Temperature (°C)	250 / 325
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	121
Fuel enrichment (%)	< 20
Core Discharge Burnup (GWd/ton)	N/A
Refuelling Cycle (months)	120 – 144
Reactivity control mechanism	Control rod driving mechanism
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	20 000 (basic design)
RPV height/diameter (m)	6 / 2.4
Seismic Design	7 per Richter scale (basic design)
Distinguishing features	Natural circulation in the primary circuit
Design status	Final design

1. Introduction

The ABV-6E is reactor plant (RP) as a part of nuclear power system (NPS) that produces 14 MW(t) and 6 MW(e) in cogeneration mode or 9 MW(e) in condensation mode. ABV-6E integral PWR adopts natural circulation of the primary coolant. The ABV-6E design was developed using the operating experience of PWR reactors and recent achievements in the field of nuclear power plant (NPP) safety. The main objective of the project is to develop small, shipyard fabricated, multipurpose transportable NPP for safe operation over 10 to 12 years without refuelling at the berthing platform or on the coast. Plant maintenance and repair, refuelling and nuclear waste removal will be carried out at dedicated facilities.

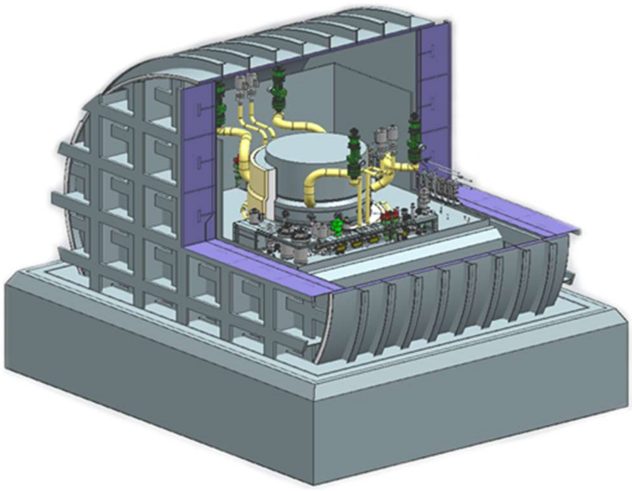
2. Target Application

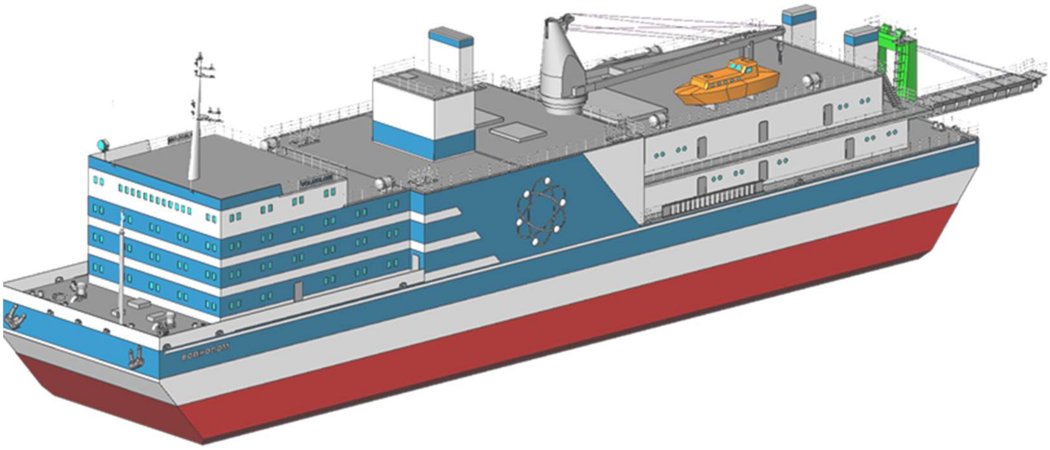
The ABV-6E RP is intended as a multi-purpose RP. The RP is designed with the capability of powering a floating power unit (FPU) as a part of floating nuclear power plant (FNPP) with a maximum length of 91.6 m, a beam of 26 m, a draft of 3.6 m and a displacement of 8100 t. Depending on the needs of the region, the FNPP can generate electric power or provide heat and power cogeneration or can be used for other applications. Besides, a land-based configuration of the plant is also applicable.

3. Design Philosophy

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

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





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

4. Main Design Features

(a) Reactor Core

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

(b) Reactivity Control

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

(c) Reactor Pressure Vessel and Internals

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

(d) Reactor Coolant System

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

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

The safety systems include:

- Passive heat removal system;
- Passive core cooling system;
- Reactor caisson water flooding system;
- Backup liquid absorber injection system

(b) Decay Heat Removal System

In emergency modes, a combined-type residual heat removal system (RHRS) is used to remove decay heat. This system functions on natural physical processes and - because there is an air heat exchanger cooled by the atmospheric air - ensures that the decay heat is being removed from the reactor for an unlimited time in all types of accidents. Because of this, and considering the measures taken to enhance the reliability of the passive RHRS, there are no active RHRS channels in the ABV 6E reactor design, which allows the output of emergency power supply sources to be reduced. The passive RHRS is made of two independent channels connected to two SGs each. Either channel, independently of the operability of the other channel, is capable of performing the RHRS functions, i.e. of maintaining the parameters of the primary circuit in the design limits for an unlimited time.

(c) Emergency Core Cooling System

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

(d) Containment System

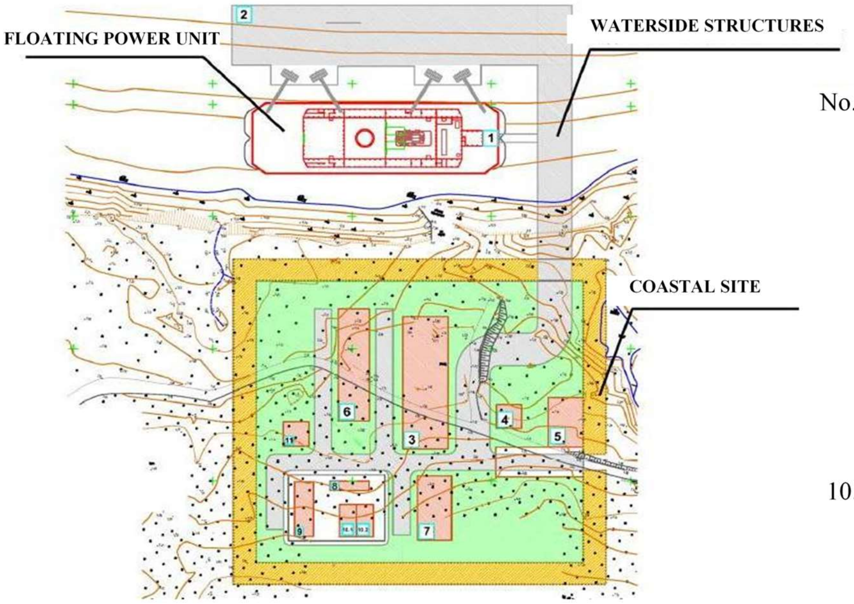
The metal-and-water shielding (MWS) tank is a substantial structure for the equipment of the RP. RPV, two pressurizers and the cooler of the purification and heat removal system are enclosed inside the dry caissons of the MWS tank. The passive reactor caisson water flooding system is designed to protect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

6. Plant Safety and Operational Performances

The NPP with ABV-6E generates electricity and heat in the power range of 20–100%N_{nom} with the continuous operation time of 26 000 hours. The NPP is designed for the manoeuvring rate of up to 0.1%/s. As a protection against the external events, the NPP is equipped with both ground and waterside security structures. The structures are designed for the sites in the Arctic zone with the frost penetration as deep as 2 m. The FPU and NPP design is intended to withstand the 10-ton aircraft crash. As the analysis of emergencies has shown, the radiation and ecological impact to the personnel, public and the environment during normal operation, abnormal operation, including the design-basis accidents, does not lead either to the excess of the radiation doses established for the personnel and public, or release of any of radioactive content in the environment.

This impact is also limited in beyond-design-basis accidents.

7. Plant Layout Arrangement



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

8. Design and Licensing Status

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

9. Development Milestones

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

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

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Afrikantov OKBM JSC, Rosatom, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	175 / 50
Primary coolant system circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	15.7 / 3.83
Core inlet/outlet coolant temperature (°C)	282 / 318
Fuel type/assembly array	UO_2 (cermet fuel) pellets / hexagonal
Number of fuel assemblies in the core	241
Fuel enrichment (%)	<20
Refueling cycle (months)	Up to 120
Core discharge burnup (GW·d/ton)	–
Reactivity control mechanism	Control rod drive mechanism; within the control and protection system (CPS)
Approach to safety system	Combined active and passive system
Design service life (years)	60
Ship footprint (m^2)	3360
RPV height/diameter (m)	8.6 / 3.45
RPV weight (metric ton)	265
Seismic design (SSE)	0.3g
Requirements for or approach to the operating cycle	Without on-site refueling
Distinguishing features	Integral reactor, in-vessel corium retention, double containment
Design status	6 prototype reactors were manufactured and installed on icebreakers (2 reactors are in the process of testing)

1. Introduction

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

2. Target Application

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

3. Design Philosophy

RITM series reactors are the evolutionary development of the reactors (OK-150, OK-900, KLT-40 series) for Russian nuclear icebreakers with a total operating experience of more than 60 years (more than 400 reactor-years). Incorporation the steam generators into the reactor pressure vessel, the reactor system and containment is very compact compared to the KLT-40. RITM design makes it possible to increase electric output (40% more) and reduces the dimensions (45% less) and the mass (35% less) in compare with KLT-40S.

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Cooling System

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

(e) Steam Generator

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

(f) Pressurizer

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

5. Safety Features

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

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

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

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

(b) Residual Heat Removal System

The residual heat removal system (RHRS) consists of four safety trains:
- Active safety loop with forced circulation through steam generator.
- Active safety loop with forced circulation through primary-third circuit heat exchanger of primary circuit coolant purification loop.
- Two passive safety loops with natural coolant circulation from water tanks through steam generators.

Evaporated is steam generators water condenses in air cooled heat exchangers and flow back to tanks with water heat exchangers. After complete water evaporation from the tanks, the air cooled exchangers continue provide cooling for unlimited time.

(c) Emergency Core Cooling System

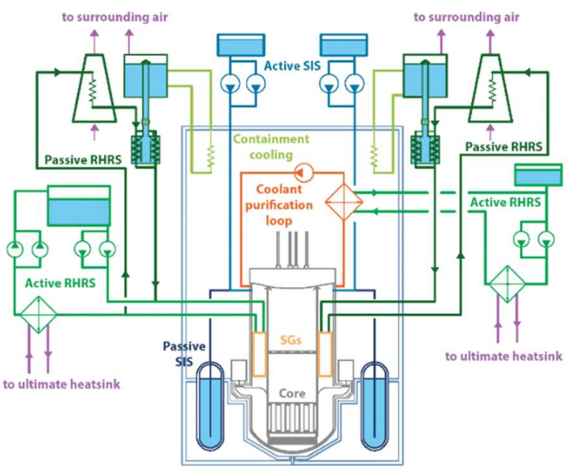
The ECCS consists of safety injection system (SIS) for water injection in primary circuit to mitigate the consequences of a break loss-of-coolant accident. The system is based on active and passive principles with redundancy of active elements in each channel and consists of (i) two passive pressurized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

(d) Containment System

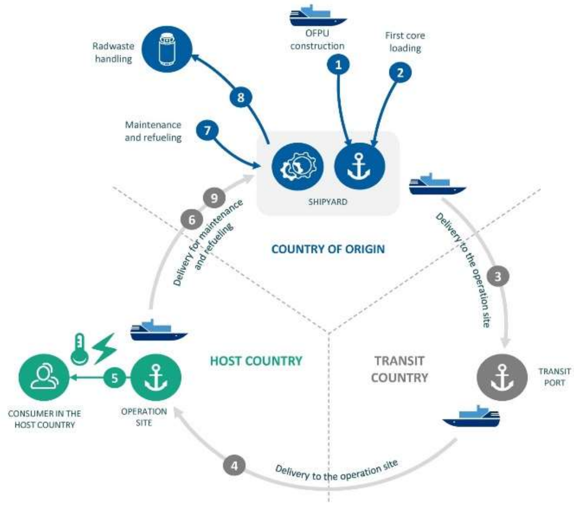
RITM is placed within the hermetically sealed envelope with dimensions of 6.6 m×6.4 m×16.2 m around the reactor vessel to localize possible radioactive releases. In case of severe accident thick wall of the reactor vessel keeps molten corium within the reactor. Water filled caisson under the reactor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

6. Plant Safety and Operational Characteristics

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



RITM-200M reactor safety system



OFPU life cycle

7. Instrumentation and Control System

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

8. General Layout of the Plant

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

OFPU fitted with the RITM-200M reactor plant

9. Testing to Check and Validate the Design

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

10. Design and Licensing Status

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

11. Approach to the Operating Cycle

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

12. Waste Management System and Waste Disposal Plan

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

13. Development Milestones

2016	The first RITM-200 installed on board the icebreaker *Arktika*
2020	The icebreaker *Arktika* under testing
2020	A conceptual design of the OFPU with RITM-200M

VBER-300 (JSC “Afrikantov OKBM”, Russian Federation)

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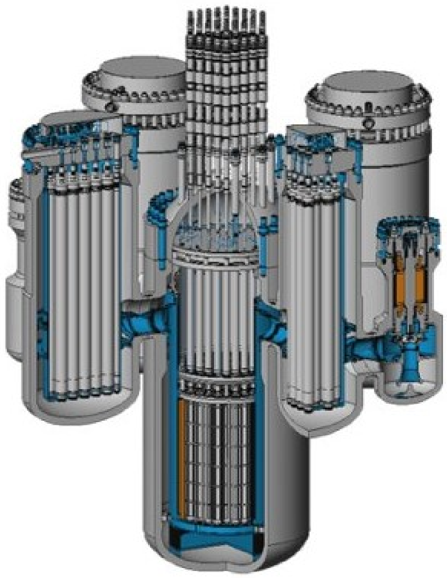


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, Rosatom, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	917 / 325
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	16.3
Core Inlet/Outlet Coolant Temperature (ºC)	292 / 328
Fuel type/assembly array	UO_2 pellet/hexagonal
Number of fuel assemblies in the core	85
Fuel enrichment (%)	4.95
Core Discharge Burnup (GWd/ton)	50
Refuelling Cycle (months)	72
Reactivity control mechanism	Control rod driving mechanism and soluble boron
Approach to safety systems	Hybrid (active and passive) system
Design life (years)	60
RPV height/diameter (m)	9.3 / 3.9
Seismic Design (SSE)	0.25g
Distinguishing features	Power source for transportable Floating NPPs, cogeneration options, compact design
Design status	Licensing stage

1. Introduction

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

2. Target Application

The VBER-300 nuclear plants are intended to supply thermal and electric power to remote areas where centralized power is unavailable, and to substitute capacities of available cogeneration plants on fossil fuels. The design is also proposed to be used as a power source for seawater desalination complexes. The VBER-300 nuclear plant has two reactor units that operate in the steam-condensing mode and can generate 600 MW(e) to satisfy power demands of a city with a population of 300 000. According to the OKBM’s data, when VBER-300 has cogeneration capabilities, the total electric output will reduce to 200 MW(e) providing 460 Gcal/hr for process heat applications.

3. Design Philosophy

VBER-300 design using ship-based modular configuration enhances the safety philosophy through proven marine modular technologies. The reactor design has no pipelines in the primary circulation circuit. The reactor unit incorporates the reactor and four steam generators – MCPs two-vessel units. The compact reactor system comprises the steam generating system in a limited space of the reactor compartment, and has enhanced reliability and long refuelling cycle. VBER-300 can also be configured as a transportable FNPP and can be arranged to operate individually or as multi-module plant, increasing the power output by means of scaling up the equipment and with the same reactor system configuration.

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

The reactor core comprises 85 hexagonal fuel assemblies (FAs) which are placed in the reactor cavity in nodes of a regular triangular lattice with a space interval of 236 mm. Pelletized UO_2 fuel with an enrichment of up to 5% licensed and tested in VVER reactors is used. FA of unified design to increase the fuel efficiency is utilized. Each FA contains guide tubes that allow insertion/withdrawal of control rods. Reactor core also uses gadolinium fuel elements, which contain gadolinium in the UO_2 fuel pellet and has the same geometry as the regular fuel pellet.

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

(f) Steam Generator

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

(g) Pressurizer

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

The safety assurance and engineering solutions incorporated in the design focus on accident prevention measures, design simplification, inherent safety; passive safety systems and enhancement of safety against external events (including acts of terrorism); and mitigation of severe accident consequences. The RPV and connecting piping that usually form the primary pressure boundary represent an additional physical barrier. The leak-tight carbon steel containment and protective enclosure with filtration forms the ultimate barrier separating the reactor system from the environment. For all cogeneration applications, the separation of heat exchangers represents a physical barrier to prevent radioactive release.

(b) Decay Heat Removal System

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

(c) Emergency Core Cooling System

The ECCS contains two stages accumulators with different flow-rate characteristics to ensure emergency core cooling for 24 hours, makeup pumps and a recirculation system. If electrical power is available during accidents, makeup pumps and an active recirculation system ensures emergency core cooling beyond the initial 24 hours. The VBER-300 emergency shutdown system consists of the CRDMs, two trains of liquid absorber injection, and two trains of boron control from the make-up system. Emergency residual heat removal system (RHRS) by means of passive cooling channels with water tanks and in-built heat exchangers, ensure reliable cooling up to 72 hours and longer. The system is actuated by passive means—hydraulically operated pneumatic valves. The emergency core cooling accumulators are part of the passive water injection system as injection is done using compressed gas. Containment depressurization systems prevent containment damage and reduce radioactive release in design basis accidents (DBA) and beyond DBAs. A small and medium loss of coolant accidents (LOCA) are prevented by a combination of a sprinkler system, low-pressure emergency injection system, and core passive flooding system.

(d) Containment System

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

6. Plant Safety and Operational Performances

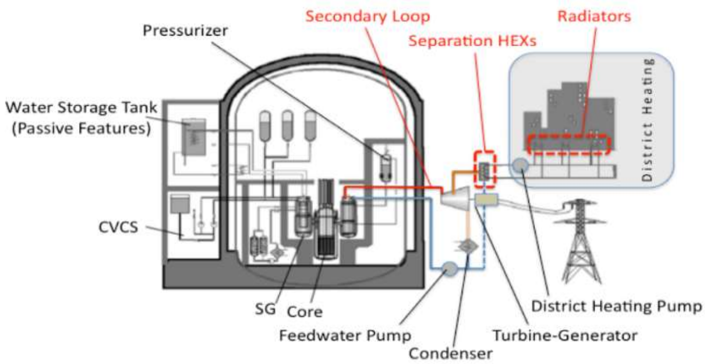
The VBER-300 safety concept is based on the defence-in-depth principles. With the modular configuration, it has increased resistance to impact loads in case of earthquake and aircraft crash. Results of strength analysis under seismic loads of up to 8 points according to the MSK-64 scale carried out for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

7. Plant Layout Arrangement

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

(a) Reactor Building

The inner steel shell is a leak-tight cylindrical enclosure 30 m in diameter that is covered with the semi-spherical dome 15 m in radius and that has an elliptical bottom. The height of the leak-tight enclosure is 47 m. The steel shell is designed for parameters of the maximum DBA with the excess pressure of 0.4 MPa and the temperature of 150°C. The outer protective enclosure is made of one-piece reinforced concrete without preliminary tensioning of the steel and consists of a cylindrical portion of the semi-spherical dome. Building structures of the outer protective enclosure are designed for external accidental exposures, including an aircraft crash and air shock wave.



(b) Balance of Plant

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

(c) Turbine Generator Building

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

8. Design and Licensing Status

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

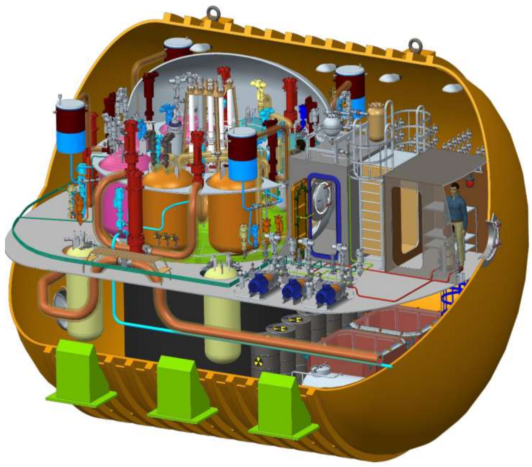
9. Fuel Cycle Approach

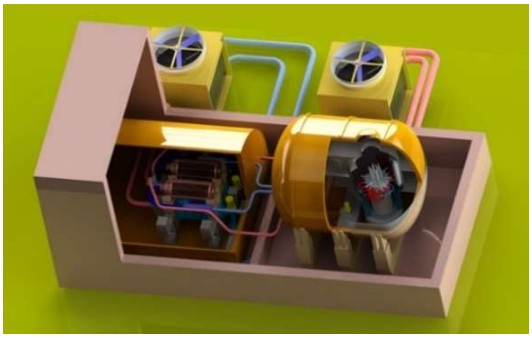
The VBER-300 design concept allows a flexible fuel cycle for the reactor core with standard VVER FAs. The fuel cycles are 3×2 years and 4×1.5 years. The number of FAs in the refuelling batch is either 15 or 30; maximal fuel burnup does not exceed 60 GWd/ton U for the cycle with 30 fresh FAs in the reloading batch and maximum initial uranium enrichment.

10. Development Milestones

2001	Design activities to develop VBER reactors started
2002	Technical and commercial proposal for the two-unit VBER NPP
2004	Preliminary design 1 approved by the Scientific and Technical Board and State Nuclear Supervision Body (GosAtomNadzor)
2006	JSC "Kazakhstan-Russian company "Atomic stations" was established to promote the VBER-300 design.
2007–2009	Technical Assignment for the NPP design and final designs of the reactor plant, automated process control system, and heat-generating plant; feasibility, economy, and investment studies of the VBER-300 RP NPP for the Mangistau Region, Kazakhstan
2007–2008	Development of the 100–600 MW VBER plant
2008–2011	R&D for the VBER-460/600 NPP design
2011–2012	Development of the VBER-600/4 NPP based on the heat exchange loop of the increased capacity
2012–2015	Technical and economic optimization of the VBER-600/4 plant

SHELF-M (NIKIET, Russian Federation)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Integral PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	35.2 / up to 10
Primary circulation	Natural (for 20 % of rated power) / forced
NSSS operating pressure (primary/secondary), MPa	14.7 / 3.7
Core inlet/outlet coolant temperature (°C)	271 / 308
Fuel type/assembly array	Cermet (UO_2 + metallic silumin) / hexagonal
Number of fuel assemblies in the core	241
Fuel enrichment (%)	Up to 19.7
Refuelling cycle (months)	96
Core discharge burnup (GWd/ton)	162.4
Reactivity control mechanism	Control rods driving mechanism
Approach to safety	Combined active and passive
Design life (years)	60
Plant footprint (m^2)	16 800
RPV height/diameter (m)	Power capsule: 10 / 8 RPV: 4 / 2.6
RPV weight (metric ton)	Power capsule: 350 RPV: 75
Seismic design (SSE)	VIII by MSK-64 scale
Fuel cycle requirements/approach	Once-through fuel cycle
Distinguishing features	Autonomous passive decay heat removal system, integral transportable containment
Design status	Basic design

1. Introduction

The SHELF-M is a modernized NPP based on SHELF reactor unit with an increased power capacity of 35.2 MW(t) and a rated power output of up to 10 MW(e) developed based on NIKIET's experience in marine nuclear reactors design. The plant modular design and small containment vessel makes it possible to deliver reactor unit on site fully assembled and tested, minimizing construction costs and time. Refuelling and nuclear waste removal will be carried out at dedicated facilities. Optimized core design allowed to increase refuelling cycle for up to 8 years.

2. Target Application

The SHELF-M is a land-based NPP intended to supply electric power to remote areas with decentralized power supply. The plant does not require local water sources - residual and decay heat is removed by external heat exchangers cooled by atmospheric air. The SHELF-M is designed to be operated with low staffing level with only 15 staff members required to be present of site during normal operation.

3. Design Philosophy

The SHELF-M is based on proven technology derived from NIKIET's experience in design and development of marine propulsion PWR with the addition of advanced passive safety systems. Integral RPV layout with a combined forced and natural primary coolant circulation makes it possible to achieve significant reduction in power unit weight and size. RPV and all necessary normal operation and safety systems are assembled inside a cylindrical power capsule with 8 m in inner diameter and 10 m long. Capsule's shell serves as a containment. Compared to SHELF, SHELF-M has optimized core with extended fuel cycle (up to 8 years) and higher thermal capacity (35.2 MW).

4. Main Design Features

(a) Nuclear Steam Supply System

The SHELF-M power unit implements a traditional two-circuit NSSS. SG provides overheated steam to two turbines located in the engine room adjacent to power capsule room and connected to power capsule with 4 independent steam pipelines. Each pipeline is equipped with shut-off valves installed both inside and outside of power capsule and can be independently isolated without reactor shutdown in case of leaks of primary coolant into the secondary coolant.

(b) Reactor Core

The SHELF-M reactor core consists of 241 cylindrical fuel assemblies (FA) of different types with variable fuel enrichment. Fuel assemblies are placed in nodes of a regular hexagonal lattice. Control rods are located between FA and united in 8 groups. Each fuel assembly contains 55 fuel rods with cermet fuel placed in nodes of regular hexagonal lattice and 6 stationary rods of burnable absorber around fuel rods. Emergency protection rods are located inside 18 fuel assemblies in place of 19 central fuel rods. Cermet fuel consists of uranium dioxide fuel particles embedded in silicon-aluminium (silumin) matrix. Fuel rod shell is made of chromium nickel alloy. High thermal conductivity of fuel matrix as well as absence of zirconium in structural elements of fuel rod improves design safety in transients and prevents hydrogen production in steam oxidation of zirconium alloys.

(c) Reactivity Control

The reactor core contains two independent reactor shutdown systems. Control rods with boron carbide absorber at the top and dysprosium titanate at the bottom provide normal operation reactivity control and shutdown. Emergency protection rods with boron carbide ensure reliable shutdown in case of emergency. The SHELF-M is also equipped with ultimate shutdown system — emergency liquid absorber injection system (ELAIS), which is able to shutdown the core in case of severe accident by injecting boron acid in primary coolant.

(d) Reactor Pressure Vessel and Internals

The RPV accommodates the core with reactivity control rods, the primary coolant circuit, the mechanical filter, the thermal shielding, the steam generator and heat exchangers of emergency decay heat removal system. The RPV has an elliptical bottom, cylindrical shells and two rector covers (central and peripheral). All pipelines are connected to the covers at the top of the RPV to prevent the core drainage in case of LOCA.

(e) Reactor Coolant System

Reactor coolant system is based on conventional two-circuit methodology. The SHELF-M primary circuit cooling under normal operating conditions is done using forced circulation by two primary coolant circulation pumps (PCCP). Cooling by natural circulation is also possible with thermal output at the level 20 % of rated thermal power and during shutdown.

(f) Steam Generator

The SHELF-M utilises once-through steam generator with helically coiled tubes located inside the cylindrical annulus between the RPV and the core barrel. The SG comprises a tubing, collection and distribution chambers above and below the tubes, steam and feedwater lines inside the RPV. SG coil-pipes, alongside with emergency decay heat removal system (EDHRS) heat exchanger (HX) coil-pipes form a single coil with 48 independent modules grouped into 8 sections: 4 for SG and 4 for EDHRS HX, which makes it possible to cut off individual section in case of primary coolant leaks.

(g) Pressurizer

The SHELF-M adopts pressure compensation gas system common in Russian marine PWR. It utilizes an external pressurizer with no active components, such as sprinkler system or electrical heating. Pressurizer system consists of 5 separate vessels: gas collector, intermediate pressurizer, end pressurizer and two expansion vessels. Pressurizer is connected to the reactor central cover by a pipeline with a restrictor to reduce coolant leak rate in large-break accidents.

(h) Primary pumps

The SHELF-M utilizes two electrical primary coolant circulation pumps (PCCP) located on peripheral cover of the RPV.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The SHELF-M is designed with combined passive and active safety systems that comprises emergency decay heat removal system (EDHRS), emergency reactor cooling system (ERCS), emergency liquid absorber injection system (ELAIS) and overpressure protection systems.

(b) Decay Heat Removal System

During normal shutdown, decay heat is removed from the reactor core by forced circulation of primary coolant to SG and then by secondary coolant to the condenser. In case of emergency (primary or secondary circuit pumps failure, secondary circuit loss of coolant etc), emergency decay heat removal system (EDHRS) is used to cool down the core. The system consists of 4 independent loops with two passive circuits in each loop (intermediate circuit and low-boiling circuit) with natural circulation of coolants and stand-alone heat exchangers between circuits. The heat is transferred to the atmospheric air via heat exchangers, located on the roof of the reactor building.

(c) Emergency Core Cooling System

The emergency reactor cooling system is designed to supply the in-vessel circulation circuit with water during accidents with loss of the primary circuit integrity. The system is based on passive principals and doesn't require any manual or automatic activation. The system contains 4 vessels with coolant under high pressure, which is passively injected into RPV in case of primary coolant pressure drop.

(d) Containment System

The SHELF-M utilizes two containment systems. The first one — safety vessel, located inside power capsule, that encloses RPV and all primary circuit equipment. The second — power capsule shell, which contains safety vessel with RPV, steam and feedwater pipelines, EDHRS intermediate heat exchangers, reactor unit auxiliary equipment and tanks with liquid and solid radioactive waste. Both containment system are designed to withstand pressure and temperature of primary coolant in case of large-break accidents. Both containments are unvisited during normal operation and filled with inert gas (nitrogen) with absolute pressure below atmospheric to prevent minor leaks of radioactive media and reduce the probability and possible consequences of fire inside containment volume. Both containments are equipped with overpressure protection systems.

6. Plant Safety and Operational Performances

The electric power of a single SHELF-M unit is up to 10 MW(e), and the thermal power is 35.2 MW(t). The current supplied to the consumer system is alternate and three-phase with voltage 0.4 kV ± 2 %, frequency 50 Hz ± 1 Hz). The NPP base operation mode is power operation in range from 20 to 100 % of rated power with the capability to vary the consumed power daily and annually. The power increase and decrease rate is 1 % (with forced primary coolant circulation). Design capacity factor - 0.9. Refuelling outage time is 30 days once in 8 years.

7. Instrumentation and Control System

The automated process control system (APCS) of a NPP with SHELF-M reactor unit is to control major and auxilary electricity generation processes in all modes of the unit operation:

1. Normal operation comprises of phased automated initiation, operation at steady power levels in a range of 20 to 100 % of rated power with forced primary coolant circulation, operation at steady power levels at 20 % of rated power with natural coolant circulation, switchover from natural primary coolant circulation to forced circulation and scheduled automated deactivation.
2. Anticipated operational occurrences like emergency power reduction and operation with a decreased steam supply due to failures of the reactor facility's key components or feedwater supply and steam receipt systems.
3. Emergency: emergency deactivation in the event of reactor facility parameters deviation beyond the safe operation limits or in the event of equipment failures leading to the safe operation limits being violated.

8. Plant Layout Arrangement

The SHELF-M plant consists of main building, administrative building, 4 stand-alone fan cooling towers, 4 emergency diesel generators, start-up boiler module, water storage tanks, fresh and spent fuel open storage area and auxiliary modules. The main building is based on lightweight steel frame structure and houses the reactor hall with power capsule and external concrete protection, and the turbine hall. The building is 48×48×25 m is size and is designed as quick assembly building based on prefabricated structures, that allows to significantly reduce construction costs and time.

The administrative building is assembled from standard containers and houses administrative offices and control room.

9. Testing Conducted for Design Verification and Validation

Experimental activities are built and in operation. The SHELF-M reactor unit fuel rods reactor tests are under way.

10. Design and Licensing Status

Licensing process to be started in 2025.

11. Fuel Cycle Approach

The SHELF-M implements a standard PWR fuel cycle. All fuel assemblies are discharged at the end of refuelling cycle. On-site (for FOAK plant) and remote (for subsequent plants) refuelling strategies are designed. For the FOAK plant, the fresh and spent open air fuel storage is located on site.

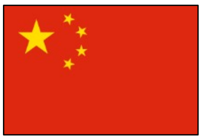
12. Waste Management and Disposal Plan

Liquid radioactive waste management system is located inside power capsule. The system is designed to gather radioactive leaks from reactor cooling systems and storage liquid waste inside power capsule.
Gaseous radioactive waste system is designed to remove gaseous radioactive waste from within the power capsule during the scheduled maintenance (one a year). Nitrogen from within power capsule passes through filters, where radioactive aerosols are absorbed and stored.

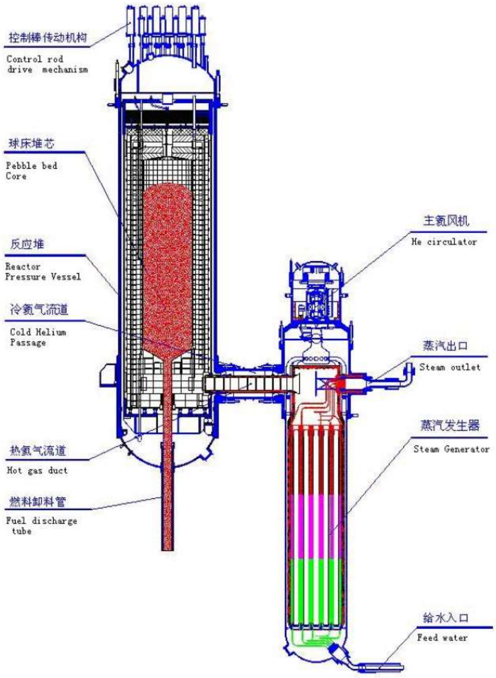
13. Development Milestones

2018	Start of the project SHELF-M (achieved)
2019	Conceptual design (achieved)
2023	Preliminary design (ongoing)
2025	Licensing, detailed design, start of construction
2030	Operation testing

PART II. HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS



HTR-PM (Tsinghua University, China)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	INET, Tsinghua University, China
Reactor type	Modular pebble bed HTGR
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	2 × 250 / 210
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	7 / 13.25
Core Inlet/Outlet Coolant Temperature (^oC)	250 / 750
Fuel type/assembly array	Spherical elements with coated particle fuel
Number of fuel assemblies in the core	420 000 (in each reactor module)
Fuel enrichment (%)	8.5
Core Discharge Burnup (GWd/ton)	90
Refuelling Cycle (months)	On-line refuelling
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Combined active and passive
Design life (years)	40
Plant footprint (m^2)	256 100
RPV height/diameter (m)	25 / 5.7 (inner)
RPV weight (metric ton)	800
Seismic Design (SSE)	0.2 g
Fuel cycle requirements / Approach	LEU, open cycle, spent fuel intermediate storage at the plant
Distinguishing features	Inherent safety, no need for offsite emergency measures
Design status	In operation

1. Introduction

In 1992, the China Central Government approved the construction of the 10 MW(t) pebble bed high temperature gas cooled test reactor (HTR-10) in Tsinghua University's Institute of Nuclear and New Energy Technology (INET). In 2003, the HTR-10 reached its full power operation. After that, INET has completed many experiments on the HTR-10 to verify crucial inherent safety features of modular HTRs, including (i) loss of off-site power without scram; (ii) main helium blower shutdown without scram; (iii) withdrawal of control rod without scram; and (iv) helium blower trip without closing outlet cut-off valve.
The next step of HTR development in China began in 2001 when the high-temperature gas-cooled reactor--pebble-bed module (HTR-PM) project was launched. The first concrete of the HTR-PM demonstration power plant was poured on 9 December 2012, in Rongcheng, Shandong Province. In support of manufacturing first of a kind equipment and licensing, large scale engineering test facilities were constructed, and all tests have been completed. The civil work of the nuclear island's buildings has been completed in 2016 with the first of two reactor pressure vessels installed in March 2016. Currently all major equipment has been manufactured and already installed. The power plant is scheduled to start power generation in 2021.

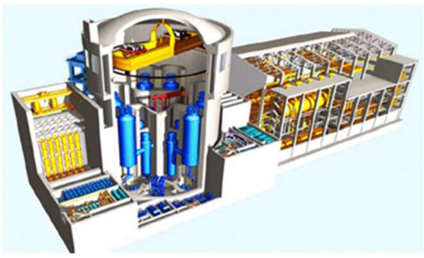
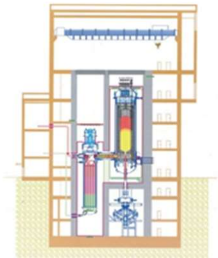
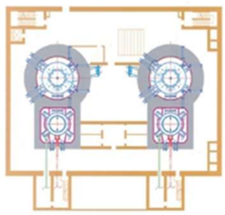
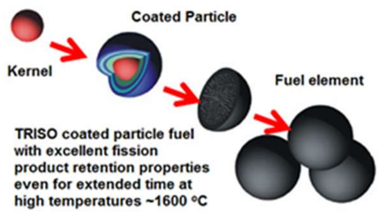
2. Target Application

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

A standard design has been finished for the 600 MW(e) multi-module HTR-PM600 nuclear power plant, which consists of six reactor modules or Nuclear Steam Supply System (NSSS) modules coupling to one steam turbine. Each NSSS module has the same design as the HTR-PM demonstration plant, with independent safety systems and shared non-safety auxiliary systems. The footprint of a multi-module HTR-PM600 plant is not substantially different from that of a PWR plant generating the same power. Future sites have been identified for possible deployment. There is a big market demand for high temperature steam supply from HTR-PM/HTR-PM600 in China.

3. Design Philosophy

The HTR-PM consists of two NSSS modules coupled with a 210 MW(e) steam turbine, as shown below. Each NSSS module includes a reactor that contains reactor pressure vessel, graphite, carbon, and metallic reactor internals; a steam generator; a main helium blower; and a hot gas duct. The thermal power of each reactor module is 250 MW(t), the helium temperatures at the reactor core inlet/ outlet are 250/750°C, and steam parameters is 13.25 MPa/567°C at the steam turbine entrance.

4. Main Design Features

(a) Reactor Core and Power Conversion Unit

The primary helium coolant works at 7.0 MPa with the rated mass flow rate of 96 kg/s. Helium coolant enters the reactor in the bottom area inside the RPV with an inlet temperature of 250ºC. Helium coolant flows upward in the side reflector channels to the top reflector level where it reverses the flow direction and flow into the pebble bed in a downward flow pattern. Bypass flows are introduced into the fuel discharge tubes to cool the fuel elements there and into the control rod channels for control rods cooling. Helium is heated up in the active reactor core and then is mixed to the average outlet temperature of 750ºC and then flows to the steam generator.

(b) Fuel Characteristics

Illustrated above, fuel elements are spherical ones. Every fuel element contains 7 grams of heavy metal. The enrichment for the equilibrium core is 8.5% of ^{235}U. Uranium kernels of about 0.5 mm diameter are coated by three layers of pyro-carbon and one layer of silicon carbon. Coated fuel particles are dispersed in matrix graphite, 5 cm in diameter. Surrounding the fuel containing graphite matrix is a 5 mm thick graphite layer.

(c) Fuel Handling System

The operation mode of HTR-PM adopts continuous fuel loading and discharging: the fuel elements drop into the reactor core from the central fuel loading tube and are discharged through a fuel extraction pipe at the core bottom. Subsequently, the discharged fuel elements pass the burn-up measurement device one by one. When a fuel sphere reaches the target burnup they will be discharged into the spent fuel storage tank, otherwise they are re-inserted into the reactor to pass the core once again.

(d) Reactivity Control

Two independent shutdown systems are installed: a control rod system and a small absorber sphere (SAS) system, both placed in holes of the graphite side reflector. Reactivity control is performed using 24 control rod assemblies, and 6 SAS shutdown systems serve as a reserve shutdown system. The control rods are used as a regulating group during normal plant operation and for emergency shutdown. Furthermore, turning off the helium circulator is also efficient for reactor trip. Drop of all control rods can achieve long term shutdown. The SAS system is used to reduce the shutdown temperature for the purpose of in-service inspection and maintenance. Absorber material of control rods and small absorbers is B_4C.

(e) Reactor Pressure Vessel and Internals

The primary pressure boundary consists of the reactor pressure vessel (RPV), the steam generator pressure vessel (SGPV) and the hot gas duct pressure vessel (HDPV), which all are housed in a concrete shielding cavity. The three primary pressure vessels are composed of SA533-B steel as the plate material and (or) the 508-3 steel as the forging material. The ceramic structures surrounding the reactor core consist of the inner graphite reflector and outer carbon brick layers. The whole ceramic internals are installed inside a metallic core barrel, which itself is supported by the RPV. The metallic core barrel and the pressure vessel are protected against high temperatures from the core by the cold helium borings of the side reflector, which act like a shielding temperature screen.

5. Safety Features

The HTR-PM is designed with the following safety features: (1) radioactive inventory in the primary helium coolant is very small during normal operation conditions, and even if released there is no need to take any emergency measures; (2) for any reactivity accident or loss of coolant accident, the rise of the fuel elements' temperature will not cause a significant additional release of radioactive substances; (3) the consequences of water or air ingress accidents depend on the quantity of such ingresses. The ingress processes and the associated chemical reactions are slow, and can readily be terminated within several dozens of hours (or even some days) by taking very simple actions. And air ingress accident is classified as design extension condition.
The HTR-PM incorporates the inherent safety principles of the modular HTGR. The lower power density, good coated particle fuel performance and a balanced system design ensures that the fundamental safety functions are maintained. A large negative temperature coefficient, large temperature margin, low excess reactivity (due to on-line refuelling) and control rods ensure safe operation and limit accident temperatures. The decay heat is passively removed from the core under any designed accident conditions by natural mechanisms, such as heat conduction or heat radiation, and keeps the maximum fuel temperature below 1620°C, so as to contain nearly all of the fission products inside the SiC layer of the TRISO coated fuel particles. This eliminates the possibility of core melt and large releases of radioactivity into the environment.
Another feature of the HTR-PM design is the long-time period of accident progression due to the large heat capacity of fuel elements and graphite internal structures. It requires days for the fuel elements to reach the maximum temperature when the coolant is completely lost.

(a) Engineered Safety System Approach and Configuration

When accidents occur, a limited number of reactor protection actions shall be called upon by the reactor protection system. No or very limited actions through any systems or human interventions are foreseen after the limited reactor protection actions are activated. The limited reactor protection actions shall be to trip the reactor and the helium circulator, to isolate the primary and secondary systems. When there is large leak or rupture of steam generator heat transfer tubes, a dumping system is designed to minimize the amount of water ingress into the primary circuit.

(b) Reactivity control

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

(c) Reactor Cooling Philosophy

Normally the reactor is cooled by steam generating system. Under accident conditions, the main helium blower shall be stopped automatically. Because of the low power density and the large heat capacity of the graphite structures, the decay heat in the fuel elements can dissipate to the outside of the reactor pressure vessel by means of heat conduction and radiation within the core internal structures, without leading to unacceptable fuel temperature. And the fuel temperature increase in this phase will compensate accident reactivity and shutdown the reactor automatically via negative temperature feedback. The decay heat shall be removed to heat sink passively by reactor cavity cooling system (RCCS). Even if the RCCS fails, the decay heat can be removed by transferring it through the concrete structure of reactor cavity while the temperatures of fuel elements are under design limit.

(d) Containment Function

Retention of radioactivity materials is achieved through multi-barriers. The fuel elements with coated particles serve as the first barrier. The fuel elements used for HTR-PM have been demonstrated to be capable of retaining fission products within the coated particles under temperatures of 1620ºC which is not expected for any plausible accident scenarios. The second barrier is the primary pressure boundary which consists of the pressure vessels of the primary components. The vented low-pressure containment (VLPC) is designed according to ALARA principle to mitigate the influence of accidents, consisting of the reactor cavity inside reactor building and some auxiliary systems such as sub-atmosphere ventilation, burst disc and filters.

(e) Chemical control

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

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control Systems

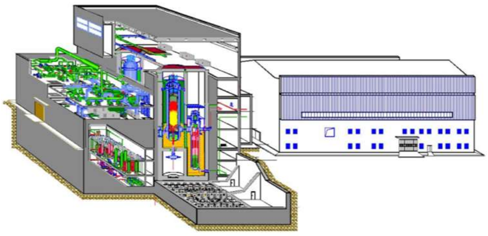
The instrumentation and control system of HTR-PM is similar to those of normal PWR plant. The two reactor modules are controlled in a coordinated manner to meet various operational requirements.

8. Plant Layout Arrangement

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

9. Design and Licensing Status

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



HTR-PM Main Control Room

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

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

12. Development Milestones

2001	Launch of commercial HTR-PM project
2004	Standard design of HTR-PM started
2006	HTR-PM demonstration power plant approved as one of National Science and Technology Major Projects
2006	Huaneng Shandong Shidaowan Nuclear Power Co., Ltd, the owner of the HTR-PM, established by the China Huaneng Group, the China Nuclear Engineering Group Co. and Tsinghua University
2006-2008	Basic design of HTR-PM completed
2009	Assessment of HTR-PM PSAR completed
2012	First Pour of Concrete of HTR-PM
2013	Fuel plant construction started
2014	Qualification irradiation tests of fuel elements completed
2015	Civil work of reactor building finished
2016	RPV and core barrel etc. delivered, installation of main components ongoing
2017	Fuel plant achieved expected production capacity
Q4/2020	Startup commissioning test of primary circuit
2021.8	Operation License
2021.9	First criticality
2021.12.20	Grid connection
2022	Full power operation

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

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

1. Introduction

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

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

2. Target Applications

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

3. Design Philosophy

StarCore is designed to operate in the harshest environment in remote locations anywhere in the world. To this end the reactors are contained in steel-walled concrete silo structures 30 metres below grade. Each reactor silo has two supporting silos with their access hatches in the refuelling room, also below grade. This room is filled with helium at low pressure during refuelling operations, which are carried out by an automatic system.

StarCore plants have between one and six reactors per plant, giving us a range of outputs between 14 MW(e) and 360 MW(e). The plants are load-following, and can also output High Temperature Gas or Steam, with the outputs dynamically changed as required.

The plants meet all planned Remote Siting Requirements, which include that they must be: Inherently Safe; Passively Secure; Load Following; Fully Automated; Have a Remote Shutdown (Intervention) Capability; and after suitable qualification (subject to local regulations) be operated with a zero-radius exclusion zone.

The plants also have 69 KV 100 km HVDC transmission lines included in the cost of the plant. The StarCore Build-Own-Operate-Decommission Business Plan includes all capital costs, licensing, operating and decommissioning logistics, thus eliminating the cost of entry to all countries and communities. The HVDC is inverted to local frequency standards.

4. Main Design Features

(a) Reactor Core

The prismatic core is made up of hexagonal graphite blocks that are 360 mm across the flats and 793 mm long. Cylindrical fuel compacts (26 mm diameter and 39 mm long) are inserted into holes drilled in the graphite blocks, and burnable (neutron) poison elements will also be inserted as needed. The helium flow will be through vertical holes drilled in the blocks. The number of blocks used depend on the output of the core.

The helium coolant enters at the bottom of the core, flows up the outside including through the reactivity control mechanisms, and then down through the core prismatic blocks. There are automatic bi-stable valves at the inlet and outlet, which seal off the core in the event of pressure loss.

(b) Reactivity Control

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

(c) Reactor Pressure Vessel and Internals

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

(d) Reactor Coolant System

The first stage coolant Energy Transport System (ETS-1) is helium at 7.3 MPa; this then transfers energy to ETS-2 through IHX-1 which contains nitrogen at 6.7 MPa. Nitrogen is used in this system at this pressure to minimize delta-P across IHX-1 and allow compact exchanger designs to be sued; in addition, the pressure gradient ensures that any gas migration will be in the direction of ETS-2.

The Nitrogen in ETS-2 then passes to an aero-derivative gas turbine that has an annular heat exchanger in place of the usual burner cans, and the turbine exhaust gas at 300 ºC can be used for district heating or cooling. The energy in ETS-2 can also be sent to a heat exchanger that provides high-temperature gas or steam to external plants.

5. Safety Features

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

6. Plant Safety and Operational Performance

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

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

7. Instrumentation and Control Systems

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

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

8. Plant Layout Arrangement

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

9. Design and Licensing Status

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

10. Fuel Cycle Approach

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

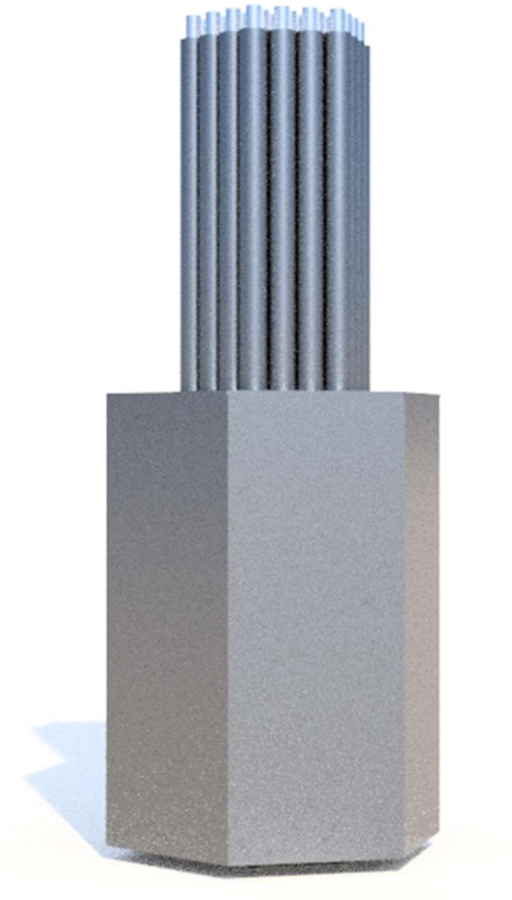
11. Waste Management and Disposal Plan

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

12. Development Milestones

209- 2013	Preliminary studies and initial pre-conceptual design.
2013-2017	Pre-conceptual design phase, technology validation and vendor contracts and qualification
2017-2020	Fund raising and PPA contracts
2021-2026	Projected deployment (start of construction to commissioning)

JIMMY (JIMMY ENERGY SAS, France)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JIMMY ENERGY SAS France
Reactor type	High temperature gas-cooled reactor
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 to 20 / n.a.
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	1.5 / 3.0
Core inlet/outlet coolant temperature (°C)	300 / 700
Fuel type/assembly array	UCO TRISO particles packed into pellets
Number of fuel assemblies in the core	19
Fuel enrichment (%)	19.5
Refuelling cycle (months)	No refuelling
Core discharge burnup (GWd/ton)	110
Reactivity control mechanism	B_4C Control rods
Approach to safety systems	Passive
Design life (years)	10 to 20
Plant footprint (m^2)	150
RPV height/diameter (m)	5 / 5
RPV weight (metric ton)	15
Seismic design (SSE)	Zone 4
Fuel cycle requirements/approach	No refuelling, no on-site storage
Distinguishing features	Compact, low-weight design; simplicity of manufacturing, flexibility of operation; intrinsic passive safety
Design status	Detailed design

1. Introduction

Jimmy is a high-temperature gas-cooled reactor with a thermal power of 10 to 20MW(t), designed to provide competitive, low-carbon industrial process heat. The reactor is based on UCO TRISO particles, a graphite moderator and a helium coolant (primary loop). A CO_2 secondary loop allows to deliver heat up to 550°C to industrial processes, selecting the secondary heat exchanger according to industrial needs. Valuing the existing HTGR return of experience, the approach consists in delivering a simple, safe and robust heat generator. As a result, the design relies mostly on well-tested solutions, as well on a passive safety demonstration, to optimize the time- and cost-to-market. The first system will be available in 2026, with a first client site already identified.

2. Target Application

Jimmy targets the industrial heat market, where industrial suffer from both economic and environmental pressure to reduce their use of fossil fuels, while no alternative solution available. Jimmy will allow to replace industrial gas-burners and provide competitive, low-carbon industrial process heat. The design and small reactor footprint and limited capacity (up to 20MWt) allows to integrate into most industrial sites, with one or several generators in a row. The primary target is the French market, especially steam-consumers such as the Agro or Chemistry industry, with needs for steam up to 550°C. After massive deployment to process heat applications in and outside of France, the main applications are district heating (offices, residential buildings…) and hydrogen production.

3. Design Philosophy

The main goal of the design is to provide a simple, safe and financially viable alternative to industrial heat gas-burners. This goal is achieved by using HTGR well-tested solutions (especially regarding safety) and then optimizing the design to optimize safety, reduce cost, and allow easy manufacturing, transportation and assembly. For instance, the use of a secondary CO_2 loop allows to distinctly separate the nuclear perimeter from the industrial zone in terms of safety. Furthermore, the manufacturing and assembly aspects have been taken into account since the beginning of the detailed design: design teams are in contact with presented suppliers to screen commercially available components and integrate them in the design. It ensures that a viable industrial scheme is available and to anticipate the orders of long-lead items, as well as to optimize instruction time by safety authorities.

4. Main Design Features

(a) Power Conversion

The Jimmy generator is designed to provide heat only, i.e., there is no energy conversion into electricity. A primary exchanger allows to transfer heat from the primary helium-based loop to the secondary CO2-based loop. The CO2-based loop pressure is 3.0 MPa and temperature is 600°C. A secondary exchanger is then selected, according to each industrial site's specific needs (fluid, temperature, pressure, etc.).

(b) Reactor Core

The reactor core consists of 19 prismatic fuel assemblies inside a graphite reflector. Theses fuel assemblies are designed to be assembled in a factory and transportable inside existing fuel transportation packages.

(c) Fuel Characteristics

The fuel are pellets of HALEU TRISO particles that are conformed to the AGR program envelope to reach the expected burnup.

(d) Reactivity Control

Reactivity control is made with 24 independent B_4C control rods and B4C neutronic poisons that cross the core or the reflector.

(e) Reactor Pressure Vessel and Internals

The Jimmy reactor does not feature a pressure vessel but 19 different pressure tubes that constitute the core. This whole core is then contained into a containment vessel. This geometry facilitates the transport and assembly of the reactor on its installation site.

(f) Reactor Coolant System

The primary coolant is a helium pressurized circuit, whose circulation is forced on the cold branch by a helium blower. The circuit divides and reunites before and after the pressure tubes of the core.

(g) Secondary System

The secondary circuit connects to the primary circuit through a plate heat exchanger. It contains pressurized carbon dioxide whose circulation is also forced.

(h) Steam Generator

Jimmy's generator does not directly produce steam, but its industrial client may produce steam based on the heat it gets from the secondary system through a secondary exchanger.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Jimmy's safety is mainly based on the leverage of three properties that the HTR can achieve: passive decay heat removal, TRISO particles robustness and high reactivity coefficients. Based on these intrinsic properties, most of the safety systems are around redundant and highly independent control rods.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

During normal operation, the reactor is cooled thanks to the industrial site that use the transferred heat for its process.
During accidental operation, the graphite plays the role of a thermal absorber. The low power density of the reactor easily ensure that the fuel remains below the limit temperature that it can endure.

(c) Spent Fuel Cooling Safety Approach / System

Calculations and modelling show that spent fuel from a Jimmy's reactor cools passively and does not require temporary on-site storage in a spent fuel pool.

(d) Containment System

The design's containment system is based on the "defence in depth" philosophy and includes a set of technical measures aimed at preventing environmental exposure to radiation and radioactive substances from the reactor. The set of barriers includes:
- The coating layers of TRISO particles;
- The graphite of the core
- The primary system
- The containment of the building

(e) Chemical Control

The main chemical control is a helium purification system that supplies a highly pure helium and monitor the health of the core

6. Plant Safety and Operational Performances

The goal of Jimmy is to ensure that large radioactivity release frequency is less than 10^{-7}/reactor year.

7. Instrumentation and Control System

The I&C system is under design and does not contain any major disruption

8. Plant Layout Arrangement

The layout of Jimmy's reactor is a building of w: 12m * l: 8m * h: 15 m. For security reasons, walls are thick to prevent aggression and only few systems go through the wall (mainly the RCCS and the secondary system that reaches the industrial line). All the access for maintenance and installation are on the roof to stay out of an easy reach.

9. Testing Conducted for Design Verification and Validation

Jimmy's philosophy is to use already tested technologies. Thus, most of the required tests are validation tests on the behaviour on well modelled pieces. Jimmy does not plan to conduct a specific test program.

10. Design and Licensing Status

Interaction with Nuclear Safety Authority – Formal licensing process under way;
(Launch of licensing procedure with the first regulatory application (Dossier d'Options de Sûreté) in April 2020, allowing the French Safety Authority to start the instruction)

Site permit for FOAK plant – To be developed.
(Application to be submitted in June 2023 – First site for FOAK has already been identified, with seismic and other environmental studies underway)

11. Fuel Cycle Approach

In order to simplify the design and the operation, and to maximize safety, no refuelling will be done. After the 10-to-20-year life (depending on the nominal power chosen by the client), the entire vessel is extracted from the generator and replaced. Jimmy has a long-term strategy to foster fuel recycling between its systems.

12. Waste Management and Disposal Plan

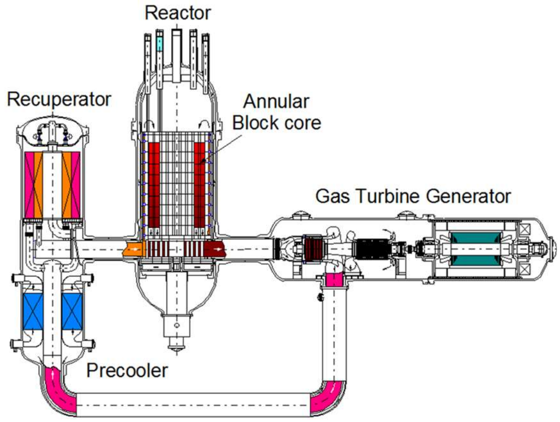
By design, the Jimmy reactor does not release any waste during normal operation. The only possible source of waste would be the leaks from the helium loop, which is filtered and stored.
At the end of reactor lifetime, the reactor produces waste that can be collected and disposed of in accordance with applicable regulation, relying on existing French waste management infrastructures with minimal adaptations.

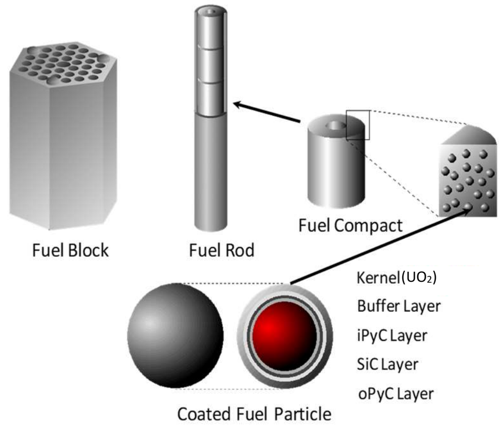
13. Development Milestones

2021	Creation of the company Seed fundraising allowing to finance Basic Design and first regulatory application
Done so far in 2022	Basic Design completed Peer review of Basic Design Feasibility study for a first industrial client (Agro industry) completed Launch of licensing procedure with the first regulatory application (Dossier d'Options de Sûreté) and instruction by the French safety authority (ASN)
Current goals	Preliminary Design completed Peer review of Preliminary Design First orders placed for long-lead items First design team completed (20+ engineers)
2023	Detailed Design completed Peer review of Detailed Design

	Second licensing application (Demande de Décret d'Autorisation de Création) and beginning of the instruction by the French safety authority Manufacturing Design completed Peer review of Manufacturing Design Orders placed for medium-lead items
2024	Orders placed for all items Equipment qualification Communication among local population
2025-2026	Module pre-assembly (out of site) On-site land development Obtaining of licensing approval On-site reactor assembly & tests Fuel loading & tests Commissioning

GTHTR300 (JAEA Consortium, Japan)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JAEA, MHI, Toshiba/IHI, Fuji Electric, KHI, NFI, Japan
Reactor type	Prismatic HTGR
Coolant/moderator	Helium / graphite
Thermal/electrical capacity, MW(t)/MW(e)	<600 / 100~300
Primary circulation	Forced by gas turbine
NSSS Operating Pressure (primary/secondary), MPa	7 / 7
Core Inlet/Outlet Coolant temperature (°C)	587-633 / 850-950
Fuel type/assembly array	UO_2 TRISO ceramic coated particle
Number of fuel assemblies in the core	90
Fuel enrichment (%)	14
Refuelling Cycle (months)	48
Core Discharge Burnup (GWd/ton)	120
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Active and passive
Design life (years)	60
Plant footprint (m^2)	~250 x 250 (4-reactor plant)
RPV height/diameter (m)	23 / 8
RPV weight (metric ton)	~1 000
Seismic Design (SSE)	>0.18g automatic shutdown
Fuel cycle requirements / approach	Uranium once through (initially)
Distinguishing features	Multiple applications of power generation, cogeneration of hydrogen, process heat, steelmaking, desalination, district heating
Design status	Basic design

1. Introduction

The 300 MW(e) Gas Turbine High Temperature Reactor (GTHTR300) is a multi-purpose, inherently safe and site-flexible small modular reactor (SMR) being developed by Japan Atomic Energy Agency (JAEA) for commercialization in 2030s. As a Generation-IV technology, the GTHTR300 offers important advances over current light water reactors. The coolant temperature is significantly higher in the range of 850-950°C. Such high temperature capability as proven in the JAEA’s HTTR test reactor operation enables a wide range of applications. The design employs a direct-cycle helium gas turbine to simplify the plant by eliminating water and steam systems while generating power with enhanced efficiency of 45-50%.

2. Target Applications

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

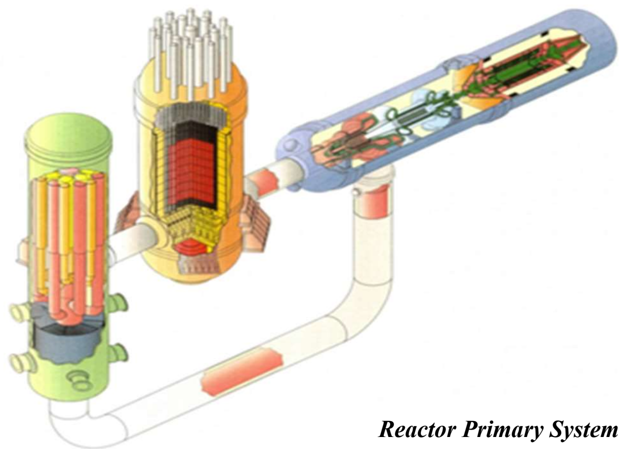
3. Design Philosophy

The overall goal of the GTHTR300 design and development is to provide a family of system options capable of producing competitive electricity, hydrogen, desalination, other products, and yet deployable in the near term. The development of the multiple systems simultaneously does not necessarily suggest having investment and risk multiplied. Rather, the development requirement is minimized by pursuing system simplicity, economic competitiveness and originality, namely the SECO design philosophy.

4. Main Design Features

(a) Power Conversion

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.

(c) Fuel Characteristics

The fuel design is coated fuel particle of less than 1 mm in diameter. Each particle consists of a UO_2 kernel coated by four layers of low and high density pyro-carbon and silicon carbide. The all ceramic particle fuel is heat resistant up to 1600ºC. Approximately 10 000 particles are packaged into a compact of the size of a thumb. The compacts are then assembled into graphite-clad fuel rods. The fuel rods are inserted into the bore holes of a hexagonal graphite fuel block of about 1 m long and 41 cm across, where the annulus formed between the fuel rod and the bore hole provides coolant flow channels. The fuel blocks are loaded into the reactor core. The more fuel blocks are placed in the core, the higher the power output of the reactor.

(d) Fuel Handling System

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

(e) Reactivity Control

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

(f) Reactor Pressure Vessel and Internals

The reactor core consists of graphite hexagonal blocks, one-third of which are fuel blocks arranged in an annular region while the other two-thirds are reflector blocks arranged inside and outside of the fuel region. Each fuel block has 57 coolant holes with fuel rods forming annular-shaped coolant channels. A permanent reflector which surrounds side replaceable reflectors contains coolant channels. The helium coolant from the reactor inlet is introduced to the channel. A core barrel is installed between reactor internals and reactor pressure vessel to support internal structure laterally.

(g) Reactor Coolant System

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

5. Safety Features

The reactor delivers fully inherent safety due to three enabling design features:
- Ceramic coated particle fuel maintains containment integrity under a temperature limit of 1600°C.
- Reactor helium coolant is chemically inert and thus absent of explosive gas generation or phase change.
- Graphite-moderated reactor core provides negative reactivity coefficient, low-power density, and high thermal conductivity.

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

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal System / Reactor Cooling Philosophy

GTHTR300 design removes the core decay and residual heat from the outside surface of the reactor vessel by the natural convection and radiation, and to transfer it to an ultimate heat sink in the operational states and in the accident conditions so that the design limits for fuel, the reactor coolant pressure boundary and structures important to safety are not exceeded.

(c) Spent Fuel Cooling Safety Approach / System

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

(d) Containment System

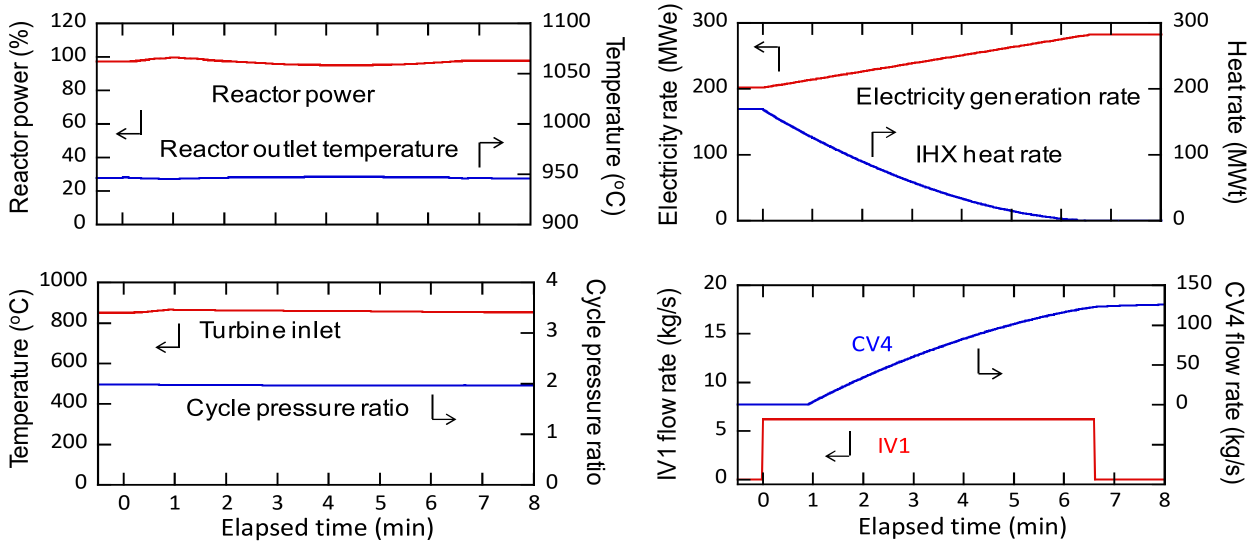
GTHTR300 employs a vented confinement rather than a conventional high-pressure, airtight containment used in LWRs. The confinement is designed to meet requirements designated in safety design. The confinement is designed to release helium coolant blown out from primary system in case of depressurized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

(e) Chemical Control

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

6. Plant Safety and Operational Performance

The ability to follow variable power and heat loads is simulated as shown in the following figure. The simulation shows the plant response to an electric demand increase of 5%/min with corresponding reduction in heat rate, which is the maximum required for ramp load follow. The reactor remains at 100% power at all times. Starting from a base cogeneration ratio of 203 MW(e) electricity and 170 MW(t) net heat, the turbine power generation is increased to follow the increasing electric demand by increasing primary coolant inventory with opening of inventory valve IV1. The IHX heat rate is lowered by lowering secondary loop flow of the IHX. The power sent out to external grid increases to 276 MW(e) in as little as a few minutes. The pressure in the reactor increases to 7 MPa from 5 MPa.



7. Instrumentation and Control System

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

8. Plant Layout Arrangement

The reactor building is a below-grade, steel concrete structure housing 4 units of reactor systems consists of subsystems including reactor module, gas turbine module and heat exchanger modules.

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

The design is developed at pre-licensing basic design stage. The design and development are planned to be concluded to prepare for the demonstration plant operation in 2040s.

11. Fuel Cycle Approach

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

12. Waste Management and Disposal Plan

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

13. Development Milestones

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

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

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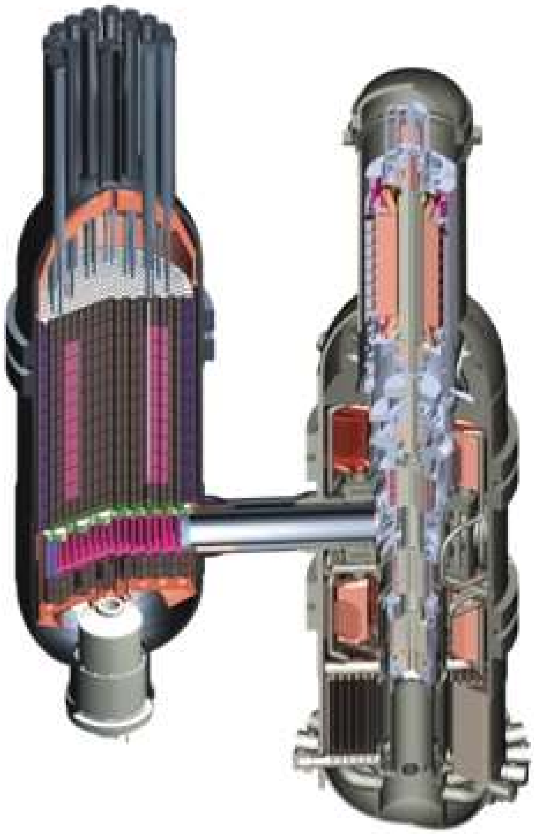


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", Russian Federation
Reactor type	Modular Helium Reactor
Coolant/moderator	Helium /graphite
Thermal/electrical capacity, MW(t)/MW(e)	600 / 288
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	7.2 / -
Core Inlet/Outlet Coolant Temperature (°C)	490 / 850
Fuel type/assembly array	Coated particle fuel in compacts, hexagonal prism graphite blocks of 0.36 m
Number of fuel assemblies in the core	~1020
Fuel enrichment (%)	14-18% LEU or WPu
Core Discharge Burnup (GWd/ton)	100-720 (depends on fuel type)
Refuelling Cycle (months)	25
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Hybrid (active and passive)
Design life (years)	60
Plant footprint (m^2)	9110
RPV height/diameter (m)	29 / 8.2
RPV weight (metric ton)	950
Seismic Design (SSE)	8 points (MSK 64)
Fuel cycle requirements / Approach	Standard LEU or WPu / No recycling; high fission product retention
Distinguishing features	Inherent safety characteristics; no core melt; high temperature process heat capabilities; small number of safety systems
Design status	Preliminary design completed; key technologies are being demonstrated

1. Introduction

The gas turbine modular helium reactor (GT-MHR) couples a HTGR with a Brayton power conversion cycle to produce electricity at high efficiency. As the reactor unit can produce high coolant outlet temperatures, the modular helium reactor system can also efficiently produce hydrogen, e.g. by high temperature electrolysis or thermochemical water splitting.

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

The modular helium reactor design proved unit modularity with a wide power range of a module (from 200 to 600 MW(t)) and NPP power variation as a function of module number. This provides good manoeuvring characteristics of the reactor plant (RP) for regional power sources.

2. Target Application

The GT-MHR can produce electricity at high efficiency (approximately 48%). As it can produce high coolant outlet temperatures, the modular helium reactor system can also efficiently produce hydrogen, e.g. by high temperature electrolysis or thermochemical water splitting.

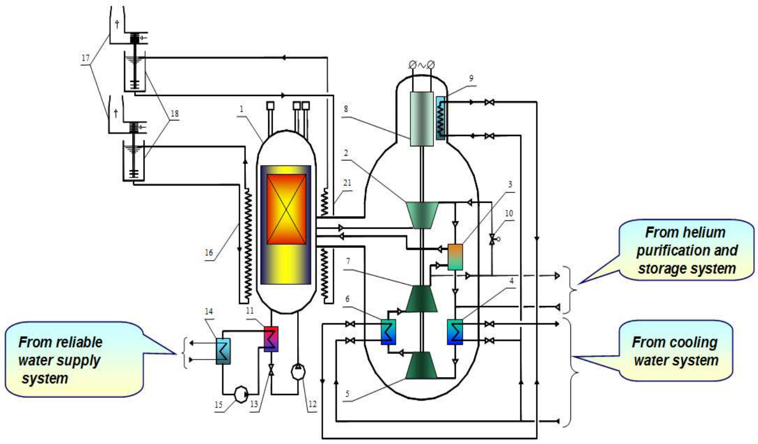
3. Design Philosophy

The GT-MHR direct Brayton cycle power conversion system contains a gas turbine, an electric generator and gas compressors. The layout can be seen in the figure above. The GT-MHR gas turbine power conversion system has been made possible by utilizing large, active magnetic bearings, compact, highly effective gas to gas heat exchangers and high strength, high temperature steel alloy vessels. The use of the gas-turbine cycle application in the primary circuit leads to a minimum number of reactor plant systems and components. The GT-MHR safety design objective is to provide the capability to reject core decay heat relying only on passive (natural) means of heat transfer without the use of any active safety systems. The GT-MHR fuel form presents formidable challenges to diversion of materials for weapon production, as either fresh or as spent fuel.

4. Main Design Features

(a) Power Conversion System Flow Diagram

1. Reactor
2. Gas Turbine
3. Recuperator
4. Pre-Cooler
5. Low Pressure Compressor
6. Intercooler
7. High Pressure Compressor
8. Generator
9. Generator cooler
10. Bypass Valve
11~15. SCS Components
16. Surface Cooler Of reactor
17. Air Ducts
18. Heat Exchanger with Heat pipes



The Brayton power conversion with direct gas turbine contains a gas turbine, an electric generator and gas compressors. The GT-MHR gas turbine power conversion system has been made possible by utilizing large, active magnetic bearings, compact, highly effective gas to gas heat exchangers and high strength, high temperature steel alloy vessels.

(b) Reactor Core and Fuel Characteristics

Coated particle fuel is used. The fuel kernel (U or Pu oxide) is coated by a first porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with thousands of compacts inserted into the fuel channels of the hexagonal prismatic graphite blocks or fuel assemblies. The coated particles will contain almost all fission products with temperatures up to 1600°C. About 1 billion fuel particles of the same type were manufactured and tested in Russia. The standard fuel cycle for the commercial GT–MHR utilizes low enriched uranium (LEU) but alternative cycles including the disposition of plutonium were also studied in detail.

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

5. Safety Features

Safety objectives for the GT-MHR are achieved, first, by relying on the *inherent safety features* incorporated in the plant design. The design features, which determine the inherent safety and ensure thermal, neutronic, chemical and structural stability of the reactor unit, are the following:

Using of helium coolant, which has some specific properties. During plant operation, helium is not affected by phase transformations, does not dissociate, is not activated and has good heat transfer properties. Helium is chemically inert, does not react with fuel, moderator and structural materials. There are no helium reactivity effects;

Core and reflector structural material is high-density reactor graphite with substantial heat capacity and heat conductivity and sufficient mechanical strength that ensures core configuration preservation under any accident;

Nuclear fuel in the form of coated fuel particles with multilayer ceramic coatings, which retain integrity and effectively contain fission products under high fuel burnup and high temperatures;

The temperature and power reactivity coefficients are negative, what provides the reactor safety in any design and accident conditions.

Safety is ensured by application of passive principles of system actuation. The decay and accumulated heat is removed from the core through reactor vessel to reactor cavity cooling system and then to atmosphere by natural physical processes of heat conductivity, radiation, convection without excess of fuel safe operation limits including LOCA, in case of all active circulation systems and power sources failure.

(a) Engineered Safety System Approach and Configuration

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

(b) Reactor Cooling Philosophy

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

(c) Containment Function

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

6. Plant Safety and Operational Performances

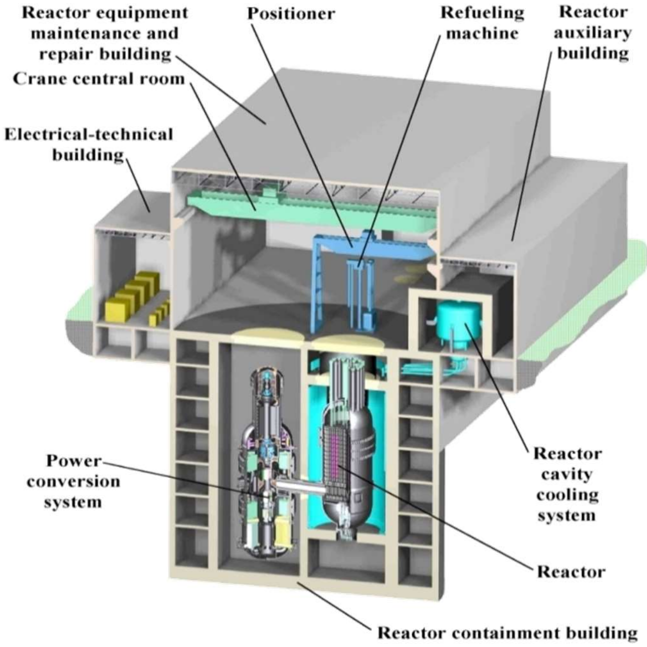
All safety systems are designed with two channels. Fulfilment of the regulatory requirements on safety, proven by a compliance with both deterministic and probabilistic criteria, is secured by an exclusion of the active elements in a channel or by applying the required redundancy of such active elements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

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement

The plant layout is shown on the right.



9. Design and Licensing Status

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

10. Fuel Cycle Approach

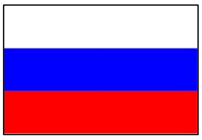
The GT-MHR fuel cycle approach is a once through mode without reprocessing. Fuel handling operations are performed using the protective containers to avoid fuel assembly damage and radioactive product release. Appropriately shielded containers are provided to protect the personnel against radiation impacts during dismantling of the reactor unit components at fuel reloading. These measures are also applied at spent fuel management. Spent fuel shows good proliferation resistance characteristics, producing less materials of proliferation concern (total plutonium and ^{239}Pu) per unit of energy produced.

11. Waste Management and Disposal Plan

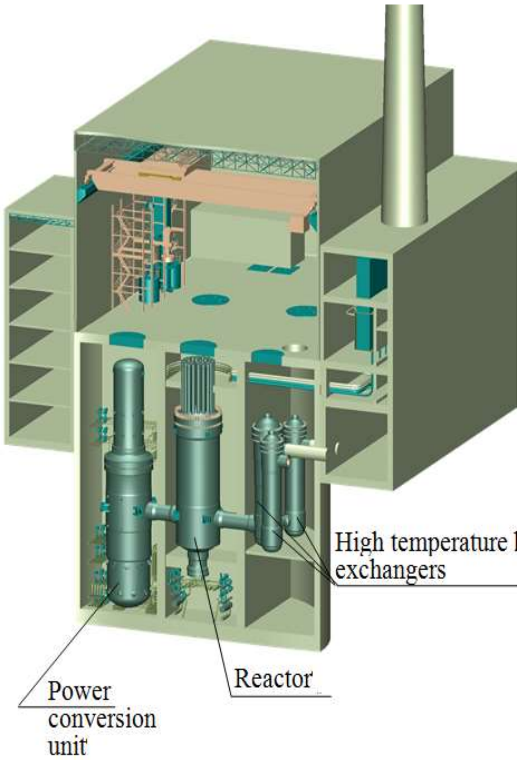
Facilities for long-term storage of spent nuclear fuel (SNF) and solid/solidified radioactive waste (RW) are included in the complex of a GT-MHR commercial 4-unit NPP. The capacity of the designed SNF storage is determined from the condition of capability to store fuel unloaded from the NPP for 10 years.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

12. Development Milestones

1993	Minatom / General Atomics MOU on joint GT-MHR development for commercial units
1994	Russia proposes to build GT-MHR at Seversk to burn Russian WPu
1996	Framatome& Fuji Electric join the GT-MHR program
1997	Conceptual design completed
1998	GT-MHR becomes an option within the US/RF Pu disposition strategy
1999	Conceptual design review by international group of experts
2000	Work started on preliminary design
2002	Project review by Minatom of Russia and US DOE experts
2002	Reactor plant preliminary design completed
2003	Begin demonstration of key technologies
2014	Completion of demonstration of key technologies for Pu-fuelled core
Since 2014	Use of principal reactor unit design features as a basis for MHR-T design (U-fuelled option)



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



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC "Afrikantov OKBM", Russian Federation
Reactor type	Modular helium high-temperature reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	4 × 600 / 4 × 205.5
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	7.5 / –
Core Inlet/Outlet Coolant Temperature (°C)	578 / 950
Fuel type/assembly array	Coated particle fuel in compacts, hexagonal prism graphite block of 0.36m
Number of fuel blocks	~ 1 020
Fuel enrichment (%)	< 20
Core Discharge Burnup (GWd/ton)	125
Fuel cycle (months)	30
Reactivity control mechanism	Control rods
Approach to safety systems	Hybrid (active and passive)
Design life (years)	60
RPV height/diameter (m)	32.8 / 6.9
RPV weight (metric ton)	950
Seismic design	8 points (MSK 64)
Fuel cycle requirements / Approach	Standard LEU / No recycling; high fission product retention
Distinguishing features	Multi-module HTGR dedicated to hydrogen production / high temperature process heat application.
Design status	Conceptual design

1. Introduction

The MHR-T reactor/hydrogen production complex makes use of the basic GT-MHR reactor unit design as the basis for a multi-module plant for hydrogen production. The hydrogen production through the steam methane reforming process or high-temperature solid oxide electrochemical process is performed by coupling the plant with the modular helium reactor(s). The use of modular helium reactor units makes the system flexible and allows the possibility to use various power unit schemes: with gas-turbine cycle (GT-MHR design), steam-turbine cycle and with the circuit supplying high-temperature heat to industrial applications (this design). The modular high temperature gas-cooled reactor unit possess salient safety features with passive decay heat removal providing high level of safety even in case of total loss of primary coolant.

2. Target Application

The most perspective technologies for Russia are hydrogen production through the steam methane reforming process or high-temperature solid oxide electro-chemical process coupled with a modular helium reactor called MHR-T. The chemical-technological sector with steam methane reforming is considered as an option for short-term perspective.

3. Design Philosophy

The MHR-T complex includes the chemical-technological sector (hydrogen production sector) and the infrastructure supporting its operation. The chemical-technological sector includes hydrogen production process lines, as well as systems and facilities supporting their operation.
The following processes are considered as the basic processes for the chemical-technological sector: (i) steam methane reforming; and (ii) high-temperature solid oxide electrochemical process of hydrogen production from water. Heat shall be transferred directly from primary coolant to chemical-technological sector medium in a high-temperature heat exchanger. The key component of chemical-technological sector medium circulating through the high-temperature heat exchanger is water steam. The high-temperature electrolysis option allows the consideration of two- and three-circuit plant configurations. The technical concept is based on:

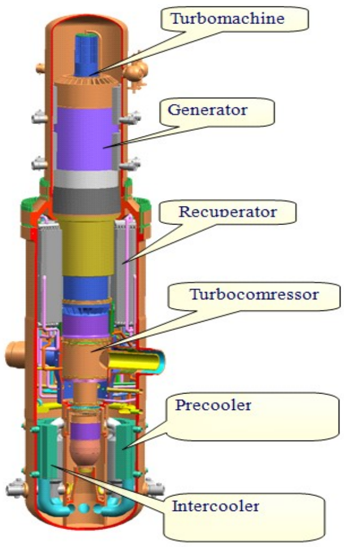
- Modular helium-cooled reactors with typical high level of inherent safety;
- Fuel cycle based on uranium dioxide in the form of multi-layer coated particles, high burnup and burial of fuel blocks unloaded from the reactor without any additional processing;
- Electromagnetic bearings operating almost without friction and applied in various technical areas;
- Highly efficient high-temperature compact heat exchangers, strong vessels made of heat resistant steel.

The thermal energy generated in the reactor is converted to chemical energy in a thermal conversion unit (TCU) where, in the MHR-T option with methane reforming, the initial steam-gas mixture is converted to hydrogen-enriched converted gas (mixture of water steam, CO, H_2, CO_2, and CH_4) in the course of a thermochemical reaction.

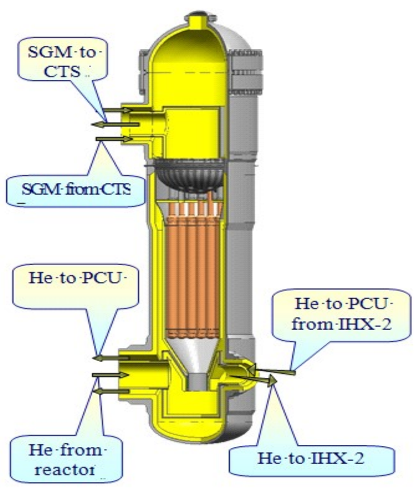
4. Main Design Features

(a) Power Conversion System

A power conversion unit is integrated in a single vessel and includes a vertical turbomachine, highly efficient plate heat exchanger, and coolers. A high-temperature heat exchanger (IHX) for the MHR-T option with methane steam reforming is an integral part of the thermal conversion unit and is partitioned as a three-stage heat exchanger. Arrangement of the heat exchanger sections along the primary coolant flow is parallel, and downstream of the coolant in the chemical-technological sector (CTS) is sequential. Each section is designed as a separate heat exchanger consisting of several modules.



Power Conversion Unit



High Temperature Heat Exchanger Section

The material of the heat exchange surface of the module is a heat-resistant alloy. The turbomachine includes a generator and turbo-compressor mounted on a single shaft on electromagnetic suspension. The gas turbine cycle of power conversion with the helium turbomachine, heat exchanger and intercooler provide a thermal efficiency of 48%.

(b) Reactor Core and Fuel Characteristics

Coated particle fuel is used. The fuel kernel (U oxide) is coated by a first porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with thousands of compacts inserted into the fuel channels of the Hexagonal Prism graphite blocks or fuel assemblies.
The coated particles will contain almost all fission products with temperatures up to 1600ºC. About 1 billion fuel particles of the same type were manufactured and tested in Russia. The standard fuel cycle is to utilize low enriched uranium (LEU) in a once through mode. The MHR-T show good proliferation resistance characteristics. It produces less total plutonium and ^{239}Pu (materials of proliferation concern) per unit of energy produced. The fuel form presents formidable challenges to diversion of materials for weapons production, as either fresh or as spent fuel.

(c) Reactor Coolant System

Working media in circulation circuits are helium of the primary circuit and steam-gas mixture (SGM) in the CTS. The peculiarity of heat exchangers for the production of hydrogen by methane reforming process is the transfer of heat from high temperature helium of the primary circuit to the chemically aggressive medium of hydrogen production circuit.

(d) Reactivity Control

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

(e) Reactor Pressure Vessel and Internals

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

5. Safety Features

The safety features of the MHR-T reactor are the same as for the GT-MHR. Safety objectives for the MHR-T are achieved, first of all, by relying on the inherent safety features incorporated in the plant design. The design features are as follows:

- Using helium as the coolant. During operation, helium is not affected by phase transformations. It does not dissociate and has good heat transfer properties. Helium is chemically inert. It does not react with fuel, moderator and structural materials. There are no helium reactivity effects;
- Core and reflector structural material is high-density reactor graphite with substantial heat capacity and heat conductivity and sufficient mechanical strength that maintain core configuration integrity;
- Nuclear fuel in the form of coated fuel particles with multilayer ceramic coatings, which retain integrity and effectively contain fission products under high fuel burnup and high temperatures;
- The temperature and power reactivity coefficients are negative that provides the reactor safety in any design and accident conditions.

Safety is ensured by application of passive principles of system actuation. The decay and accumulated heat is removed from the core through reactor vessel to reactor cavity cooling system and then to atmosphere by natural physical processes of heat conductivity, radiation, convection In case of LOCA with failures of all active circulation systems and power sources, operation limits of the fuel are not exceeded.

(a) Engineered Safety System Approach and Configuration

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

(b) Reactor Cooling Philosophy

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

(c) Containment Function

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

6. Plant Safety and Operational Performances

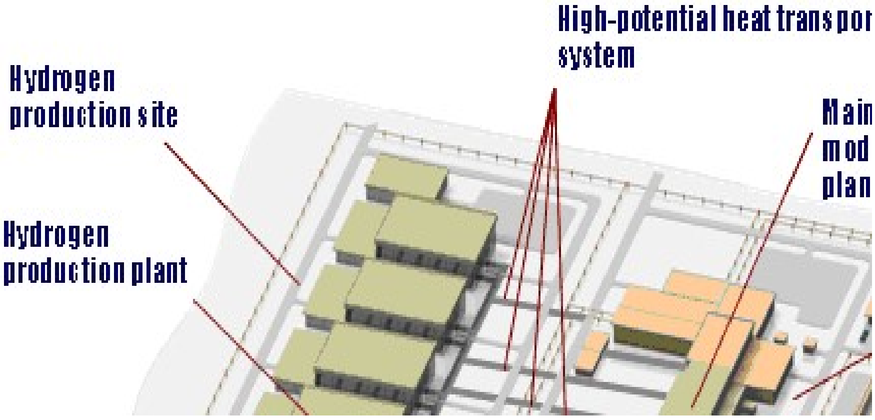
All safety systems are designed with two channels. Fulfilment of the regulatory requirements on safety, proven by a compliance with both deterministic and probabilistic criteria, is secured by an exclusion of the active elements in a channel or by applying the required redundancy of such active elements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

7. Instrumentation and Control Systems

The MHR-T NPP control and support safety systems (CSS) are intended to actuate the equipment, mechanisms and valves, localizing and support safety systems in the pre-accidental conditions and in accidents; to monitor their operation; and to generate control commands for the equipment of normal operation systems used in safety provision algorithms. The CSS are based on the principles of redundancy, physical and functional separation, and safe failure.

8. Plant Arrangement

The main components of each NPP module are arranged in isolated premises of the underground containment of the NPP main building. The chemical-technological sector equipment is arranged outside the containment of the NPP main building. The MHR-T energy-technological complex is designed for a specific site on the basis of design solutions selected with account of climatic conditions typical of central Russia and special external impacts such as seismicity, aircraft crash, air shock wave. The interfaces between the four-module NPP and the chemical-technological sector must be designed to except faults that could cause failure of more than one MHR-T module. The main reactor equipment is arranged in a vertical vessel located in a separate cavity parallel to the power conversion unit and high-temperature heat exchanger vessels.



9. Design and Licensing Status

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

10. Fuel Cycle Approach

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

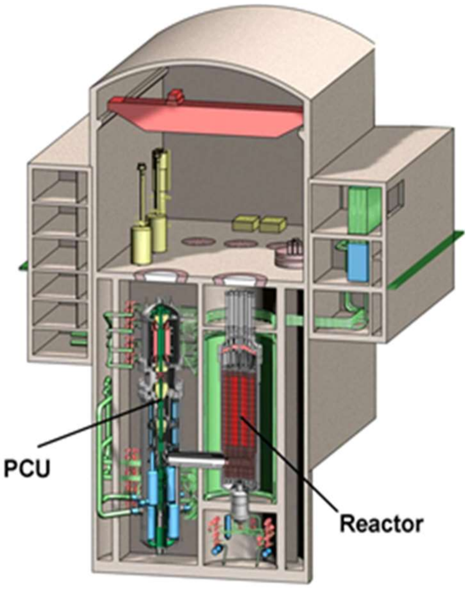
11. Waste Management and Disposal Plan

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

12. Development Milestones

2001	Pre-conceptual proposal
2005	Conceptual design completed
2007	Elaboration of technical requirements
2017	Feasibility study of plant application for large-scale hydrogen production
2020	Development of basic design solutions for chemical-technological part of the facility

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



MHR-100 GT layout

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC “Afrikantov OKBM”, Russian Federation
Reactor type	Modular helium reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	215 / 25 – 87 (depends on configuration)
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	4 – 5 / depends on configuration
Core Inlet/Outlet Coolant Temperature (^{o}C)	490 – 553 / 795 – 950 (depends on configuration)
Fuel type/assembly array	Hexagonal prism graphite blocks with coated particle fuel
Number of fuel assemblies	About 1600 blocks with more than 600 fuel compacts in each block
Fuel enrichment (%)	< 20%
Reactivity control mechanism	Control rod insertion
Approach to safety systems	Hybrid (active and passive)
Design life (years)	60
Plant footprint (m^2)	Depends on configuration
RPV height/diameter (m)	Similar to 1000 MW(e) LWR
RPV weight (metric ton)	Similar to 1000 MW(e) LWR
Seismic Design (SSE)	8 points (MSK 64)
Fuel cycle requirements / Approach	Once through U; Pu and Th cycle also possible
Distinguishing features	A multipurpose reactor for cogenerations of electricity, heat and hydrogen; high-temperature heat supply to oil refinery plant
Design status	Conceptual design

1. Introduction

The designs are based on the global experience in the development of experimental HTGR plants. Russia has more than 40 years of experience in the development of HTGR plants of various power (from 100 to 1000 MW) and for various purposes. It has established the experimental facilities for the R&D work, the fuel element and material fabrication technology, including the fabrication and mastering of pilot equipment, and various activities in hydrogen generation technology. Today, conventional power stations, with electric capacity ~300 MW(t), are deployed all over the territory of Russia. These are adapted to regional systems and provide the electric power needs. They consist mainly of cogeneration plants producing about 40% of electric power and 85% of heat generation. Analysis shows that SMRs with HTGR have therefore good prospects to add to or replace these regional generation. Innovative nuclear power systems to be implemented on this basis are therefore considered as an important area of the nuclear power industry development up to the middle of the century. Based on study of the power market development and demands, pre-conceptual work is performed for commercial MHR100 with modular helium reactor and several power conversion layouts for various power-industrial applications. The following options of MHR100 for industrial applications were studied:

- Electricity and district heat productions by core thermal power conversion to electric one in direct gas-turbine Brayton cycle – MHR-100 GT;
- Electricity and hydrogen generations by high-temperature steam electrolysis – MHR-100 SE;
- Hydrogen generation by steam methane reforming method MHR-100 SMR;
- High-temperature heat supply to oil refinery plant – MHR-100 OR.

2. Target Application

The MHR-100 is intended for regional power generation and heat production in the Russian Federation. A single reactor unit design can be implemented in various plant configurations.

Major Technical Parameters of MHR-100 GT		
Parameters	**Power Mode**	**Cogeneration Mode**
Reactor heat capacity (MW)	215	215
Net power generation efficiency (%)	46.1	25.4
Helium temperature at reactor inlet/outlet (°C)	558 / 850	490 / 795
Low-pressure helium temperature at recuperator inlet (°C)	583	595
Helium flow rate through the reactor (kg/s)	139.1	134
Helium bypass flow rate from HPC outlet to recuperator outlet at high-pressure side (kg/s)	–	32.2
Helium pressure at reactor inlet (MPa)	4.91	4.93
Expansion ratio in turbine	2.09	1.77
Generator/TC rotation speed (rpm)	3000 / 9000	3000 / 9000
PCU cooling water flow rate (kg/s)	804	480
Delivery water temperature at inlet/outlet (°C)	–	70 / 145

3. Design Philosophy

The reactor power and its design are universal for all the different power and process heat options with only the coolant parameters that are different. The reactor unit power level (215 MW(t)) was selected according to: (i) the regional power industry and district heat supply needs; (ii) the manufacture needs in high- and medium-temperature heat supply for technological processes; and (iii) process capabilities of national enterprises in fabrication of RP main components including vessels.

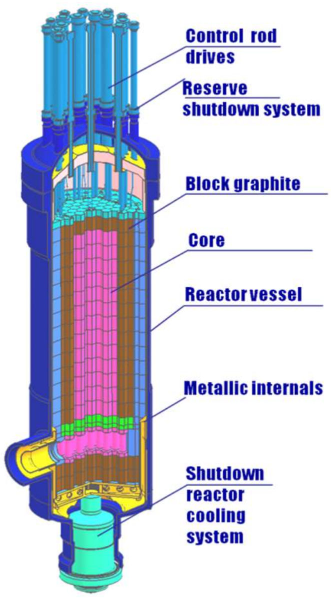
4. Specific Design Features

(a) Power Conversion Unit

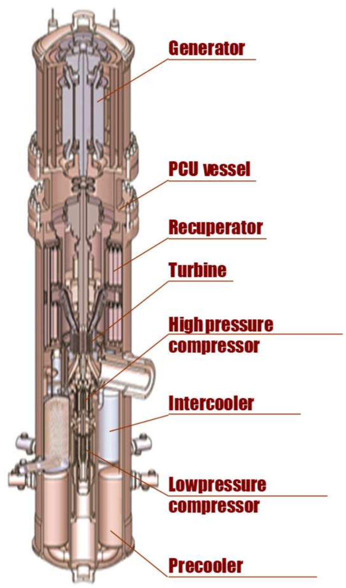
A power conversion unit is integrated in a single vessel and includes a vertical turbomachine, highly efficient heat exchanger, and coolers. The turbomachine includes a generator and turbo-compressor mounted on a single shaft on electromagnetic suspension. Gas turbine cycle of power conversion with the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

(b) Reactor Core and Fuel Characteristics

Coated particle fuel is used. The fuel kernel (U oxide) is coated by a first porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with hundreds of compacts inserted into fuel channels of each hexagonal prism graphite block (0.2 m × 0.65 m height). The core is a cylindrical arrangement of vertical stacks of fuel blocks (fuel columns). The standard fuel cycle is to utilize low enriched uranium (LEU) in a once through mode.



Reactor Unit



Power Conversion Unit

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal / Reactor Cooling Philosophy

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

(c) Containment Function

Passive localization of radioactivity is provided by the containment designed for the retention of helium-air fluid during accidents with primary circuit depressurization. The containment is also designed for external loads due to seismic impacts, aircraft crash, air shock wave, etc. Activity release from the containment into the environment is determined by the containment leakage level, which is about 1%vol/d day at 0.5 MPa.

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control

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

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

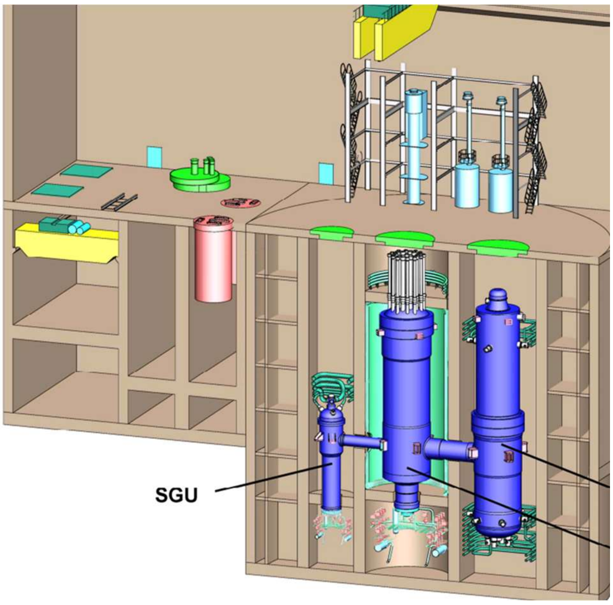
The modular reactor consists of the core with hexahedral prismatic fuel assemblies, uses helium as a coolant, and has inherent self-protection. The technical concept of studied reactor plant MHR-100 is based on:

- Modular high-temperature helium-cooled reactors with typical high level of inherent safety;
- Fuel cycle with fuel in the form of multilayer UO_2-based coated fuel particles, high burnup and possibility to dispose the spent fuel blocks without additional reprocessing;
- High-performance high-temperature compact heat exchangers, high-strength casings of heat-resistant steel;
- Direct gas-turbine cycle with high-efficiency recuperation and intermediate coolant cooling;

- Experience in high-efficiency gas turbines application in power engineering and transport;
- Electromagnetic bearings used in power conversion system.

The coolant is circulated in the primary loops by the main gas circulator or by the power conversion unit (PCU) turbomachine (TM) compressors. The MHR-100 option consists of power and process parts. The power part is unified to the maximum for all options and is a power unit consisting of a reactor unit with a thermal power of 215 MW and a gas-turbine PCU for power generation and (or) heat-exchange units, depending on the purpose. The process part of MHR-100 is either a process plant for hydrogen production or circuits for high-temperature heat supply to various technological processes, depending on the purpose.

The unified gas-turbine PCU is planned to be used in MHR-100 GT and MHR-100 SE options. Vertical oriented TM is the main feature of the PCU and consists of the turbo-compressor (TC) and generator with rotors, which have different rotation speed of 9000 rpm and 3000 rpm respectively. Electromagnetic bearings are used as the main supports. The generator is located in air medium outside the helium circuit. The PCU pre-cooler and intercooler are arranged around TC while the recuperator is located at the top of the vessel above the hot duct axis. Waste heat from the primary circuit is removed in the PCU pre-cooler and intercooler by the cooling water system, then in dry fan cooling towers to atmospheric air.

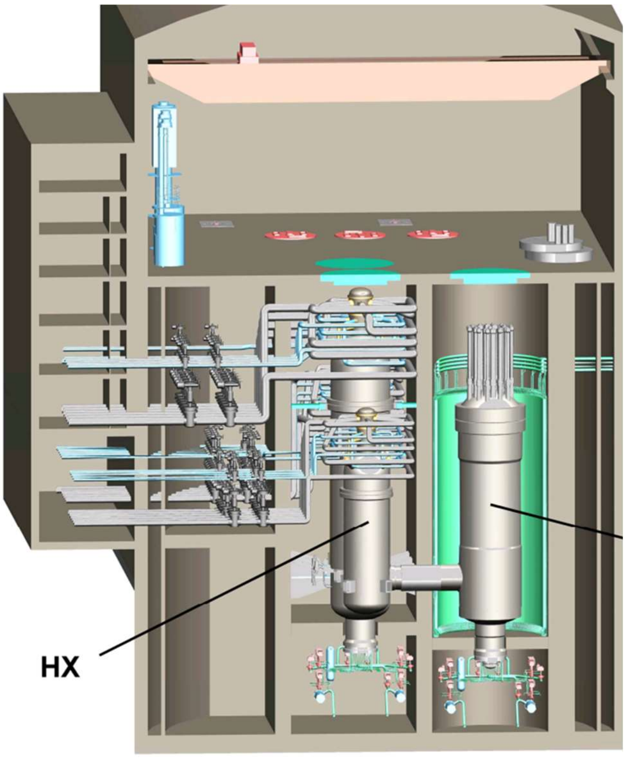


MHR-100 SE

Heat exchange blocks are intended to transfer heat power from the reactor to the consumer of power-technological applications. Depending on the working fluid, process type and probability of radioactivity ingress to the process product and contamination of equipment with radioactive products, two- or three-circuit RP configuration can be used. So, two circuit configurations are used in MHR-100 SE NPP for hydrogen generation and in MHR-100 SMR for steam methane reforming. Water steam is the main component of process fluid in these processes. The analysis shows that the effects of hydrogen-bearing products ingress in potential accidents with depressurization of the steam generator (SG) or high-temperature heat exchanger (HX) are reliably checked by reactor control and protection systems.

MHR-100 OR-based power source for heat supply to petrochemical applications and oil refinery plants has three-circuit thermal configuration. Heat from RP is transferred to the consumer via high-temperature intermediate helium-helium HX (IHX) and intermediate helium circuit and then to network circuit of petrochemical applications. This decision restricts radioactivity release to the network circuit and provides radiological purity of the process product and minimum contamination of the primary circuit with process impurities.

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

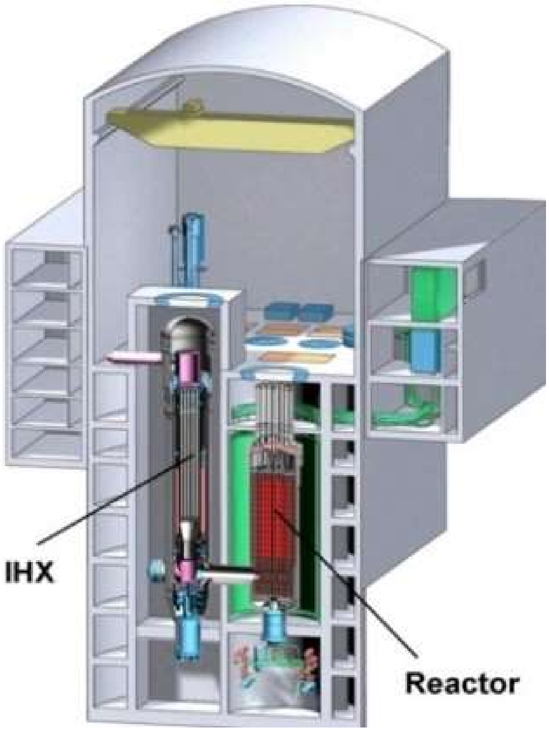


MHR-100 SMR

9. Design and Licensing Status

Optimization of reactor core design. Feasibility study of MHR-100-SMR plant application for large-scale hydrogen production, technical and economical evaluation of the plant potential to supply hydrogen to the expected market. Studies of safety issues, with the emphasis on mutual influence of nuclear and hydrogen production components of the facility.

MAJOR TECHNICAL PARAMETERS	
Parameters	**Values**
Reactor heat capacity (MW)	215
Helium temperature at reactor inlet/outlet (°C)	300 / 750
Helium flow rate through the reactor (kg/s)	91.5
Helium pressure at reactor inlet (MPa)	5.0
IHX capacity (MW)	217
Primary/secondary helium flow rate through IHX (kg/s)	91.5 / 113
Primary helium temp. at IHX inlet/outlet (°C)	750 / 294
Secondary helium temp. at IHX inlet/outlet (°C)	230 / 600
Secondary helium pressure at IHX inlet (MPa)	5.50



MHR-100 OR

10. Fuel Cycle Approach

The MHR-100 fuel cycle approach is a once through mode without reprocessing. Fuel handling operations are performed using the protective containers to avoid fuel assembly damage and radioactive product release. Appropriately shielded containers are provided to protect the personnel against radiation impacts during dismantling of the reactor unit components at fuel reloading. These measures are also applied at spent fuel management.

11. Waste Management and Disposal Plan

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

12. Development Milestones

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

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

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

1. Introduction

When the PBMR® was defined in the mid-1990s, it was based on the industrially demonstrated technology of the German *Arbeitsgemeinschaft Versuchsreaktor* (AVR) proof of concept reactor, and the German Thorium High Temperature Reactor (THTR300) commercial scale reactor, integrated with a direct Brayton cycle helium turbine power conversion unit, based on existing industrial gas turbines.

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

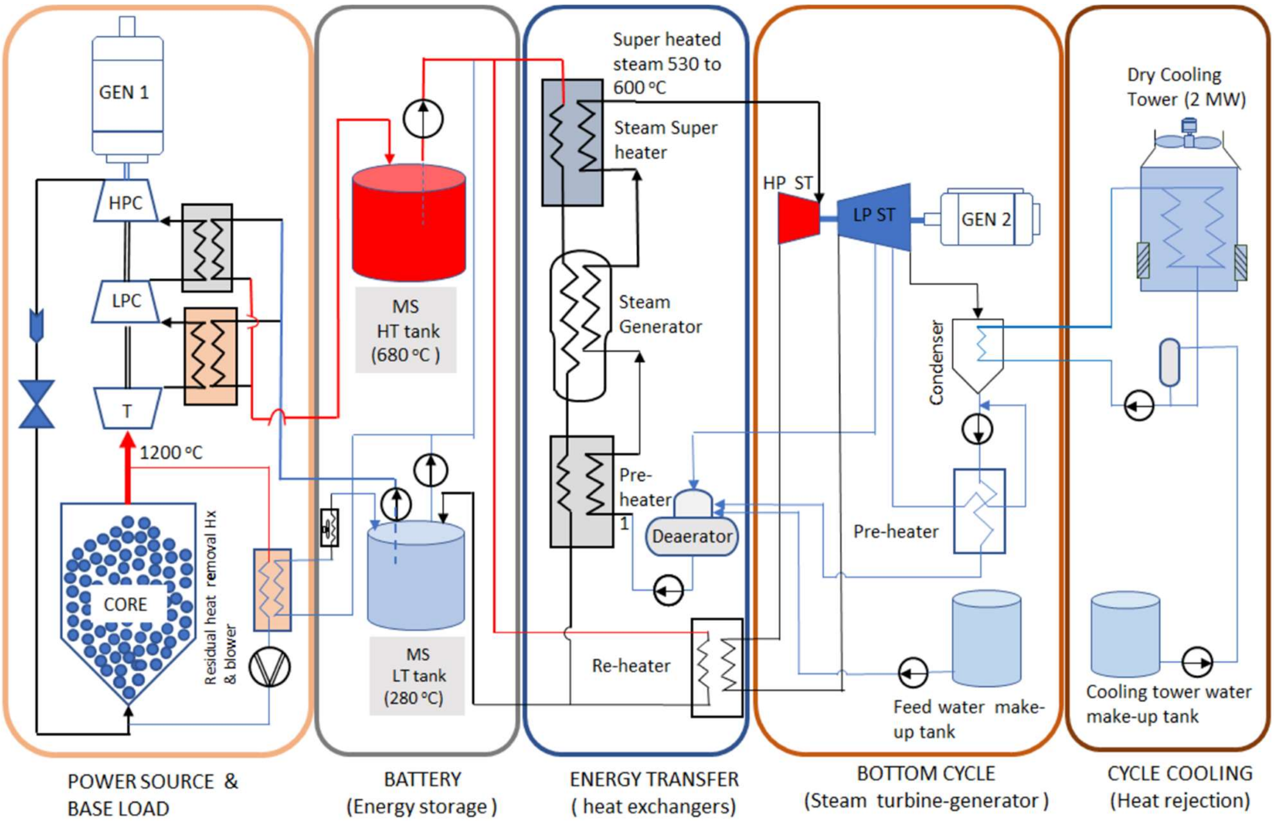
Given these potential advances available with current international technology, but not planned to be applied by other programs, significant performance improvements could be achieved over the performance envisaged for the original PBMR®.

In particular the use of carbon-composite materials to achieve higher operating temperatures (more than 1000°C), and the use of molten salts as heat storage, would result in higher thermal efficiencies and better flexibility. While all these technologies have been demonstrated at different scales in other industries, their

detailed application to High Temperature Reactor (HTR) designs would require an industrial-scale reactor plant to prove their suitability for a commercial reactor. In order to implement these advances in a demonstration plant, the AHTR-100 was conceptualised by Eskom Holdings SOC Ltd. – PBMR SOC Ltd. is wholly owned by Eskom Holdings SOC Ltd.
With an output of temperature of 1200°C the AHTR-100 is classified as a Very High Temperature Reactor (VHTR). Specific demonstrated nuclear technologies, such as the fuel design, will however remain the same as that of the PBMR®.

2. Target Application

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



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

3. Design Philosophy

As in the PBMR®, the AHTR-100 is a high-temperature helium-cooled, graphite moderated pebble bed reactor but with a once-through fuelling scheme. The design safety targets and features means that the reactor can be deployed close to the end user since there shall be no design base or credible beyond design base event that would need anyone living near the site boundary to take shelter or be evacuated. To achieve this goal there shall be no need for engineered or moving mechanical components to ensure this target is met while the exposure to plant personnel shall also be significantly lower than today's best international practice.

4. Main Design Features

(a) Power Conversion System

A Brayton power conversion with direct gas turbine is adopted as topping cycle. It is a closed cycle where the helium coolant is used to transport heat directly from the core to power turbine. The design incorporates a single shaft for the turbine, the compressors and the power generator. Heat exchangers (up to 3) to remove heat to the bottoming cycle is included.
From the reactor unit the hot helium enters directly to the turbine where energy is used to drive the shaft and therefore the electric generator and compressors. From the turbine the helium then passes consecutively through the primary side of the first high temperature heat exchanger, then the pre-cooler, the low pressure compressor, intercooler, high pressure compressor and then on to the high-pressure intercooler before re-entering the reactor unit.
The direct gas cycle is attractive since it promises the benefits of simplification, with the potential of lowering the capital and operational costs. Due to the high outlet gas temperature, one will also expect a substantial increase in the overall system efficiency.

This primary cycle will operate in baseload maximum capacity continually and provide 30% of the total plant electricity. This limits reactor operating transients to startup, full load operation, and shut down.
The heat exchangers contain molten salt coolant in the secondary side removing heat from the primary circuit and storing it for use in a load following Rankine cycle.

(b) Reactor Core and Fuel Characteristics

The core neutronic design results in a small cylindrical core with a diameter of 2.6 m. The effective cylindrical core height is 9.35 m. In steady state (equilibrium core) operation the fuel sphere powers (average 0.91 kW) and operational temperatures (1200ºC) fulfil the design criteria. The core contains ~110 250 fuel spheres or 'pebbles' with a packing fraction of 0.61. The fuelling scheme employed is the continuous on-line once-through method. Fresh fuel elements are added to the top of the reactor while used fuel pebbles are removed at the bottom to keep the reactor at full power.
The fuel kernel (UO_2) is coated by a first porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. About 13 330 coated particles and graphite matrix material are made into an inner fuel zone and surrounded by a 5 mm outer fuel free zone to make up the 6 cm diameter fuel sphere or pebble.

(c) Reactivity Control

Excess reactivity is limited by once through, continuous refuelling cycle while adequate passive heat removal ensures an inherent safe design with no event with significant fission product release being possible. Adequate reactivity control and long-term cold shutdown capability are provided by two separate and diverse control rod and small absorber sphere (SAS) systems while the overall negative reactivity temperature coefficient is negative over the total operational range.

(d) Fuel Handling System

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

(e) Reactor Pressure Vessel and Internals

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

5. Safety Features

The safety philosophy for modular HTRs has been described a number of times in the past 30 years and has been adopted with a few modifications by AHTR-100 in the same manner as with the PBMR®. Its basis is that an accident equivalent to severe core damage must be inherently impossible by limiting reactivity increases and ensuring that decay heat can be removed passively after a loss of coolant event. The AHTR, like the PBMR® has a simple design basis, with passive safety features that require no human intervention and that cannot be bypassed or rendered ineffective in any way. If a fault occurs during reactor operations, the system, at worst, will come to a standstill and merely dissipate heat through a heat pipe system on a decreasing curve without any core failure or release of radioactivity to the environment.

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal/ reactor Cooling Philosophy

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

(c) Containment Function

As with the PBMR®, the most important barriers to fission product release are the coatings of the fuel particles. A second barrier is provided by the Helium Pressure Boundary. A third barrier is the confinement building. The vented confinement is designed for very low leakage at low pressure, and to prevent damage to components important to safety, as well as to contain the build-up of higher activity gas in the delayed phase of a depressurisation event. Depending on the size of a pressure boundary break the system may be vented and then closed again with the released gas filtered as required.

6. Plant Safety and Operational Performances

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

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

7. Instrumentation and Control Systems

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

8. Design and Licensing Status

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

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

9. Fuel Cycle Approach

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

10. Waste Management and Disposal Plan

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

11. Development Milestones

2010	PBMR® Project in care and maintenance since 2010.
2016	AHTR-100 R&D activities commence.
2017	AHTR-100 Version 1 concept completed.
2018	R&D activities continue.
2019	R&D put on hold pending funding availablility.

PBMR®-400 (PBMR SOC Ltd, South Africa)

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Pebble Bed Modular Reactor SOC Ltd (PBMR®), South Africa
Reactor type	Modular high temperature gas cooled reactor
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	400 / 165
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	9 (direct cycle / no secondary steam)
Core Inlet/Outlet Coolant Temperature (°C)	500 / 900
Fuel type/assembly array	Pebble bed with coated particle fuel
Number of fuel assemblies	~452 000 in core
Fuel enrichment (%)	9.6% LEU or WPu
Reactivity control mechanism	Control rod insertion, negative temperature coefficient
Approach to safety systems	Active
Design life (years)	40
Plant footprint (m^2)	4200 (main structures only)
RPV height/diameter (m)	30 / 6.2 (inner)
RPV weight (metric ton)	1250 (with vessel head)
Seismic Design (SSE)	0.4g PGA for main power system design
Fuel cycle requirements / Approach	Uranium once through; spent fuel stored in tanks in the facility.
Distinguishing features	Inherent safety characteristics; no core melt; high efficiency; small number of safety systems
Design status	Preliminary design completed; test facilities demonstration; project stopped in 2010. in care and maintenance.

1. Introduction

The Pebble Bed Modular Reactor (PBMR®) is based on the evolutionary design of the German HTR-Module design. The PBMR® is designed in a modular fashion to allow for additional modules to be added in accordance with demand. In addition, the PBMR® can be used as base-load station or load-following station and can be configured to the size required by the community it serves.
Various reactor concepts have been under development since 1996. Most of these designs that are based on a direct Brayton cycle aims for higher efficiencies. The maximum achievable power levels for the reactor was increased in several design steps to achieve a set target for installed cost/kW that would be comparable to coal fired power when lifetime costs were evaluated. As a result, the design of the reactor core evolved from the original base of 200 MW(t) adopted from the HTR-Module design to reach 400 MW(t) with an annular core. Due to the world financial crisis in 2008 and short-term funding constraints, a reprioritization led to a decision to concentrate on the electricity and process heat market with a single reactor product and thus a decision was made to use an indirect steam cycle. The direct cycle design was archived with a view to further progress this design when conditions (financial and technology development in materials for the direct cycle) improve.

2. Target Application

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

3. Design Philosophy

The PBMR®-400 is a high-temperature helium-cooled, graphite moderated pebble bed reactor with a multi-pass fuelling scheme. The design objectives and features mean that the reactor can be deployed close to the end user since there shall be no design base or credible beyond design base event that would need anyone living near the site boundary to take shelter or be evacuated. To achieve this objective there shall be no need for engineered or moving mechanical components to ensure this objective is met while the exposure to plant personnel will be significantly lower than today's best international practice.

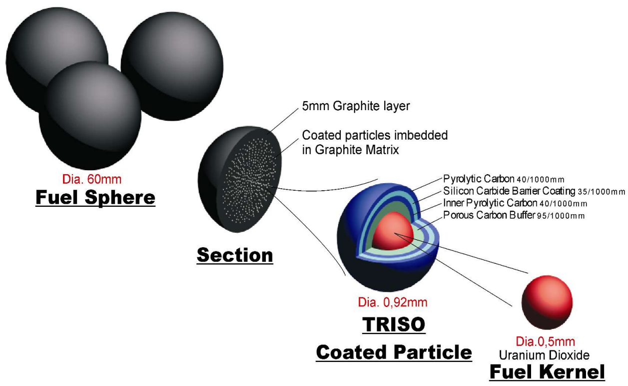
4. Main Design Features

(a) Power Conversion System

The Brayton cycle power conversion cycle with direct gas turbine is adopted. It is a closed cycle where the helium coolant is used to transport heat directly from the core to the power turbine. The design incorporates turbine, compressors and power generator in a single shaft. The flow of the Helium is depicted in the figure below. The direct gas cycle is attractive since it promises the benefits of simplification, with the potential of lowering the capital and operational costs. Due to the high outlet gas temperatures, one will also expect a substantial increase in the overall system efficiency.

(b) Reactor Core and Fuel Characteristics

The core neutronic design is an annular core with an outer diameter of 3.7 m and an inner diameter of 2 m shaped by the fixed central reflector. The effective cylindrical core height is 11 m. In steady state (equilibrium core) operation the fuel sphere power (maximum 2.7 kW per sphere) and operational temperatures (<1100^{0}C) fulfil the design criteria set. The core contains ~452 000 fuel spheres or 'pebbles' with a packing fraction of 0.61. The fuelling scheme employed is a continuous on-line multi-pass system. Fresh fuel elements are added to the top of the reactor while used fuel pebbles are removed at the bottom to keep the reactor at full power. On average fuel spheres are circulated six times through the reactor. This reduces power peaking and maximum fuel temperatures in normal operation and loss of coolant conditions.
The coated particle pebble fuel used is shown below. The fuel kernel (UO_2) is coated first by a porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. About 15 000 of these coated particles and graphite matrix material are made into an inner fuel zone and surrounded by a 5 mm outer fuel free zone to make up the 6 cm diameter fuel sphere or pebble.



(c) Reactivity Control

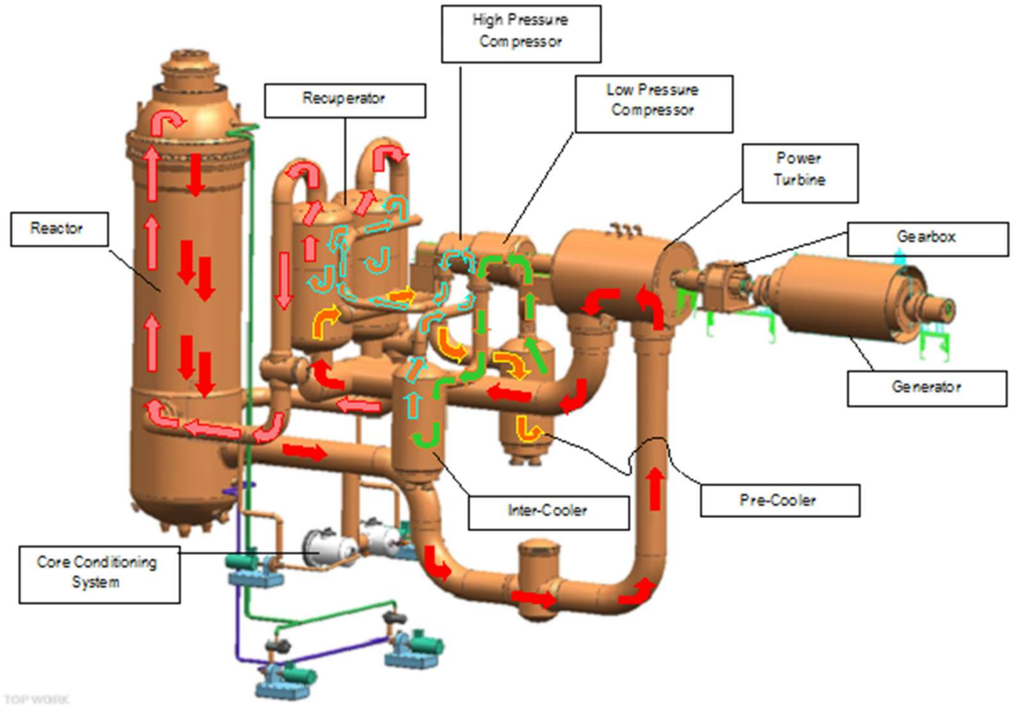
Excess reactivity is limited by continuous refuelling while adequate passive heat removal ensures an inherent safe design with no event with significant fission product release being possible. Adequate reactivity control and long-term cold shutdown capability is provided by two separate and diverse systems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

(d) Fuel Handling System

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

(e) Reactor Pressure Vessel and Internals

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



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

5. Safety Features

The safety philosophy for modular HTRs has been described a number of times in the past 30 years and has been adopted with a few modification by PBMR®. Its basis is that an accident equivalent to severe core damage must be inherently impossible by limiting reactivity increases and ensuring that decay heat can be removed passively after a loss of coolant event. The PBMR® has a simple design basis, with passive safety features that require no human intervention and that cannot be bypassed or rendered ineffective in any way. If a fault occurs during reactor operations, the system, at worst, will come to a standstill and merely dissipate heat on a decreasing curve without any core failure or release of significant radioactivity to the environment.

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal/ reactor Cooling Philosophy

The Reactor Cavity Cooling System provides a means to remove residual heat passively for a defined time, and indefinitely with the use of an active system after refilling the cooling system. For this to work, the Reactor Pressure Vessel and the core need to be long and slender. The belt region of the RPV is not insulated to allow heat radiation and convection to the water filled cavity cooler. In the event of the loss of active core cooling by the main circulation system, the cavity cooler and/or the building structural materials are able to limit the increase in fuel temperature in the most affected region of the core to below the allowable fuel temperature limit.

(c) Containment Function

The most important barriers to fission product release are the coatings of the fuel particles. A second barrier is provided by the Helium Pressure Boundary. A third barrier is the confinement building. The vented confinement is designed for very low leakage at low pressure, and to prevent damage to components important to safety, as well as to contain the build-up of higher activity gas in the delayed phase of a depressurisation event. Depending on the size of a pressure boundary break the system may be vented and then closed again with the released gas filtered as required.

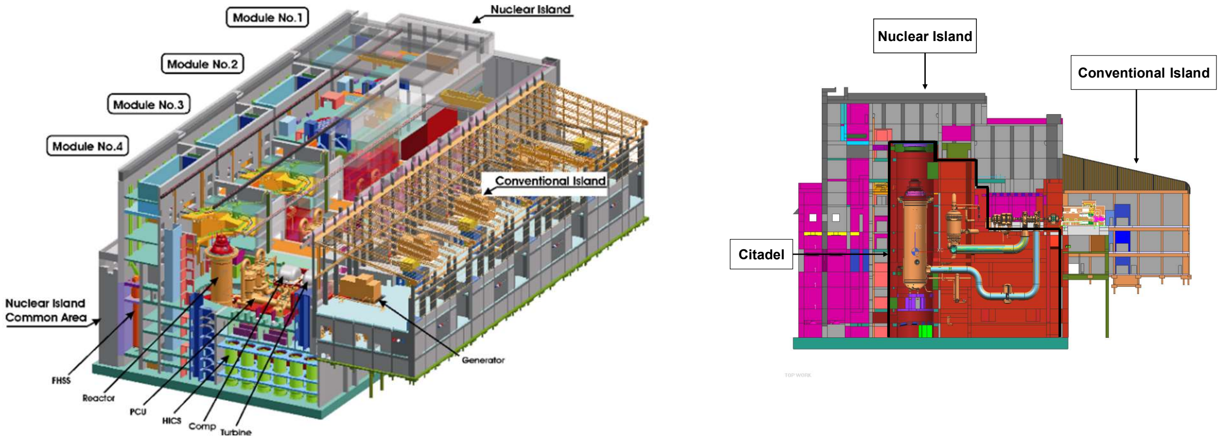
6. Plant Safety and Operational Performances

The PBMR®-400 safety does not rely on engineered systems that may fail but on the inherent design and the laws of physics. The risk metrics core damage frequency and large early release frequency are not applicable, but the same concepts are reflected in the immediate and delayed release category definitions. The design of the PBMR® represents a significant advancement in plant safety with an estimated delayed release category frequency of 1.0×10^{-5} per reactor year while maintaining an expected capacity factor of 95%.

7. Instrumentation and Control Systems

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

8. Plant Arrangement



PBMR®-400 building layout

9. Design and Licensing Status

The Reactor Plant Preliminary Design was completed and demonstration of key technologies were underway when the project was terminated in 2010.

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

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

12. Development Milestones

1993	The South African utility Eskom identifies PBMR as an option for new generating capacity.
1995	Start of the first pre-feasibility study.
1999	Design optimization: PBMR®-268 with dynamic central column.
2002	Design changed to PBMR®-400 with fixed central column.
2002	The Pebble Bed Micro Model (PBMM) demonstrated the operation of a closed, three shaft, pre- and inter-cooled Brayton cycle with a recuperator.
2004	Vertical layout of turbo machines changed to conventional single horizontal layout.
2006	Commissioning of Helium Test Facility for full scale system and component tests.
2006	Tests starts in the Heat Transfer Test Facility.
2007	Advanced fuel coater facility commissioned.
2009	Coated particles sent for irradiation testing at INL.; Alternative process heat markets and designs explored.
2010	Project closure.
2018	Project in care and maintenance.

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

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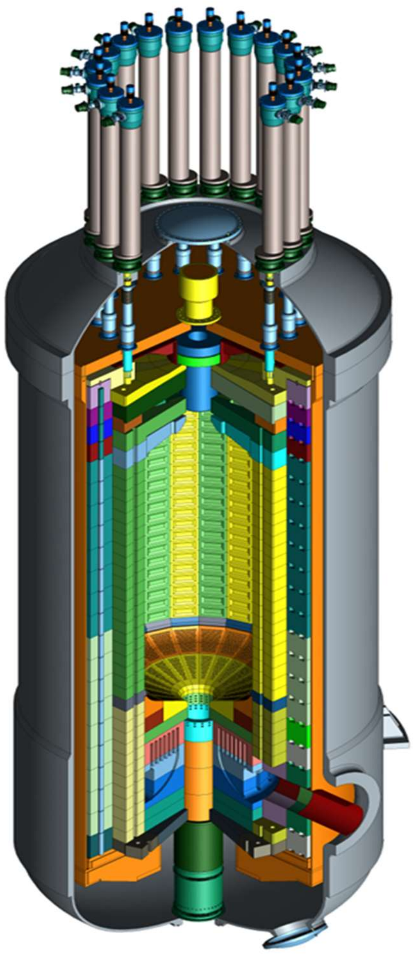


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	STL Nuclear (Pty) Ltd , South Africa
Reactor type	(HTGR) Pebble Bed
Coolant/moderator	Helium, graphite
Thermal/electrical capacity, MW(t)/MW(e)	100 / 35 single module plant
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	4 / 16
Core inlet/outlet coolant temperature (ºC)	250 / 750
Fuel type/assembly array	TRISO particles in pebbles; LEU/Th
Number of fuel assemblies in the core	~150 000; around 125 to 150 pebbles/day throughput
Fuel enrichment (%)	10
Refuelling cycle (months)	Online fuel loading
Core Discharge Burnup (GWd/ton)	80-90
Reactivity control mechanism	Control Rods in the reflector
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	5 000 (buildings only)
RPV height/diameter (m)	15.7 / 5.6 (flange outer diameter)
RPV weight (metric ton)	155
Seismic design (SSE)	0.3 g (generic site), 0.5 g under consideration)
Fuel cycle requirements/approach	Various options (see below)
Distinguishing features	No core meltdown, no active engineered safety systems
Design status	Basic design

1. Introduction

The HTMR100 (High Temperature Modular Reactor) pebble bed reactor is a high temperature gas cooled reactor, graphite moderated and cooled by forced helium flow. The existing design of the module is to produce high quality steam which is coupled to a steam-turbine/generator system to produce 35 MW(e). The steam can be used in a wide range of cogeneration applications. The reactor is also suitable to provide direct high temperature energy for process heat. The design of the reactor is based on proven technology and therefore no new basic technology development is needed. The size of the reactor and the fuel cycle were chosen to simplify the design and operation of the module. The approach to small intrinsic safe modular units ensures continuous production, easy road transportability, skid mounted sub systems, wider range of manufactures, fast construction and an enhanced licensing process.

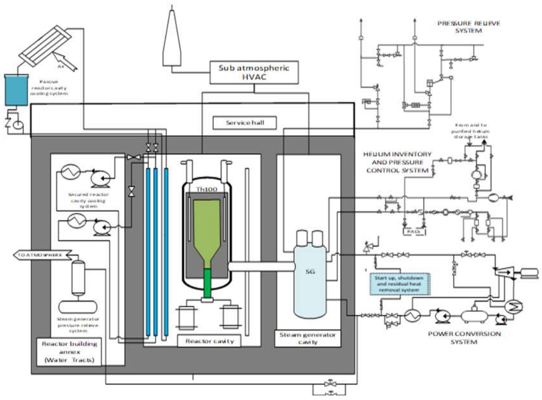
2. Target Application

The HTMR100 can supply electric power to any distribution grid and to standalone or isolated electricity users. It can be deployed as single modules or multi-module plants as well as for medium temperature process heat applications (later also upgradable to very high temperature). The HTMR100 is a perfect fit for clients who want to progressively extend their generating capability. The unique safety characteristics make it possible to introduce and construct these plants to non-nuclear countries. First-world countries that want to utilize their stock of Plutonium for peaceful applications are also markets for HTMR100 reactors.

3. Design Philosophy

The reactor has good load following characteristics which is needed for stand-alone (not grid coupled) applications. The 'Once Through Then Out' (OTTO) fuelling scheme leads to a simple and cost-effective fuel management system. The relatively low primary loop pressure requires a thinner walled pressure vessel and thus an easier manufacturing process, resulting in a wider range of vessel manufacturers. The HTMR100 plant design caters for different site and client requirements. It allows flexibility in protection against external events and flexibility in multi module configuration and power capacity.

4. Main Design Features



Reactor core and power conversion layout

(a) Power Conversion

The flow through the core is from top to bottom where the heated gas is collected in a hot plenum. From the plenum the hot gas flows through a connecting pipe to the steam generator. The power conversion system uses a helical coil steam-generator unit supplying super-heated steam to the turbine. The main system will be supplied as four skid mounted units namely the condenser, turbine, gearbox and electric generator. The turbine can be used in a back-pressure configuration or intermediate temperature steam can be taken off for process heat applications.

(b) Reactor Core

The reactor unit consists of a steel pressure vessel, a steel core barrel, graphite reflector blocks, neutron absorber rods, rod guide tubes, drive mechanisms and in-vessel instrumentation. The vessel is designed for 4 MPa pressure. The graphite structure allows for differential expansion and volumetric changes due to temperature and neutron fluence induced distortion. This is done to keep the stresses low and minimize primary fluid bypass and leaks. The side, top and bottom reflector material is nuclear grade graphite.

(c) Fuel Characteristics

The fuel elements (FE) for the HTMR100 are 60 mm diameter spheres consisting of a spherical fuel zone of approximately 50 mm diameter, in which the TRISO-coated particles are randomly distributed in the graphitic matrix material. A fuel-free shell of graphite matrix of about 5 mm in thickness is then moulded to the fuel zone. The fuel kernel and coatings serve as a fission product barrier in normal and accident operating conditions. There are various types of fuel that will be used in the HTMR100 reactor (see below for details). A Fuel Qualification and Test programme will be conducted on the fuel prior to loading of the reactor. The HTMR100 operates on a much longer burn-up fuel cycle compared to conventional reactors. The non-proliferation characteristic of the OTTO cycle is the extended time the pebbles reside inside the core, making it more difficult to divert partially burnt fuel.

(d) Fuel Handling System

A six-month supply of fresh fuel is kept in the fresh fuel storage facility. New spherical fuel elements (fresh fuel) are loaded by the fuel loading machine into a charge lock. The charge lock is purged, filled with clean helium and pressurised to system pressure, before it is opened, and fuel is gravity fed into the core cavity. The charge lock has a physical capacity for approximately one full-power day's fuel sphere inventory.

(e) Reactivity Control

Eighteen neutron absorber rods are provided in graphite sleeves inside the graphite side reflector blocks. The absorber rods can be operated independently as a group or as sub-groups, as required by the reactor operating control system. A control rod consists of several rod absorber material segments, pinned together to form articulating joints. The segments consist of sintered B4C absorber material, sandwiched between an inner and an outer tube segment. The inner tube segment allows cooling helium gas to flow form the top down in the circular channels. Each rod is equipped with a position indicator which measures the position of the rod over its entire positioning range and with position indicators for the upper and lower limit positions.

(f) Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel (RPV) is constructed in compliance with the ASME III subsection NB code. It comprises two main components reactor of vessel body and vessel head which is bolted to vessel body. The reactor vessel body consists of several forged ring-components circumferentially welded together.

The core structures consist of the metallic parts and the graphite structures. The function of these internal structures is to provide stable core geometry, neutron reflection, cold and hot gas channelling, fuel element flow, shielding, thermal insulation and support of the control and shutdown systems guide tubes and the neutron source. The functional design of the structural core internals is such that they can withstand the steady state and transient loadings during normal operation, anticipated operational occurrences and design basis accidents.

The shape and structure of the inner side reflector wall and the 30° angled core bottom permit uniform fuel element flow. The loads borne by the ceramic internals are transferred to the steel core barrel and then to the reactor pressure vessel through metallic components such as the lower support structure and the core barrel axial and radial supports.

All areas of the core internals are designed for the service life of the reactor. Access for ceramic structure inspections can be done through the fuel loading channel and the reflector rod holes.

(g) Reactor Coolant System

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

In principle the plant is designed to perform its safety functions without reliance on the automated plant control system, or the operator. The engineered safety system of the plant has no engineered safety systems in terms of active human or machine intervention to assure nuclear safety. Provision for beyond design basis conditions is made. Beyond design basis scenarios include the non-functioning/non-insertion of all active control and shutdown systems. The reactor core characteristics e.g. small excess reactivity and strong negative reactivity coefficient with temperature will shut down the reactor and maintain a condition where no damage to the fuel, core structures and reactor vessel occurs. Excessive reactivity increases during water or water vapor ingress (increasing moderation) is prevented by designing the reactor for limited heavy metal content of the fuel.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

The Reactor Cavity Cooling System (RCCS) removes heat radiated from the reactor towards the reactor cavity walls. It consists of welded membrane tubes arranged side-by-side on the inside of in the reactor cavity wall. Water is circulated through the tubes to form a cold wall. The RCCS is a passive system and consists of three independent cooling trains and is designed for all postulated design basis conditions.

(c) Spent Fuel Cooling Safety Approach / System

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

(d) Containment System

The primary fission product barrier is the SiC layer of the TRISO coated fuel particles, which keeps the fission products contained under all postulated events, even if the second barrier (the primary pressure vessel system) and the third barrier (the building filter system) fails.

(e) Chemical Control

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

6. Plant Safety and Operational Performances

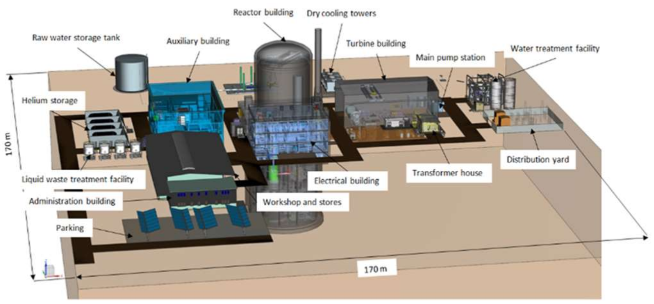
The central consideration is the demand for high availability of process steam supply and/or electricity generation. To reduce or minimize the NSSS daily or weekly load changes of the reactor, the preference is to change the ratio between steam supply and electricity supply. Excess steam and/or electricity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

7. Instrumentation and Control System

The Automation System (ATS) comprises that group of safety and non-safety C&I systems that provide automated protection, control, monitoring and human-system interfaces. The three specific systems in the HTMR100 system structure define control and instrumentation are plant control, data and instrumentation system, equipment/investment protection system and protection system.

8. Plant Layout Arrangement

The reactor building contains the safety equipment that provides the necessary functions for the safe shutdown of the reactor under all design basis conditions. The reactor building is partially submerged below ground level such that the reactor and steam generator cavities are completely protected against postulated external threats. The depth can be further adapted to suit the geological conditions of the specific site to provide for the necessary level of seismic protection.



Plant Layout

The reactor building, electrical building and auxiliary buildings are connected by means of underground tunnels, providing protection for interlinked services and, it also ensures that spent fuel is never brought above ground level. Provision is made for the storage of all spent fuel produced during the operating life of the plant. The reactor building is seismically designed to withstand a design basis earthquake (DBE) and together with the spent fuel storage bunker, is the only safety related building structure of the HTMR100.

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

The Electric Building houses the main control and computer rooms, primary and secondary plant security alarms rooms and provides the primary access facilities for the nuclear island and the energy conversion area. This centre also provides space for activities associated with plant administration and security services. The plant control, data, and instrumentation system control/display panels and computers are housed in the control room.

9. Testing Conducted for Design Verification and Validation

Under design.

10. Design and Licensing Status

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

11. Fuel Cycle Approach

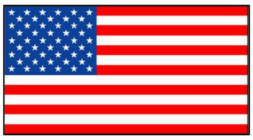
The reactor design can accommodate various fuel types with different fuel cycles, including mixtures of thorium and plutonium or thorium and uranium. Studies have shown the reactor can utilise: (i) 10% LEU (7-10g HM/sphere); (ii) Th/LEU mix of 50% LEU (20% enriched) and 50% Th (10-12g HM/sphere); (iii)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

12. Waste Management and Disposal Plan

The HTMR100 fuel elements can be stored and disposed of a fuel sphere but available technology needs to be assessed for volume reduction. Disposal of spent spherical fuel elements from the HTMR100 is executed in the following sequence: (i) Direct transfer of spent fuel elements into a flask inside the cast iron high energy spent fuel casks (Hi-cask); (ii) Immediately after filling the H-cask they are sealed and transferred to the spent fuel cool-down facility on site; (iii) Once cooled down, the flask filled with fuel is transferred from the H—cask to a low energy spent fuel concrete cask (Low-Cask); iv) The Lo-Cask is transported to the low energy on-site interim storage facility; v) For offsite transport the flask is transferred to a shipping/transport cask for shipping to an ultimate repository. Approximately 55 000 fuel elements will be discharged for on full-power year of operation and only one or two flasks containing physically damaged fuel spheres, singled out by the fuel unloading machine (failed fuel separator), may be required in the lifetime of the core. As in normal and accident conditions the coated particles maintain its excellent fission product retention capabilities and fission products are almost entirely retained within the fuel element kernels. Also, the release of these nuclides into the cask or flask atmosphere from the number of fuel element particles with defective SiC coatings is very low.

13. Development Milestones

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



Energy Multiplier Module (EM^2) (General Atomics, United States of America)

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

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	General Atomics (GA), United States of America
Reactor type	Modular high temperature gas-cooled fast reactor
Coolant/moderator	Helium / none
Thermal/electrical capacity, MW(t)/MW(e)	500 / 265
Primary circulation	Forced cooling
NSSS operating pressure (primary/secondary), MPa	13.3
Core inlet/outlet coolant temperature (°C)	550 / 850
Fuel type/assembly array	Uranium carbide pellet/ hexagonal
Number of fuel assemblies in the core	85
Fuel enrichment (%)	14.5
Refueling cycle (months)	360
Core discharge burnup (GWd/ton)	130
Reactivity control mechanism	Control rod drive mechanism
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	90 000 (four modules)
RPV height/diameter (m)	12.5 / 4.6
RPV weight (metric ton)	700
Seismic design (SSE)	0.3 g
Fuel cycle requirements/approach	Open fuel cycle
Distinguishing features	Convert-and-burn/Silicon carbide composite cladding/ Fission gas collection
Design status	Conceptual design

1. Introduction

Energy Multiplier Module (EM^2) is a helium-cooled fast reactor with a core outlet temperature of 850°C. It is designed as a modular, grid-capable power source with a net unit output of 265 MW(e). The reactor converts fertile isotopes to fissile and burns them in situ over a 30-year core life. EM^2 employs a direct closed-cycle gas turbine power conversion unit (PCU) with a Rankine bottoming cycle for 53% net power conversion efficiency assuming evaporative cooling. EM^2 is multi-fuel capable, but the reference design uses low-enriched uranium (LEU) with depleted uranium (DU) carbide fuel material with silicon carbide (SiC) composite cladding material (i.e., SiGA®).

2. Target Application

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

3. Design Philosophy

EM^2 design philosophy is to develop a new nuclear plant to address the following goals for enhancing the likelihood of commercial success: i) economic parity with fossil fuel generation in the US, ii) improved siting flexibility via dry cooling and site accessibility, iii) passive safety for sustained protection during long-term station blackout and other severe accidents, and iv) improved fuel resource utilization, reduced nuclear waste, and high proliferation-resistance.

4. Main Design Features

The EM^2 core was specifically designed to extend the fuel burnup to maximize the fuel utilization with a reasonable amount of initial uranium loading. From this perspective, a fast neutron spectrum was chosen. For the thermal efficiency of the plant, high temperature operation was chosen. These design choices require use of high temperature material for the fuel and core structure. To accommodate high fuel burnup, the fission gases are removed from the fuel and stored in a collection system, which maintains the pressure in the fuel slightly lower than the primary system pressure.

(a) Power Conversion

The power conversion is based on a combined cycle with a direct helium Brayton cycle and a Rankine bottoming cycle. The helium Brayton cycle is provided by located in the PCU, while the Rankine cycle is implemented in the facility outside the reactor building. The Brayton cycle incorporates the turbo-compressor (T/C) and generator, which are mounted on an in-line vertical shaft suspended by active magnetic bearings. The cycle also incorporates two heat exchangers (HX), a recuperator and a precooler. The generator uses a permanent magnet (PM) rotor to eliminate losses associated with a wound rotor and exciter.

(b) Reactor Core

The core is supported by the support floor through the core barrel, which is attached to the vessel below the cross-duct. The upper carbon-composite (C-C) heat shield protects the top head elements from the hot helium. The vessel is internally insulated with silica/alumina fibrous insulation retained with a C-C cover plate. To achieve high fuel utilization, the EM^2 core utilizes the "convert and burn" concept, in which the core is divided into fissile and fertile sections. The fissile section is the "critical" section at beginning-of-life. It contains ~14.5% LEU to sustain the chain reaction and provide excess neutrons to convert DU fertile to fissile material. The average enrichment of the total active core is 7.7%. The reflector consists of an inner section of zirconium-alloy blocks and an outer section of graphite blocks. The basic building block of the EM^2 fuel system is the hexagonal assembly, of which there are 85 in the core. Eighty-one assemblies are joined into 27 tri-bundles and four remain as individual assemblies.

(c) Fuel Characteristics

Uranium carbide (UC) is used to meet the high uranium loading requirement; UC has a very high thermal conductivity, is compatible with the SiGA cladding, and has a suitably high melting point. Each annular fuel pellet is a sintered "sphere-pac" with a specified interstitial and internal distributed porosity to allow for faster migration of volatile fission products. SiC composite is especially attractive due to its stability under long-term irradiation as demonstrated in a multi-year irradiation campaign. Both the fuel and cladding materials meet design criteria temperature limits for both normal operations and accident conditions.

(d) Fuel Handling System

The core is accessed by a refueling machine from the maintenance hall floor. An articulated arm extends through the containment and reactor vessel penetration to select and withdraw a tri-bundle assembly and load it into a sealed, air-cooled storage container. The container is moved to the end of the maintenance hall, where it is lowered into the fuel storage facility. This facility has the capacity for 60 years of operation. The spent fuel is cooled within the sealed containers by passive natural convection of air. No water or active cooling is required.

(e) Reactivity Control

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

(f) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) is constructed from welded ring-forgings or rolled plate. The vessel contains large penetrations for the two cross-vessels and a flanged hatch. The top head has the penetration for refueling access and control elements. The RPV has no external insulation but is internally insulated. This insulation maintains the vessel well below 371°C during normal operation and design basis accidents, which allows the use of SA-533 grade B material.

(g) Reactor Coolant System

The reactor coolant system (RCS) encloses the reactor and PCU and includes all subsystems that contain, transfer, regulate, purify, monitor, and store primary coolant helium. The RCS provides controlled pressurized helium for transport of heat from the reactor to the PCU during normal operation and from the reactor to the direct reactor auxiliary cooling system (DRACS) during shutdown conditions when the PCU is not available. It also provides a Safety Class I barrier against the release of radioactivity to the containment during normal and abnormal conditions.

5. Safety Features

The EM^2 safety design uses a defense-in-depth approach, which employs three successive, encompassing barriers against the release of radionuclides. Each barrier relies ultimately on passive means for protection of its integrity for normal and abnormal operation. The first barrier is the SiGA cladding. The second barrier is the primary vessel system, which encompasses the reactor, PCU, and DRACS. The third barrier is the free-standing, below-grade containment. Because the fuel is vented, the fission gas collection system (FGCS) is an extension of the first barrier.

(a) Engineered Safety System Approach and Configuration

The engineered safety features are designed to detect incipient conditions that pose a threat to the fission product barriers and initiate protective actions including reactor shutdown, sustained heat removal from the core, and protected integrity of the primary pressure boundary and/or containment. EM^2 incorporates several systems classified as engineered safety features, emergency systems or systems with safety significance such as DRACS, containment system, reactivity control system, reactor protection system, Class 1E Alternating Current (AC) and Direct Current (DC) power systems, and fission product vent system (FPVS).

(b) Decay Heat Removal System/Reactor Cooling Philosophy

The DRACS safely removes core decay heat during normal shutdown and accident conditions when the PCU is not available. It provides controlled core heat removal during an anticipated transient without scram (ATWS) and active core heat removal during special maintenance conditions characterized by need for low temperatures and/or low helium pressures.

(c) Spent Fuel Cooling Safety Approach/System

The spent fuel storage facility (SFSF) accommodates 60 years of spent fuel storage. The SFSF provides adequate passive dry cooling of the spent fuel, protection from external threats, and monitoring for spent fuel storage canister (SFSC) leakage. The SFSF is located below grade and is protected by an extension of the protective shield that covers the reactor building maintenance hall. The SFSF roof is aligned with the maintenance floor to allow easy transport of the SFSCs to the SFSF. The SFSF has redundant, elevated air intakes and outlets for cooling.

(d) Containment System

The primary heat transport system (PHTS) is enclosed by a sealed, below-grade containment, which is divided into three connected chambers with structural ligaments around the reactor chamber that also serve as shielding to all access to the two side chambers. The containment is hermetically sealed with an inert (argon) atmosphere at ~20 psig (0.14 MPa). The peak pressure rating is 90 psig (0.62 MPa). The design leakage rate is less than 0.2% per day.

(e) Chemical Control

Helium coolant is maintained at a high level of purity by the helium purification system (HPS). The HPS maintains the total oxidant (O_2, CO, H_2O, CO_2) concentration in the helium below 0.35 ppmv and N_2 and H_2 concentrations below 10 ppmv; at these levels of purity, chemical interactions between coolant contaminants and the fuel are not life limiting. The high temperature adsorber (HTA) is shared with the FPVS. The HTA removes halides and metals. The effluent from the HTA is transported to the reactor auxiliary building, where the remaining oxidants and noble gases are removed.

6. Plant Safety and Operational Performances

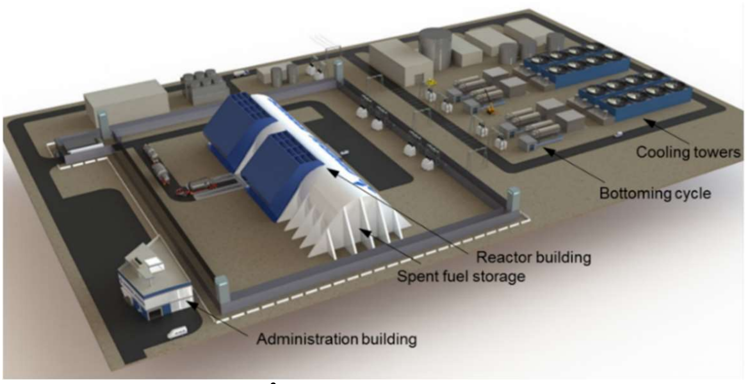
Unlike large power reactors, each EM^2 module utilizes a unique, non-synchronous, variable speed generator with a frequency inverter and generator load commutator to follow load demand. In the automatic load following mode, the generator speed is set by modulating the generator speed. A field-oriented control algorithm in the frequency converter controls the generator torque that decreases or increases the generator speed, which in turn determines helium flow. The ability to control the turbo-generator speed through field control replaces traditional mechanical control elements with digital electronic control. An advantage of the variable speed control is that it maintains primary system structures at near constant temperature with load so that rapid load following should be possible.

7. Instrumentation and Control Systems

EM^2 deploys advanced sensors for high operating temperature condition such as solid-state and SiC neutron flux monitors. The plant control including startup, operation, and shutdown is conducted through integrated control system actions, which regulate reactor power and turbomachinery to respond to the plant transients. Plant control functions are performed by the power control and the process component control systems. The power control system includes control rod drive mechanism (CRDM) and reactor coolant flow. The process component control system includes a non-synchronous, variable speed generator and a frequency inverter.

8. Plant Layout Arrangement

The baseline EM^2 plant is composed of four 265 MW(e) modules for a combine net power of 1060 MW(e) to a utility grid for evaporative cooling and 960 MW(e) net for dry cooling. Each module consists of a complete powertrain from reactor to heat rejection such that the modules can be built sequentially and operated independently. The plant layout covers 9.3 hectares (23 acres), not including the switchyard. The maintenance hall floor is at grade level, and the roof serves as a protective shield structure. The maintenance hall serves all four reactors.



EM^2 plant layout on 9 hectares of land

9. Testing Conducted for Design Verification and Validation

The design verification and validation plan will be developed. The experimental measurements have been partially conducted for i) UC spherical kernel irradiation tests in High Flux Isotope Reactor (HFIR) Cycle 490 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 (Zr_3Si_2) reflector material in 2016.

10. Design and Licensing Status

Interaction with the Nuclear Regulatory Commission (NRC) was initiated for the fuel qualification plan. The licensing process of the EM^2 prototype plant was not launched. It will be licensed using the two-step 10 CFR Part 50, and follow-on commercial plants will be licensed under the one-step 10 CFR Part 52 process. The technology-inclusive, risk-informed, performance-based licensing framework developed by the Licensing Modernization Project will be used to develop the NRC license applications.

11. Fuel Cycle Approach

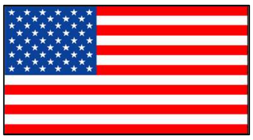
The EM^2 open fuel cycle with LEU/DU vented fuels exceeds 30 years without refueling or shuffling, leading to a reduced cost of power, low proliferation risk, high fuel utilization, and low mass of waste streams. The core is capable of burning used light water reactor (LWR), plutonium, and thorium fuels.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

12. Waste Management and Disposal Plan

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

13. Development Milestones

Phase I	Conceptual design of reactor core and power plant: high risk research and development
Phase II	Demonstration unit development: demonstration of major components; fuel design and qualification; high temperature material development
Phase III	Prototype unit development: demonstration of plant operation; licensing for commercial unit



Fast Modular Reactor (FMR) (General Atomics, United States of America)

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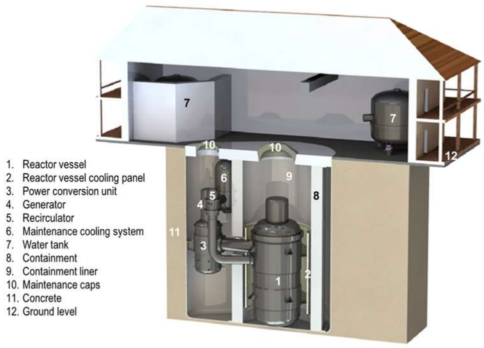

Reactor building and major systems and components arrangement designed with defense in depth

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
Thermal/electrical capacity, MW(t)/MW(e)	100 / 50
Primary circulation	Forced cooling
NSSS operating pressure (primary/secondary), MPa	7
Core inlet/outlet coolant temperature (˚C)	509 / 800
Fuel type/assembly array	Uranium dioxide pellet/ hexagonal
Number of fuel assemblies in the core	198
Fuel enrichment (%)	19.75
Refueling cycle (months)	96
Core discharge burnup (GWd/ton)	100
Reactivity control mechanism	Control rod drive mechanism
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	38 000 (four modules)
RPV height/diameter (m)	12 / 5
RPV weight (metric ton)	700
Seismic design (SSE)	0.3 g
Fuel cycle requirements/approach	Open fuel cycle
Distinguishing features	Silicon carbide composite cladding
Design status	Conceptual design

1. Introduction

General Atomics (GA) is developing a 50-megawatt electric (MWe) fast modular reactor (FMR) that provides safe, carbon free electricity, capable of incremental capacity additions. A modular design allows it to be factory built and assembled on-site to keep the cost of capital low, while the use of dry cooling facilitates siting to complement renewables in nearly any location. The FMR is being developed for a demonstration before 2030 and deployment by the mid-2030s.

2. Target Application

The FMR is being developed for the electricity generation with fast response to grid operators and a potential for high temperature use.

3. Design Philosophy

The FMR is designed to provide flexible and dispatchable carbon-free power for the 2035 US energy market. The design will develop and verify that simplified characteristics, such as the inert helium gas coolant, pellet-loaded fuel rod, installations free of heat sink requirements, and small and passive heat removal systems, will result in a safe, maintainable, cost-effective, distributed, nuclear energy-generating station.

4. Main Design Features

The FMR is a helium-cooled fast reactor with a core outlet temperature of 800°C and a grid-capable power source with a net electrical output of approximately 50 MW.

(a) Power Conversion

The direct helium Brayton cycle enables fast grid response and high overall efficiency during normal operation. The high-speed generator, along with a permanent magnet rotor and the helium turbo-compressor, has a small rotational inertia. The high-efficiency, solid-state converter accepts a wide range of input frequency and voltage and outputs the line frequency and voltage with equivalent power.

(b) Reactor Core

The reactor has a compact annular core surrounded by advanced fast neutron reflector material, i.e., zirconium silicide (Zr_3Si_2), graphite, and steel vessel. Low core power density and use of uranium dioxide (UO_2) fuel and silicon carbide (SiC) composite matrix ceramic cladding secure the safety margin of the fuel. A three-batch refueling scheme is used to enhance higher fuel utilization and reduce the requirement of excess reactivity control.

(c) Fuel Characteristics

The FMR uses conventional UO_2 as the fuel material. The average discharge burnup of ~100 gigawatt days per ton of uranium (GWd/tU) is achieved by use of high assay low enriched uranium (HALEU) and irradiation-resistant SiC composite cladding that is derisked in the current Accident Tolerant Fuel (ATF) program.

(d) Fuel Handling System

All fuel handling activities are carried out from the floor of the maintenance hall. All access to and removal of the fuel will be by remote handling. The refueling equipment includes a dry spent fuel storage canister (SFSC) and a fuel-handling machine.

(e) Reactivity Control

The excess reactivity is controlled by control rods made of boron carbide (B_4C) encapsulated in SiC sheath. Each control rod is fail-safe, by design. The shutdown rod is also made of B_4C which is fully withdrawn from the core during normal operation.

(f) Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) contains the nuclear fuel and reactor internals. The RPV is made of 316 Stainless Steel. The reactor internals include the high-temperature upper support structure, the lower core support structure, and the in-core instrumentation support structure.

(g) Reactor Coolant System

The reactor coolant system (RCS) removes heat from the reactor core and transfers it to the power conversion unit (PCU) for electricity generation. The RCS provides controlled pressurized helium for transport of heat from the reactor to the PCU during normal operation and from the reactor to the maintenance cooling system (MCS) during shutdown conditions when the PCU is not available.

5. Safety Features

The FMR uses a non-hazardous helium coolant, a chemically inert gas that is non-explosive, non-corrosive, and not activated. The reactor has an annular core surrounded by advanced reflector material and a steel vessel. Low core power density and use of high-temperature SiC composite cladding secure the safety margin in the highly unlikely event of an accident. Under such accident condition, the reactor can be cooled over the long-term by a gravity-driven reactor vessel cooling system (RVCS).

(a) Engineered Safety System Approach and Configuration

The engineered safety features (ESF) are implemented to mitigate the consequences of postulated accidents. The ESF include reactivity control system, reactor protection system, maintenance cooling system, reactor vessel cooling system, and containment.

(b) Decay Heat Removal System/Reactor Cooling Philosophy

The normal decay heat removal is conducted by the PCU. The MCS is an active system, including a recirculator and a heat exchanger. The RVCS provides a redundant, passive means of transporting core residual heat when neither the PCU nor the MCS is available, preventing the reactor vessel from exceeding design temperature limits. The RVCS design includes an array of cooling pipes, forming a "cooling panel" placed around the reactor vessel. Heat is removed from the cooling panel by natural circulation of water through the cooling panel.

(c) Spent Fuel Cooling Safety Approach/System

The spent fuel storage facility (SFSF) stores 60 years of spent fuel. The SFSF provides adequate passive dry cooling of the spent fuel, protection from external threats, and monitoring for SFSC leakage. The SFSF is located below grade and is protected by an extension of the protective shield that covers the reactor building maintenance hall. The SFSF has redundant, elevated air intakes, and outlets for cooling.

(d) Containment System

The containment system uses a below-grade, free-standing, steel containment with concrete backing to enclose the primary coolant system. The containment is safety-related and is classified as Seismic Category I. The containment will be supported from a seismic isolation platform and be hermetically sealed during normal operation.

(e) Chemical Control

The chemical and volume control system (CVS) provides reactor coolant purification, reactor coolant inventory control and makeup, chemical control, and oxygen control. The CVS performs these functions during normal modes of operation including power generation and shutdown.

6. Plant Safety and Operational Performances

The FMR combines the safety features of the high-temperature gas reactor (HTGR) with the high fuel utilization feature of the GFR, while taking advantage of UO_2 fuel and SiC composite cladding developments under the ATF program. The high-temperature, radiation resistance of SiC composite cladding enables the passive safety of the FMR system driven by the RVCS, which does not require bulk heat reservoir or injection of emergency cooling. The direct helium Brayton cycle enables fast grid response for load following, and high overall efficiency during normal operation. The automatic control of the reactor power and turbomachinery keep the reactor at a constant temperature that mitigates thermal cycle fatigue associated with most load-following reactors. Millisecond response to the grid is provided by the combination of the integrated direct Brayton cycle reactor and turbomachinery, the permanent magnet (PM) generator, and the frequency converter. The highly automated, hierarchical control system reduces the burden on the plant operator and allows for efficient grid operator dispatch.

7. Instrumentation and Control Systems

The instrumentation and control (I&C) safety system is designed to achieve an innovative, architecture-like, field programmable gate array (FPGA) technology that promotes simplicity of design and enhances verifiability of the reactor protection system (RPS). The non-safety system is a distributed control system (DCS) that is a highly automated, hierarchical control system that requires minimum operator actions to perform predefined functions (i.e., startup, normal power-range load-following, turbine warm-up, reactor shutdown, etc.) as well as during certain off-normal and abnormal operations. The DCS utilizes digital technology, such as microprocessors, FPGAs, and application-specific integrated circuits (ASICs), to take advantage of its high reliability and inherent capabilities regarding continuous self-diagnostics, on-line monitoring, high reliability, real-time data analysis and trending, etc. These elements build a nearly autonomous system that will enhance a significant reduction in operation and maintenance (O&M) cost.

8. Plant Layout Arrangement

The FMR prototype plant consists of a single reactor module sited below grade. The plant includes a reactor and support facilities within a minimized footprint of 38,000 m^2. Dry cooling structures are located adjacent to the reactor. Support equipment for helium and water processing, cooling, fire suppression, and backup power systems are all contained within the auxiliary building. All water, helium, and fire suppression systems are protected in an underground fluid transport network structure connected to the reactor.

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Compact GA 50 MWe FMR Plant

9. Testing Conducted for Design Verification and Validation

The FMR design verification and validation plan will be developed. The FMR fuel rod integrity will be experimentally verified in an advanced test reactor (ATR) and transient reactor test (TREAT) facility for high-burnup irradiation and power ramping, respectively. The high-burnup irradiation tests use the fission accelerated steady-state testing (FAST) approach to accelerate the irradiation, while the transient tests characterize the fuel cladding mechanical behavior under the initial power ramp. This test will use a modified version of the dry in-pile fracture test (DRIFT) capsule.

10. Design and Licensing Status

GA is currently executing pre-application licensing engagement with the Nuclear Regulatory Commission (NRC), including the FMR principal design criteria, quality assurance (QA) program plan, pre-probabilistic risk assessment (PRA), licensing basis event (LBE) selection, and safety classification, following regulatory guide (RG) 1.23259, RG 1.23360, and Nuclear Energy Institute (NEI) 18-0461. In the longer term, the licensing approach for the demonstration and commercialization of the FMR concept is to construct and operate a demonstration unit through two-step 10 CFR Part 50 licensing with prototype testing features.

11. Fuel Cycle Approach

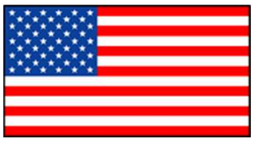
The baseline FMR design is based on once-through fuel cycle with interim storage and direct disposal. Recycling is feasible through a proliferation-resistant dry process that uses only thermal-mechanical treatment of oxide fuel without any involvement of chemicals.

12. Waste Management and Disposal Plan

The radioactive waste management systems segregate, collect, process, store, and dispose of radioactive liquid, gaseous, and solid wastes. The liquid waste system includes filtration and demineralization, discharging processed wastes to the solid waste system. The gaseous waste system reduces the activity of the waste gas to a level acceptable for discharge to the atmosphere. The solid waste systems include collection, solidification, packaging, and storage facilities for radioactive materials prior to their offsite shipment for disposal.

13. Development Milestones

2024	Conceptual design of reactor and power plant: pre-application licensing engagement; economic analysis
2030	Risk reduction: demonstration of reactor operation; demonstration of large components; fuel qualification
2035	Commercialization: prototype reactor and plant; demonstration of plant operation; licensing



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

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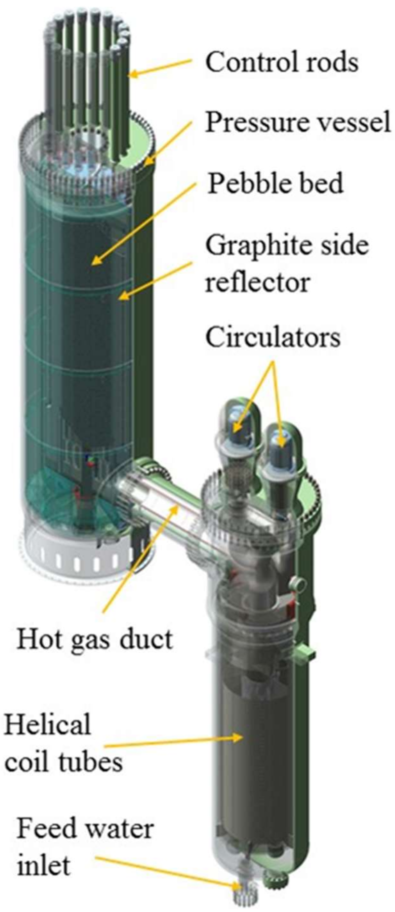


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	X-energy, LLC, United States of America
Reactor type	Modular HTGR
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	200 / 82.5
Primary circulation	Forced helium circulation
NSSS Operating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

1. Introduction

The Xe-100 is a pebble bed high-temperature gas-cooled reactor with thermal rating of 200 MW. It features a continuous refuelling system with low enriched fuel spheres or pebbles of approximately 15.5 wt% entering the top of the reactor and passing through the core six (6) times to achieve a final average burnup of 165 000 MWd/tHM.

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Reactor Core

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

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

(b) Fuel Characteristics

TRistructural ISOtropic (TRISO) particles are embedded in a graphite matrix pebble to form the fuel element. Particles contain coated uranium oxide and carbide (UCO) kernels enriched at 15.5 wt% and are slightly smaller in diameter (425 µm) than the usual UO_2 (500 µm) fuel kernels used in Germany and China. The optimized moderation ratio (NC/NA) yields a heavy metal loading of around 7 g/pebble. This enables the Xe-100, under worst case water ingress scenarios, to be shut down with its reactivity control and shutdown system (RCSS). Moreover, the graphite shell does not melt but sublimes (changes into vapor) at > 3920°C (4200K) and fuel temperature never exceeds 1100°C during normal operation. Therefore, X-energy does not have to bear the same magnitude of costs related to the pressure vessel, containment building, or safety systems as those of a traditional nuclear plant.

(c) Fuel Handling System

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

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

(d) Reactivity Control

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

(e) Reactor Pressure Vessel and Internals

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal/ Reactor Cooling Philosophy

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

(c) Containment Function

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

6. Plant Safety and Operational Performances

The design has the following inherent safety characteristics and design features:
Non-metallic fuel elements – meltdown proof and efficient retention of radionuclides in the TRISO- coated particle fuel during normal operation allows for relatively clean helium circuits and plant operations with low contamination of cooling gas and radioactivity release;
Helium – Chemically and radiologically inert helium is an effective heat transport fluid. Moreover, it does not influence the neutron balance. Helium allows for very high coolant temperatures;
Graphite core structures – allows for high-temperature operations and provides high thermal inertia to the reactor resulting in slow transient response during a loss of active cooling.

(a) Engineered Safety System Approach and Configuration

The following is credited as safety systems (active and passive):
- Coated particle fuel elements;
- Reactor protection system (RPS);
- Core support structures;
- RPV;
- Reactor building

(b) Operational transients and accidents

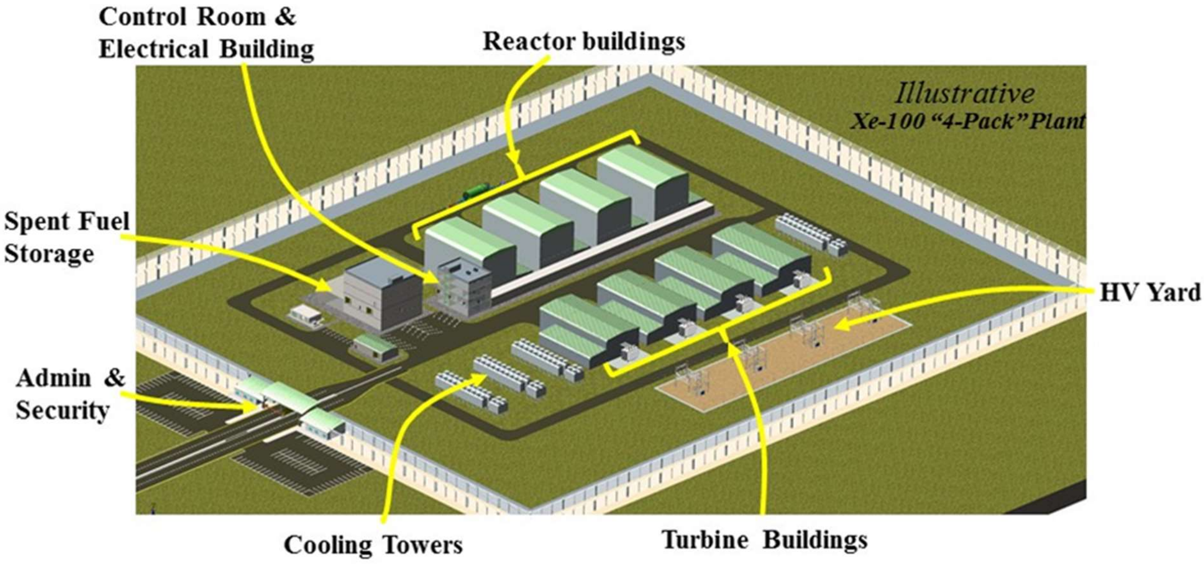
(i) Key safety features to limit plant transients:
The RCSS insertion depth during normal operation binds around 1.4 $\delta_{k\text{-eff}}$. Any spurious signal that would cause full withdrawal of the RCSS would therefore only translate to a higher temperature that would remain below an allowable value shown experimentally not to cause any fuel damage. Furthermore, because the reactor core and its internals are mostly graphite, this provides a high thermal inertia that would cause any transient to be slow-acting.
(ii) Key safety features to avoid core damage:
Features include the reactor core with a low power density, which is very robust and has a high thermal capacity to make the reactor thermally stable during all operational and controlled procedures. Strong negative temperature coefficients also contribute to the excellent inherent safety characteristics.
(iii) Key safety features to contain core damage:
Core meltdown proof – no Core Damage Frequency
(iv) Key safety features to reduce or eliminate large offsite release;
Multiple – independent fission product barriers:
- Qualified UCO TRISO coated particle fuel provides retention of fission products at the source;
- ASME designed pressure boundary provides a further reliable physical barrier to retain the small amount of fission products that may end up circulating in the helium and in graphite dust particles;
- A filtered and vented reactor building.

(v) Diversity and redundancy:
A series of independent fission product barriers provides redundancy and diversity. Failure of any one individual barrier will not impact the performance of another neighboring system/barrier.
(vi) Worst accident scenario and release:
The Depressurized Loss of Forced Coolant (DLOFC) is the worst-case accident scenario. This assumes the RCSS has also failed to insert. Under this scenario no fuel damage will be experienced.

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement



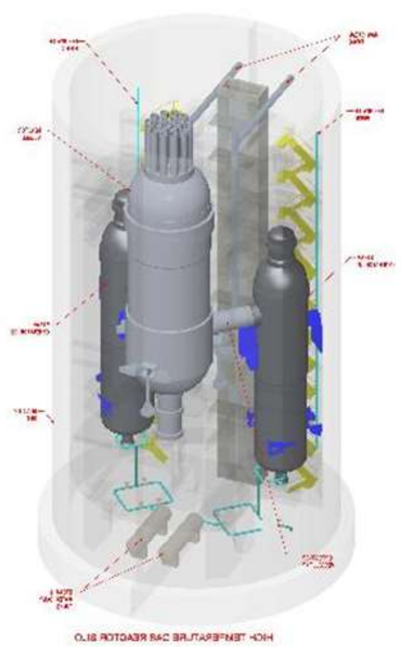
9. Design and Licensing Status

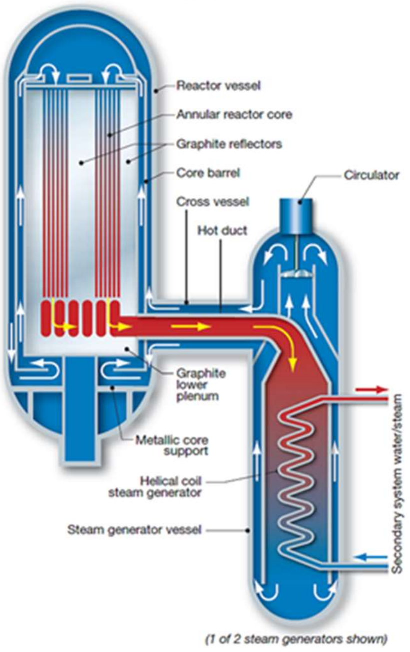
Conceptual design development and U.S. Nuclear Regulatory Commission pre-licensing phase.

10. Development Milestones

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

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





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Framatome Inc., United States of America
Reactor type	Prismatic block HTGR
Coolant/moderator	Helium/graphite
Thermal/electrical capacity, MW(t)/MW(e)	625 / 272
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	6 / 16
Core Inlet/Outlet Coolant Temperature (ºC)	325 / 750
Fuel type/assembly array	UCO TRISO particle fuel in hexagonal graphite blocks
Number of fuel assemblies in the core	Annular core, 102 column, 10 blocks per column
Fuel enrichment (%)	14.5 average / 18.5 maximum
Core Discharge Burnup (GWd/ton)	165
Refuelling Cycle (months)	½ of the core replaced every 18 months; 18 days outage
Reactivity control mechanism	Control rods (gravity insertion) Independent reserve shutdown system (gravity insertion) large negative temperature coefficient
Approach to safety systems	Active and Passive
Design life (years)	80
Plant footprint (m^2)	8,000
RPV height/diameter (m)	24 / 8.5
RPV weight (metric ton)	880
Seismic Design (SSE)	0.5g w/seismic isolators
Fuel cycle requirements / Approach	LEU once-through fuel cycle / reprocessing options for later consideration.
Distinguishing features	Coated particle fuel; passive decay heat removal; passive safety; high temperature process steam; vented reactor building; 400m EPZ (emergency planning zone); underground construction; modular construction; available in a range of stationary (s) and mobile (m) power levels: 315(s), 180(s), 54(s), 10(s) and 7m (MWth)
Design status	Preliminary design

1. Introduction

The Framatome reference scale SC-HTGR is a modular, graphite-moderated, helium-cooled, high temperature reactor with a nominal thermal power of 625 MW(t) and a nominal electric power capability of 272 MW(e). It produces high temperature steam suitable for numerous applications including industrial process heat and high efficiency electricity generation. The SC-HTGR builds on Framatome's experience of HTGR projects, as well as on the development and design advances that have taken place in recent years for modular design and construction of HTGRs. The overall configuration takes full advantage of the work performed on early modular HTGR concepts such as the General Atomics MHTGR, the KWU/INTERATOM HTR-MODUL, and ANTARES Framatome's very high temperature gas-cooled reactor design concept.

2. Target Application

The SC-HTGR produces high temperature steam suitable for numerous applications including industrial process heat, Hydrogen production, and high efficiency electricity generation. The HTGR steam cycle concept is extremely flexible. The steam cycle is also well suited to cogeneration of electricity. The SC-HTGR is configured in a verity of stationary and mobile options and power levels.

3. Design Philosophy

The SC-HTGR is designed around proven thermal neutron spectrum, helium-cooled, graphite moderated reactor technology, and passive decay heat removal. The key to the safety and economic potential of this technology is the proven radionuclides retention capabilities of UCO kernel TRISO coated fuel particles.

4. Main Design Features

(a) Power Conversion

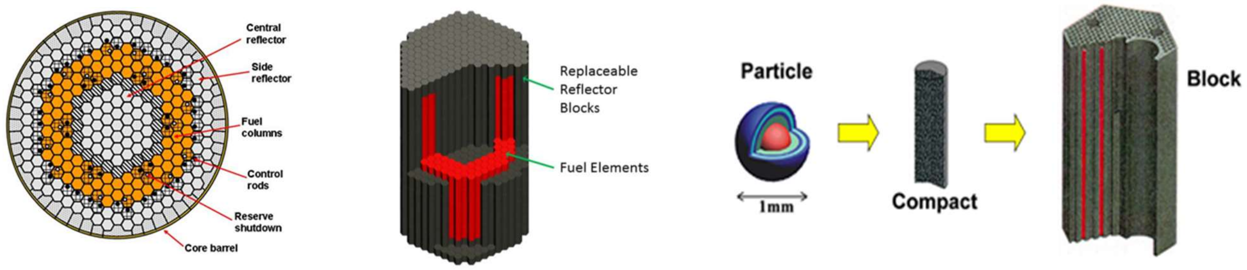
The reference four-module steam cycle concept plant uses the standard Rankine cycle turbine-generator set per reactor module. A single reference design reactor module produces 625 MW(t). This design is also available in a variety of lower powered stationary and mobile configurations.

(b) Reactor Core

The reactor inlet and outlet temperatures are 325°C and 750°C, respectively. These temperatures were selected primarily to support the desired steam outlet conditions for the target markets. These temperatures also allow the use of SA-508/533, a standard PWR vessel material, for the primary system vessels without requiring cladding, separate cooling, or special thermal protection. The active core is 10 blocks high. The reference SC-HTGR reactor module design can be scaled from 625 MW(t) to 7 MW(t) per module using the same fuel blocks in scaled arrangements.

(c) Fuel Characteristics

The TRISO coated fuel particle consists of a uranium oxycarbide (UCO) fuel kernel surrounded by multiple ceramic and pyrolytic graphitic coating layers that provide the primary fission product retention barriers under all design basis accident conditions. The total fuel inventory includes roughly 10 billion such particles per core. The particles are distributed in graphitic cylindrical compacts. Multiple compacts are contained within hexagonal nuclear grade graphite fuel blocks. The compacts are stacked in fuel holes drilled into the blocks each fuel holes is surounded by multiple helium flow channels tranporting the fission heat to the steam generators.



The core cycle length for the reference SC-HTGR is between 420 and 540 effective full-power days. This has been confirmed for the initial core, using an initial core loading of 10.36 w/o ^{235}U enriched particles with a packing fraction of 0.289 for all fuel elements in the core, and for reloads utilizing half-core replacement with fuel blocks having a 15.5 w/o ^{235}U enrichment and a packing fraction of 0.279.

(d) Fuel Handling System

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

(e) Reactivity Control

The large negative temperature coefficient of the modular SC-HTGR, along with its large thermal margins, provide for an inherent shutdown capability to deal with failures to scram the reactor. In addition, two gravity-driven and diverse and active reactivity control systems provide further confidence of the ability to shut down the reactor.

(f) Reactor Pressure Vessel and Internals

The Reactor Vessel is part of the Vessel System which is the primary pressure-retaining components and includes the Cross Vessels and Steam Generator Vessels. The reactor core, reflector elements, core support structure, and core restraint devices are installed in the reactor vessel. The reactor core components, together with elements of the reactor internal components, constitute a graphite assembly that is supported on a graphite core support pillars and restrained by a metallic core lateral support system. The reactor internal components consist of the upper core restraint elements, permanent graphite side reflector elements, graphite core support pillars, metallic core support assembly, and the upper plenum shroud.

(g) Reactor Coolant System

The concept of “Reactor Coolant” as a system is unique in gas cooled reactors. The reactor fission heat is transported to the steam generators via a heat transport medium i.e., pressurized helium gas. In a normal shut down condition the core heat is either removed by the steam generators or transported to a water-cooled shutdown cooling system (SCS). In an accident condition the core decay and residual heat is absorbed within the internal graphite structure; subsequently and passively transferred to and removed by the water-cooled reactor cavity cooling system (RCCS).

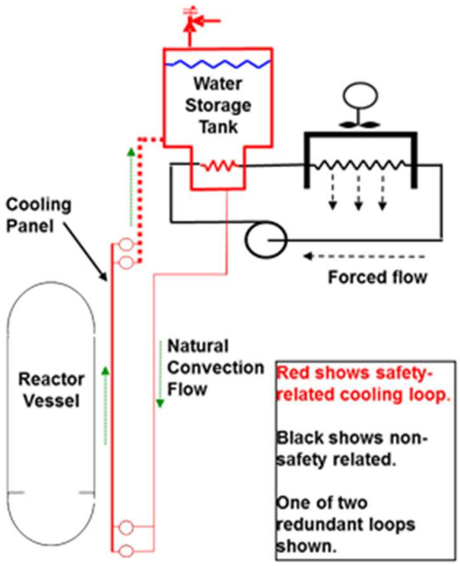
5. Safety Features

(a) Engineered Safety System Approach and Configuration

The primary safety objective of the SC-HTGR design is to limit the dose from accidental releases so that the dose limits established by the U.S. EPA Protective Action Guides are met at an exclusion area boundary (EAB) which is located a few hundred meters from the reactor. To achieve this safety objective, the design uses the high temperature fission product retention capabilities of TRISO-coated fuel particles, graphite moderator, and helium coolant, along with the passive heat removal capability of a low power-density core and an un-insulated steel reactor vessel. The SC-HTGR is designed to passively remove decay heat from the core regardless of whether the primary coolant is present. The concrete walls surrounding the reactor vessel are covered by the Reactor Cavity Cooling System panels, which provide natural circulation cooling during both normal operation and accidents, so there is no need for the system to actuate, change modes, or configuration in the event of an accident. Moreover, the thermal characteristics of the reactor are such that even if the RCCS were to fail during an accident, the safety consequences would still be acceptable. No powered safety-related systems and no operator actions are required to respond to any of the accident scenarios that have been postulated for the various modular HTGR concepts, including the SC-HTGR, throughout the modular HTGR licensing history.

(b) Decay Heat Removal System/ Reactor Cooling Philosophy

The SC-HTGR has three heat removal systems. The two main cooling loops transfer heat to the secondary circuit during normal operation. When maintenance is being performed on the main cooling loops, a separate shutdown cooling system is available. This system uses a separate and independent circulator and heat exchanger located at the base of the reactor vessel. These systems also provide cooling during refueling and normal shutdown conditions as well as most Anticipated Events and DBEs.



The Reactor Cavity Cooling System (RCCS) , shown above, is a redundant natural circulation water-cooled system that maintains acceptable concrete temperatures in the reactor cavity during normal operation and Anticipated Events, and maintains acceptable fuel, vessel, and concrete temperatures during Design Basis Accidents. Each independent loop of the safety related RCCS consists of heat collecting panels in the cavity surrounding the reactor vessel connected by a natural circulation loop to a water storage tank.

(c) Spent Fuel Cooling Safety Approach / System

The fuel blocks in the SC-HTGR family of modular reactors are stationary in the core and are replaced at an eighteen to twenty-four (18-24) months intervals. For the reference SC-HTGR (625 MW(t) per module), one half of the of the fuel assemblies (510 assemblies) are replaced at each refueling interval. The refueling operations takes place by the semi-autonomous refueling machine under remote control of the refueling machine operator.

(d) Containment Function

The radionuclides containment function in the SC-HTGR is performed primarily by the TRISO fuel coatings. The graphite core structures, primary coolant boundary, and reactor building provide supplemental containment capability. The SC-HTGR reactor building is vented to the atmosphere during a primary system depressurization phase. The building provides aircraft crash protection and supplemental fission product retention in the event of aircraft crash or a primary system depressurization accident.

(e) Chemical Control

Approximately 2-percent of the primary circuit helium is continuously extracted by the helium purification system (HPS), cooled, filtered, purified, and returned to the primary circuit. The primary circuit helium chemistry is strictly controlled by the HPS which maintains the purity and chemistry of the helium within a prespecified standard limits.

6. Plant Safety and Operational Performances

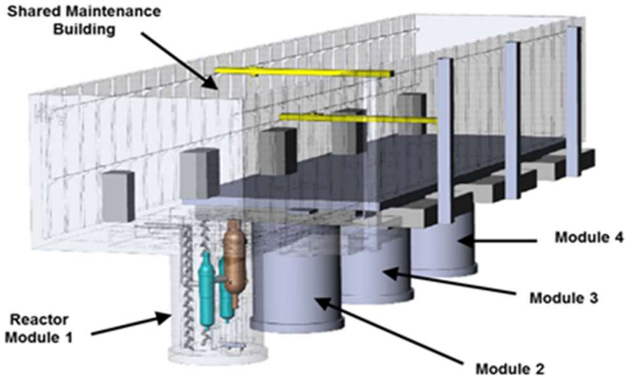
The nominal electricity generation performance of the SC-HTGR system has been evaluate, considering preliminary efficiency estimates for the helium circulators, feedwater pumps, turbine, generator, and other plant electrical loads. In an all-electric mode of operation, the net electrical output from each 625 MW(t) reactor module is 272 MW(e), for a net efficiency of 43.5%.

7. Instrumentation and Control Systems

The SC-HTGR Instrumentation and Controls (I&C) include the instruments used for plant protection, monitoring, and control. The plant design goal is to utilize commercially proven I&C hardware and software with demonstrated reliability.

8. Plant Layout Arrangement

Each reactor module is located in a separate reactor building. The standard configuration uses a fully embedded below grade reactor building design as shown below. This provides structural design advantages and superior protection from external hazards. An alternative partially embedded configuration can be used for sites where a fully embedded structure is not appropriate. The primary functions of the reactor building are to support the NSSS primary circuit components and to protect the system from external hazards.



9. Testing Conducted for Design Verification and Validation

The Framatome SC-HTGR line of high temperature reactor concept possess high technology readiness levels (TRLs) for most if not all its material needs and components. Therefore, no additional testing and V&V beyond completion of on-going fuel qualification program (AGR) and Graphite characterization research (AGC) underway by the U.S.A. Department of Energy (DOE) is necessary.

10. Design and Licensing Status

The SC-HTGR concept design and preparatory work for pre-license application is underway.

11. Fuel Cycle Approach

The high thermal efficiency and high fuel burnup of the SC-HTGR support sustainability for current once-through fuel cycles by minimizing spent fuel volume. The HA-LEU once through fuel cycle requires about 6.8 MTHM/GW(e)-yr that equates to a natural uranium feedstock utilization of about 224 MT/GW(e)-yr.

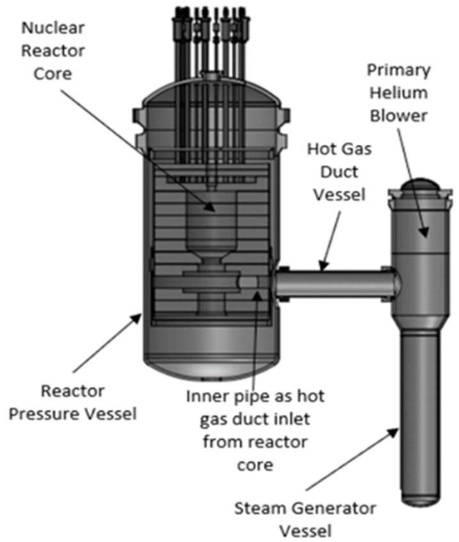
12. Waste Management and Disposal Plan

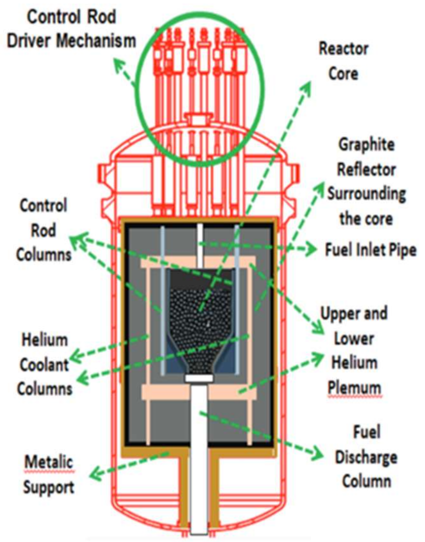
The storage and disposal requirements will be lessened by about 50% for HTGRs as compared to LWRs. The spent fuel capacity is 10 years. Coated particles represent excellent radioactivity containment characteristics. For long term storage or disposal, volume reduction and reprocessing are also options studied in the past.

13. Development Milestones

2022	Initial Design – On-going
2026	Basic Design (Pending commitment from FOAK customer)
2027	Start of pre-licencing vendor design review in the USA.
2027	FOAK plant engineering design complete; Secure necessary licenses in the USA.
2028	Start construction of a first full-scale NPP module in the USA.
2033	Commercial operation

PeLUIt / RDE (BRIN, Indonesia)





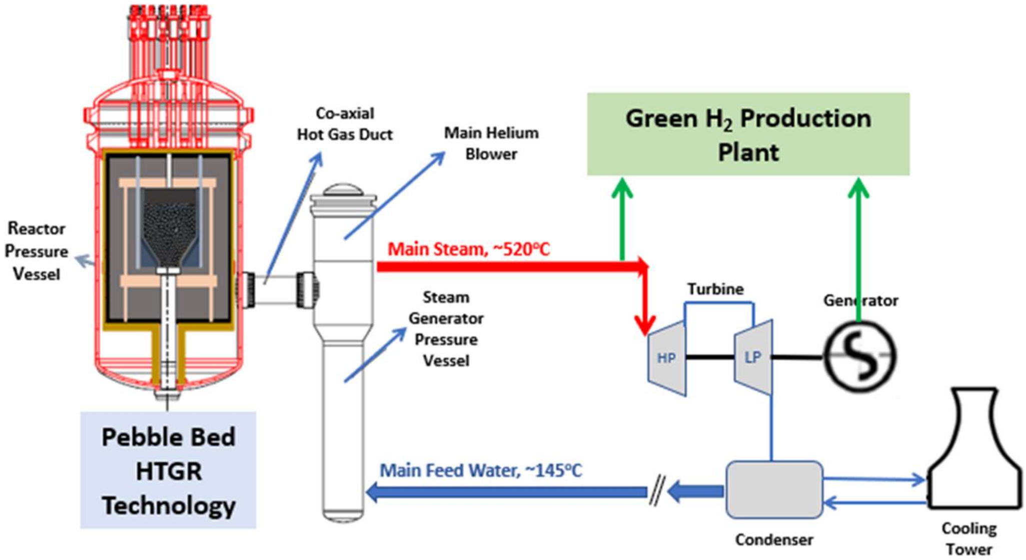
MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	National Research and Innovation Agency (BRIN), Indonesia
Reactor type	Pebble bed high temperature gas-cooled reactor
Coolant/moderator	Helium/graphite
Thermal Power	40MWt and 10MWt
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	3 / 6
Core Inlet/Outlet Coolant Temperature (°C)	250 / 750
Fuel type/assembly array	Spherical elements with coated particle fuel
Number of fuel assemblies in the core	27 000
Fuel enrichment (%)	17
Core Discharge Burnup (GWd/ton)	80
Refuelling Cycle (months)	On-line refuelling
Reactivity control mechanism	Control rod and small absorber sphere
Approach to safety systems	Combined active and passive
Design life (years)	40
Plant footprint (m²)	~ 24 000
RPV height/diameter (m)	11.1 / 4.2 (inner)
RPV weight (metric ton)	180
Seismic Design (SSE)	0.26g
Fuel cycle requirements / Approach	LEU, open cycle, spent fuel intermediate storage at plant
Distinguishing features	Inherent safety, no need for offsite emergency measures
Design status	Conceptual design of uprated 40MWt is in progress. For the initial 10MWt RDE, site license issued in 2017.

1. Introduction

PeLUIt stands for *Pembangkit Listrik dan Uap-panas Industri* (Indonesian) that means Nuclear Power Plant for Cogeneration of Electricity and Industrial Heat. In Indonesia the word also means a ‘whistle’ which represent 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 programmes to support the National Medium Term Development Plant (RPJMN) for 2015-2019. The main goal of the RDE development Programme is to build the national capability to be a nuclear reactor technology developer by mastering the design, construction project management, commissioning and operation of a nuclear power reactor. Furthermore, the nuclear reactor type selected for the programme should become the prototype design to be scaled up and commercialized to contribute in enhancing the national energy supply security. The Pebble Bed Reactor (PBR) type of High Temperature Gas-cooled Reactor (HTGR) was selected as the technology for the RDE Programme. A sound safety feature and flexible applications of the PBR technology are among the reasons of this decision. PBR has a high safety level shown by a small radioactive release to the environment on any probable accident.
As part of licensing procedure, BATAN (currently BRIN) already received the RDE Site Licensing from

National Nuclear Regulatory Body (BAPETEN) in January 2017. Design Approval phase was already started in 2018 and 2019, in which BATAN needs to submit the detail design and safety analysis report of RDE to BAPETEN. However, due to change in main policy of nuclear reactor development, the licensing processes was halted. Although the research and development of RDE are still continuing, particularly related to safety analysis and its cogeneration potential. Since the end of 2021, as worldwide trend of hydrogen as an important key in energy transition is stronger, cogeneration application for hydrogen production becoming the main focus. In particular, a collaboration with an energy state-own company called PERTAMINA was started to develop a hydrogen production system based on PeLUIt/RDE. Guided by the initial techno-economic analysis for the hydrogen production using HEEP Software from IAEA an uprated power level to have a more superheated steam and electrical output in parallel is needed. Also based on initial safety analysis, an uprate up to 40MWt, is possible with generally the same core geometry of previous RDE design while maintaining its passive safety features.

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, main steam blower, and the steam generator pressure vessel. The rest of balance-of-plant system is common to other power plant such as coal power plant. While parts of the RPV-internals are shown in the right side of 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 dimension are kept.



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

2. Target Application

The initial target of PeLUIt/RDE was as an experimental power reactor for research, which basically focuses to demonstrate the operational and safety performance of a pebble bed high temperature gas-cooled reactor (PB-HTGR). It will demonstrate the capability of the reactor to survive even in the most severe accident scenarios such as depressurized loss of forced cooling (DLOFC), in addition also to perform a cogeneration experiment utilizing its high steam temperature output.
In collaboration with national industries such as PT. Pembangunan Perumahan (PT. PP) and PT. Rekadaya Elektrika Consult (Reconsult), a direct utilization of PeLUIt as electric source for many islands in Indonesia, replacing an expensive diesel fuel, was foreseen. A joint cooperation with national industries to study and develop the techno-economic and detail design of the reactor was already started in 2018-2019, which was halted. The new focus is to utilize PeLUIt as a green hydrogen production in collaboration with PT. PERTAMINA.

3. Design Philosophy

The PeLUIt/RDE is designed based on an established pebble-bed reactor principle. The design employs the TRISO-based fuel which provide a sound fission product retention capability resulting in allowable release of radioactive material to the environment in any condition of the core including the most severe postulated accident. The core and reflector are dominantly composed by a graphite which gives a good heat transfer and neutronic features. High heat conductivity and capacity of the graphite improve the heat transfer characteristic of the design. While it helps to improve the thermal neutron spectrum of in the core due to its effective neutron thermalization capability. An inert He gas as the coolant avoids any chemical or physical reactions. The nuclear island is applying a typical side by side arrangement of the RPV and Steam Generator Vessel which connected by a hot gas duct vessel. Each reactor module includes a reactor pressure vessel (RPV); graphite, carbon, and metallic reactor internals; a steam generator; and a main helium blower. The thermal power of each reactor module is uprated to 40MWt as an additional to 10 MWt option. Both of this power level option have a different pebble fuel consumption. The helium temperatures at the reactor core inlet/ outlet are 250/700°C, and steam parameters is 6 MPa/520°C at the steam turbine entrance. The nuclear island is coupled with a ~13MWe or 3 MW(e) steam turbine.

4. Main Design Features

(a) Reactor Core and Power Conversion Unit

The primary helium coolant works at the 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 flow into the pebble bed in a downward flow pattern. Bypass flows are introduced into the fuel discharge tubes to cool the fuel elements there and into the control rod channels for control rods cooling. Helium is heated up in the active reactor core and then is mixed to the average outlet temperature of 700ºC and then flows to the steam generator. The hot helium then transfers its energy to the 243ºC feed water in the steam generator to have a superheated steam of 520ºC at 6 MPa flowing to the turbine to generate a ~13MWe or ~3MWe.

(b) Fuel Characteristics

Fuel elements are spherical ones. Every fuel element contains 5 g of heavy metal. The equilibrium core has 17% enrichment of ^{235}U. Uranium kernels of ~0.5 mm in diameter is 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.

(c) Fuel Handling System

The operation mode of PeLUIt/RDE adopts continuous fuel loading and discharging: the fuel elements are pneumatically lifted into the upper part of the reactor, drop into the reactor core using a single fuel loading tube, then move downward across the core and through a discharging tube at the core bottom. The fuels will pass one-by-one through the singulator. The geometry of discharged fuel elements is checked in the fail-fuel separator. Failed fuel with geometrical defects will be separated and diverted into the failed fuel cask, while the good ones will continue to the burn-up measurement facility. Fuel pebbles that already reached the burnup target will be collected in the spent fuel cask while the other will be redirected back into the core. In average, a single pebble fuel will pass the core 5 times to reach the average discharge burnup target of 80 MWd/kg. A Once-Through-Then-Out (OTTO) fuel management scheme is being investigated to have a simpler design which finally improve its green hydrogen techno-economic performance.

(d) Reactor Pressure Vessel and Internals

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

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

(b) Reactivity control

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

(c) Reactor Cooling Philosophy

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

(d) Containment Function

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

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control Systems

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

8. Design and Licensing Status

For the updated 40MWt PeLUIt/RDE including the hydrogen production system is still in the conceptual design phase. 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 was already started in 2018 and 2019, the licensing of 10MWt RDE was halted.

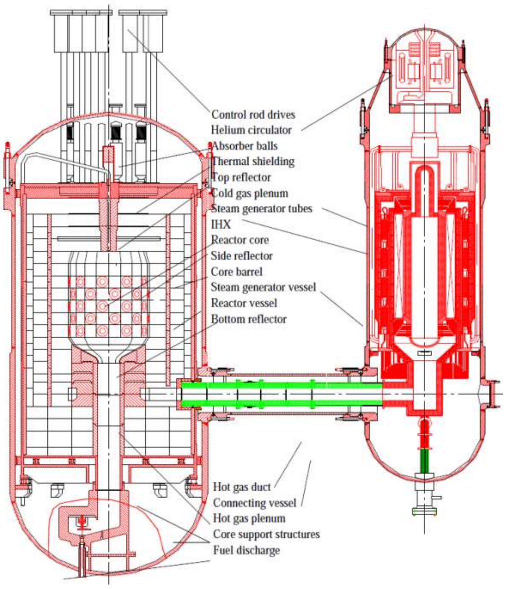
9. Fuel Cycle Approach

In the adopted fuel cycle, the pebble fuels that already reach its burnup limit are collected in the spent fuel casks. These spent fuel casks are placed in the spent fuel storage room in the reactor building. The spent fuel storage room is equipped with the radiation and temperature monitoring and control system. The casks and the room are designed to avoid criticality of the spent fuels. The cask is designed so that it can be placed in a standard LWR transport cask and be transported if necessary.

10. Development Milestones

2015-2016	Development of 10 MWt RDE Conceptual Design and start of its Site Licensing Phase
2017-2018	Site License issued by BAPETEN; RDE Basic Design development started
2018-2019	Review of 10MWt RDE Design and initial Design Approval phase was performed.
2020-2021	Licensing process of 10MWt RDE was halted. Development of safety analysis is still progressing.
2021-2022	An uprated 40MWt PeLUIt/RDE conceptual design started with focus to develop a green hydrogen production system.

HTR-10 (Tsinghua University, China)



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

1. Introduction

In 1992, the China Central Government approved the construction of the 10 MW(t) pebble bed high temperature gas cooled test reactor (HTR-10) in Tsinghua University's Institute of Nuclear and New Energy Technology (INET). In 2003, the HTR-10 reached its full power operation. Afterwards, INET conducted many experiments using the HTR-10 to verify crucial inherent safety features of modular HTRs, including (i) loss of off-site power without scram; (ii) main helium blower shutdown without scram; (iii) withdrawal of control rod without scram; and (iv) Helium blower trip without closing outlet cut-off valve. The second step of HTR development in China began in 2001 when the high-temperature gas-cooled reactor pebble-bed module (HTR-PM) project was launched.

2. Target Application

The HTR-10 is a major project on the energy sector within the Chinese National High Technology Programme, serving as the first major step of the development of modular HTGR in China. Its main objectives are to: (1)

explore the technology in the design, construction and operation of HTGRs; (2) establish an irradiation and experimental facility; (3) demonstrate the inherent safety features of modular HTGR; (4) test electricity and heat co-generation and closed cycle gas turbine technology; and (5) perform research and development work on nuclear process heat application. The aims of this project are to demonstrate the inherent safety features of the HTGR modular design and test the technologies of electricity generation, district heating as well as process heat application with modular HTGR.

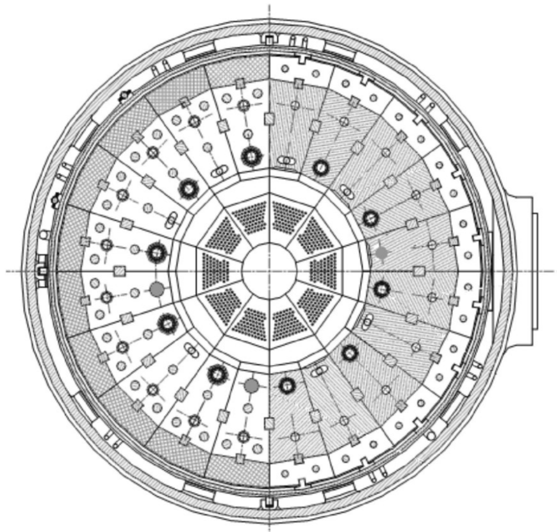
3. Design Philosophy

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

4. Main Design Features

(a) Reactor Core

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



(b) Reactor Pressure Vessel and Internals

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

(c) Reactor Coolant System

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

(d) Steam Generator

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

(e) Helium Circulator

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

(f) Fuel Characteristics

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

(g) Fuel Handling System

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

(h) Reactivity Control

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

5. Safety Features

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

(a) Reactivity control

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

(b) Decay Heat Removal System

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

(c) Containment Function

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

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control Systems

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

HTR-10 Control room

8. Plant Layout Arrangement

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

9. Design and Licensing Status

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

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

To be included in the national plan of test facilities.

12. Research and Development Plan

From 1986 to 1990, eight (8) research topics for key technologies were defined: (i) a conceptual design and the supporting reactor physics and thermal fluid design and safety software codes; (ii) manufacturing process of the fuel spheres; (iii) reprocessing technologies for the thorium-uranium cycle; (iv) core internal graphite structure design and supporting analysis; (v) helium technology establishment, (vi) pressure vessel designs, (vii) the fuel handling design; (viii) development of special materials.
Before the commissioning, the following engineering experiments were conducted: (i) a hot gas duct performance test; (ii) measurements to establish the mixing efficiency at the core bottom (limit stratification and heat streaks); (iii) two-phase flow stability tests on the once-through steam generator; (iv) fuel handling performance test; (v) control rods drive mechanism performance; (vi) V&V of the digital reactor protection systems; (vii) measurements to confirm the neutron absorption cross-section of the reflector graphite and (viii) a performance test for the helium circulator.

13. Development Milestones

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

HTTR (JAEA, Japan)

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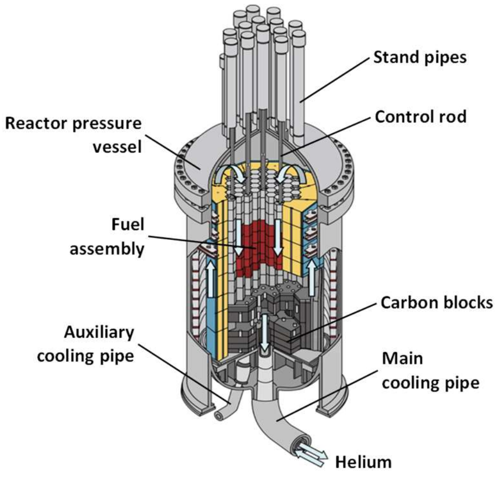
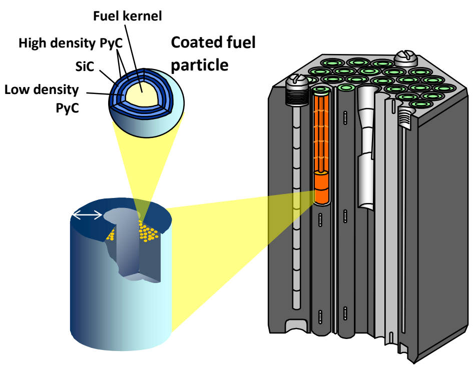




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JAEA in cooperation with MHI, Toshiba, IHI, Hitachi, Fuji Electric, NFI, Toyo Tanso, Japan
Reactor type	Prismatic HTGR
Coolant/moderator	Helium / graphite
Thermal power, MW(t)	30
Primary circulation	Forced by gas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

1. Introduction

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

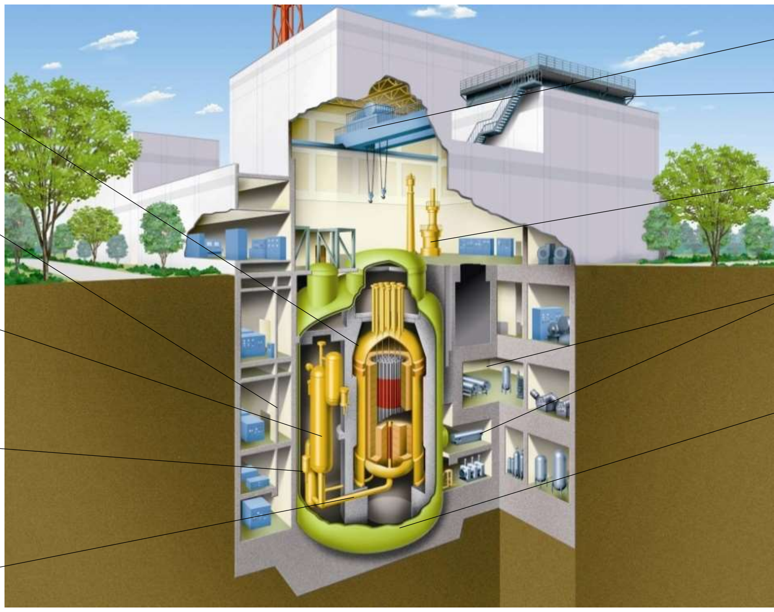
2. Target Applications

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

3. 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) Reactor Core

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

(b) Fuel

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

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

(c) Reactivity control system

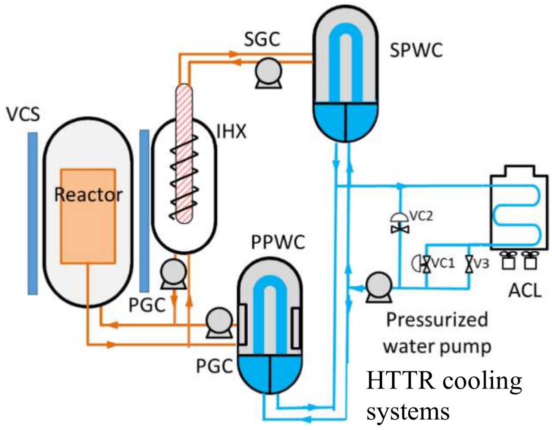
The HTTR contains two reactivity control systems, including a control rod system and a reserve shutdown system (RSS). The control rod system comprises of 16 pair of control rods made of B_4C. Each pair of control rods can move individually by control rod drive mechanisms located in standpipes at the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

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

(d) Cooling systems

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

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



HTTR cooling systems

5. Safety Features

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

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

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

6. Plant Safety and Operational Performance

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

a) Pre-operational test

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

b) Start-up physics test

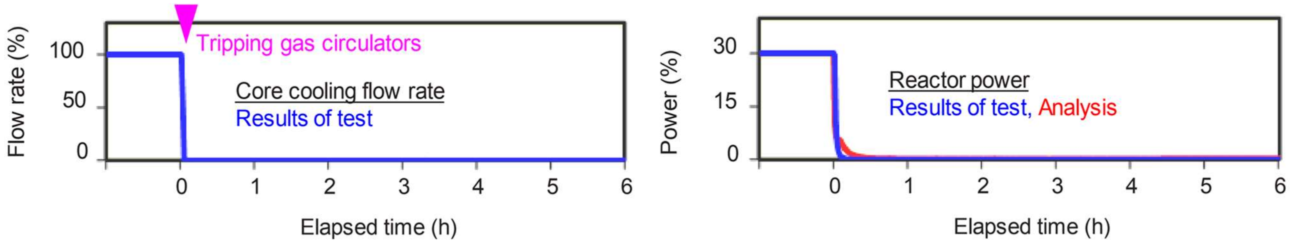
Fuel loading to the reactor started in July 1998, and the first criticality was attained on November 10th, 1998. The fuel blocks were column-wise loaded from the outer fuel columns to the inner. The first criticality was achieved successfully with 19 fuel columns loading. After that, the other inner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

c) Rise to power test

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

d) Safety demonstration test

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



e) Continuous operation

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

7. Instrumentation and Control Systems

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

8. Plant Layout Arrangement

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

9. Design and Licensing Status

The HTTR construction started in 1991 with first criticality accomplished in 1998. Details of the operational achievements is given below. The HTTR has not been in operation since the great east Japan earthquake occurred in March 2011. In conformity with new regulatory requirements by Nuclear Regulation Authority of Japan, JAEA received permission and restarted operation of the HTTR on July 2021.

10. Development Milestones

1969 – 1984	Conceptual design (4 years); System integrity design (6 years) and Basic design (3 years)
1985 – 1990	Detail design (3 years); Application and permission of construction (1 year)
1991 – 1997	Construction
1998	First criticality
2001, 2002	Reactor outlet coolant temperature of 850ºC; Safety demonstration test
2004	Reactor outlet coolant temperature of 950ºC
2007; 2010	850ºC/30 days operation; 950ºC/50 days operation and Safety demonstration tests
2014	Conformity review on the New Regulatory Requirements start toward resume operation
2020	Permission toward the restart of HTTR
2021.07.30	Operation restarted

11. Future plans

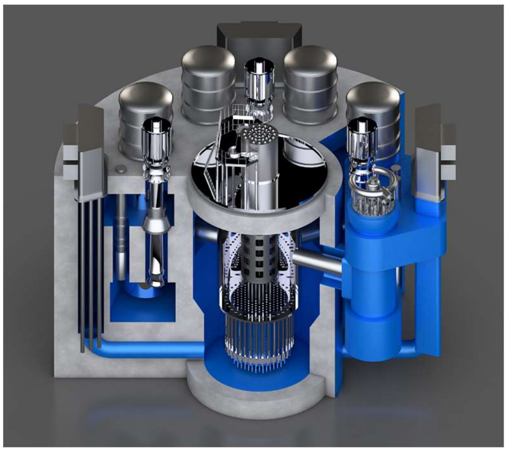
Following the restart of the HTTR, a safety demonstration test in the OECD/NEA LOFC project was carried out on January 2022. New test project to demonstrate hydrogen production by 2030 using the high temperature heat from the HTTR has been launched. Other than these, a number of activities are planned to be carried out through operation of the HTTR including international cooperation and human-resource development. As the HTTR can be used as a test bed for international cooperation, JAEA plans to launch new international projects based on the operation of the HTTR, and welcomes discussion with potential partners.

12. Reference

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

PART III. LIQUID METAL COOLED FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS

BREST-OD-300
(NIKIET, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	Liquid metal cooled fast reactor
Coolant	Lead
Thermal/electrical capacity, MW(t)/MW(e)	700 / 300
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	0.1 / 17 – 18.5
Core inlet/outlet coolant temperature (°C)	420 / 535
Fuel type/assembly array	Mixed uranium plutonium nitride
Number of fuel assemblies in the core	169
Fuel enrichment (%)	up to 14.5
Refuelling cycle (months)	36 – 78
Core discharge burnup	61.45
Reactivity control mechanism	Reactivity compensation (RC), emergency protection (EP) and automatic control (AC) members
Approach to safety systems	Passive
Design life (years)	30
Plant footprint (m^2)	80 × 80
RPV height/diameter (m)	17.5 / 26
RPV weight (metric ton)	27 000
Seismic design (SSE)	VII-MSK 64
Fuel cycle requirements/approach	Closed fuel-cycle. It uses nitride of depleted U with Pu
Distinguishing features	High level of inherent safety due to natural properties of the lead, fuel, core and cooling design
Design status	Construction in progress

1. Introduction

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

2. Target Application

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

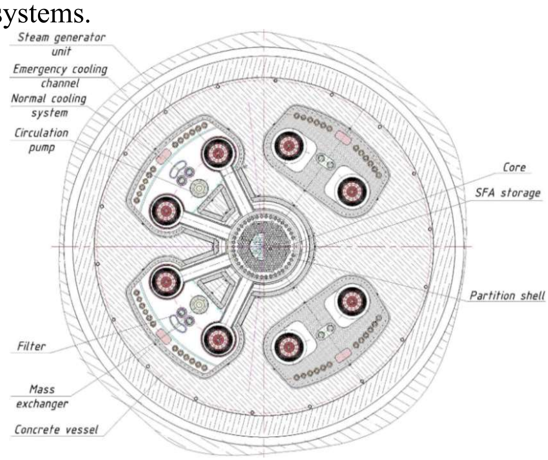
3. Design Philosophy

BREST-OD-300 is a pool-type reactor design with metal-concrete vessel. The objective of the design is to eliminate severe accidents; complete fuel breeding (equilibrium mode) for self-sustaining and exclusion of accidents caused by reactivity; integral-type arrangement of the primary circuit to avoid release of coolant outside the reactor vessel, to eliminate the loss of coolant; to use of low-activated lead coolant with high enough boiling temperature, without adverse interaction with water and air in the case of depressurizing of the circuit. The reactor facility has a two-circuit steam generating power unit that includes reactor core, steam generators (SG), main circulation pumps (MCP), fuel assembly (FA) loading system, control and protection system (CPS), steam-turbine unit, passive decay heat removal system, reactor heat-up system, reactor overpressure protection system, gas purification system and other auxiliary systems.

4. Main Design Features

(a) Power Conversion

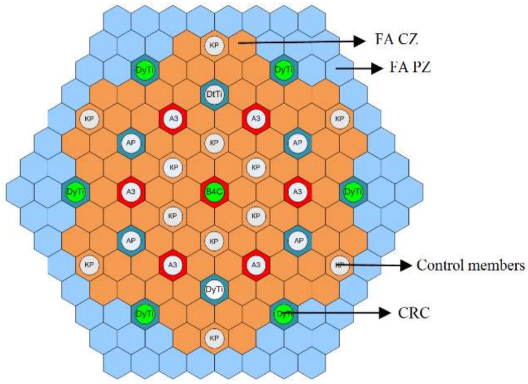
The pool-type reactor design has an integral lead circuit accommodated in one central and four peripheral cavities of the metal-concrete vessel. The central cavity houses the core barrel together with the side reflector, the CPS rods, the spent fuel assembly (SFA) storage and the shell that partitions the hot and the cold lead flows. Four peripheral cavities (according to the loop number) accommodate the SG and MCP, heat exchangers of the emergency and normal cool-down systems, filters of coolant and auxiliary components. The cavities have hydraulic interconnection.



BREST-OD-300 Layout

(b) Reactor Core

The lead coolant properties in combination with dense, high heat-conductivity nitride fuel provide conditions for complete plutonium breeding in the core (CBR ≥ 1). That results in a small operating reactivity margin ($\Delta\rho$ < βeff) and enables power operation without prompt criticality power excursions. The adopted fuel is mixed mononitride (U-Pu)N that features high density (14.3 g/cm^3) and high conductivity (20 W/m·K) and is compatible with lead and the fuel cladding of chromium ferritic-martensitic steel. The FA design allows radial coolant flow transfer in the core which prevents overheating of the damaged FA as shown in the figure on reactor core map.



BREST-OD-300 Reactor Core Map

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

Heat is removed from the reactor core through forced lead coolant (LC) circulation by pumps. The LC is pumped to the height of ~2 m relative to the lead level in the suction chamber and supplied to the free level of the annular pressure chamber. The lead further goes down to the core support grid, flows upward through the core where it is heated up to the temperature of 535°C, and enters the shared ‘hot’ coolant drain chamber. Then coolant flows up and enters the SG inlet cavities and inter-tube space via the distributing header nozzles. As flow goes down into the inter-tube space, the LC transfers heat to the secondary coolant flowing inside the SG tubes. Cooled down to ~ 420°C, the LC goes up in the annulus and flows out the pump suction chamber, where it is pumped out back to the pressure chamber. Exclusion of high pressure in the primary lead circuit and a relatively high lead freezing temperature contribute to crack self-healing, which eliminates the possibility of loss-of-core-cooling accidents and release of radioactive lead from the reactor vessel. Lead circulation through the reactor core and steam generator takes place due to the difference between the levels of cold and hot coolant generated by the pumps. Non-uniformity of lead flow through the steam generators with one of all pumps shut down is excluded, in so doing flow inertia in fast pump shutdown is provided by equalizing coolant levels in discharge and suction chambers.

(f) Secondary System

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

(g) Steam Generator

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

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

– in combination with application of (U-Pu)N fuel, complete breeding of fissile materials in the reactor core, which provides for a constant small reactivity margin preventing the disastrous effects of an uncontrolled power increase when introducing the reactivity margin because of equipment failures and personnel errors;

– to avoid the void effect of reactivity due to a high boiling point and high density of lead;

– to prevent coolant losses from the circuit in an event of the vessel damage due to high melting/solidification points of the coolant and the use of an integral (pool-type) layout of the reactor;

– to provide for high heat capacity of the coolant circuit which decreases a possibility of fuel damage;

– to allow for utilization of the high density of lead and its albedo properties for flattening the FA power distribution and the fuel pin temperatures respectively, as well as in the safety systems;

– to facilitate larger time lags of the transient processes in the circuit, which makes it possible to lower the requirements to the safety systems' rate of response.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

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

(c) Spent Fuel Cooling Safety Approach / System

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

(d) Containment System

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

(e) Chemical Control

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

6. Plant Safety and Operational Performances

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

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

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

7. Instrumentation and Control System

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

8. Plant Layout Arrangement

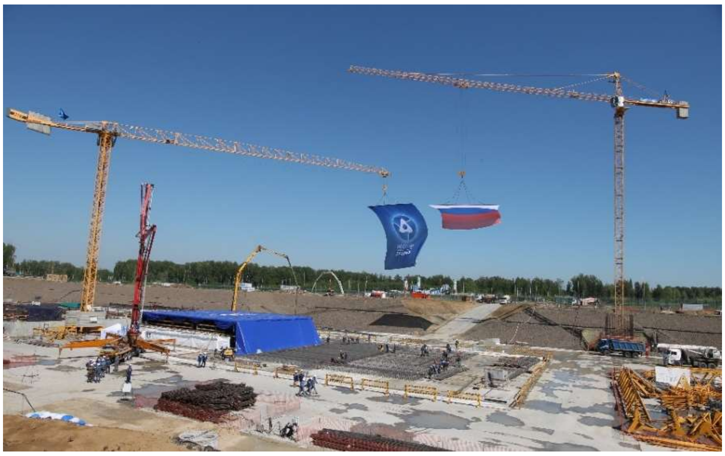
Plant main building consists of the reactor containment building, auxiliary building, compound building (CPB), emergency diesel generator building and turbine-generator building (TGB). The reactor building is mounted on a single monolithic reinforced concrete foundation plate. In order to reduce seismic inertia forces, the building is designed to be symmetrical with the footprint of 80×80 m.

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

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



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

11. Fuel Cycle Approach

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

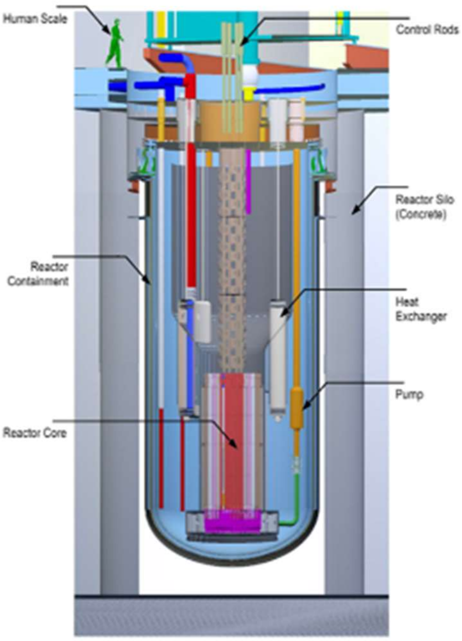
12. Waste Management and Disposal Plan

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

13. Development Milestones

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

ARC-100
(ARC Clean Energy, Canada)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ARC Clean Energy, Canada
Reactor type	LMFR (pool type)
Coolant	Sodium
Thermal/electrical capacity, MW(t)/MW(e)	286 / 100
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Non- pressurized
Core Inlet/Outlet Coolant Temperature (°C)	355 / 510
Fuel type/assembly array	Metal fuel (U-Zr alloy) based on enriched uranium
Number of fuel assemblies in the core	99
Fuel enrichment (%)	Avg. 13.1
Refuelling Cycle (months)	240
Core Discharge Burnup (GWd/ton)	77
Reactivity control mechanism	Control Rods
Approach to safety systems	Passive, diverse, redundant
Design life (years)	60
Plant footprint (m^2)	56 000
RPV height/diameter (m)	16.7 / 7.9
RPV weight (metric ton)	600
Seismic design (SSE)	0.3 PGA
Fuel Cycle requirements/approach	Metallic HALEU/Open cycle
Distinguishing features	Inherent safety with passive, diverse and redundant decay heat removal. Core lifetime of 20 years without refueling.
Design status	Preliminary design

1. Introduction

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

2. Target Application

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

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

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

3. Design Philosophy

The philosophy of the ARC-100 reactor is to rely on simple, passive safety features to achieve reactor safety under any normal operational occurrence or accident condition. The ARC-100 has adopted five traditional levels of safety for its defence in depth: i) Minimize risk by the prevention of abnormal operation and failure by maximizing safety margins; ii) Protection against abnormal operations and anticipated events via the large thermal inertia of the sodium pool; iii) Protection against DBA through diverse and redundant systems; iv) Control of severe plant conditions through designed passive and inherent safety characteristics of the facility; v) Protection of the public health and safety in case of accidents by designing the inherent and passive safety characteristics such that operator intervention and external power are not required for plant survival. Additionally, the design goal of the plant is that the evacuation zone is limited to the site boundary.

4. Main Design Features

(a) Power Conversion

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

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

(f) Secondary System

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

(g) Steam Generator

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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

In the design, the reactor's inherent reactivity feedbacks have been leveraged for safety and economics in two ways: (i) The systems provide the basis for the reactor to passively self-regulate its power production to match the heat demand from the power conversion cycle without moving control rods (passive load following); (ii) The systems provide 'defense-in-depth' protection to arrest accident progressions before the reactor reaches unsafe conditions, even if the control and safety rods fail to scram.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

The Balance of Plant steam turbine generator system is relied upon for normal shutdown heat removal. The ARC-100 emergency Heat Removal systems consist of the following: (i) Direct Reactor Auxiliary Cooling System (DRACS); (ii) Reactor Vessel Auxiliary (Air) Cooling System (RVACS). The RVACS always operates in a passive state, RVACS removes the reactor's decay heat through the Reactor Vessel and Guard Vessel walls by radiation and convection to naturally circulating air outside the Guard Vessel without exceeding structural temperature limits. The DRACS is composed of three units operating in a natural convection mode. Heat exchanger loops using NaK as an intermediate transfer heat from the cold sodium pool to air heat exchangers in which the hot air vents to the atmosphere. DRACS can be either passive or active using forced air convection

(c) Spent Fuel Cooling Safety Approach / System

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

(d) Containment System

The ARC-100 containment system is a Low-Leakage containment type, where the reactor vessel is designed to operate at near atmospheric pressure. Damage to the core does not directly relate to radioactive releases as the high chemical compatibility between the fission products and the sodium coolant trap radionuclides. The core itself is isolated from the secondary side using an intermediate heat transfer system, which itself limits if not prevents the propagation of nuclides via being pressurized above the primary sodium loop. A Cover Gas System is used to remove radionuclides from the cover gas region. Radiation monitors are installed in the head access area and cover gas service vault to detect any gas leakage. The slightly pressurized argon cover gas is circulated with a filtering stage to eliminate sodium vapor, aerosols, and any other impurities. This monitoring system is also used to survey the reactor cover gas to check for elevated fission gas levels that could indicate fuel failures. The pressurization is used to ensure that no sodium will be released into the environment if a leak occurs. Argon that is treated is reintroduced to reduce consumption.

(e) Chemical Control

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

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control Systems

The fail-safe safety-related shutdown Distributed Control and Information System (DCIS) is a three-division control and monitoring system design, each with separate and independent power supply electrical systems. Divisions are used to support automatic shutdown of the reactor and decay heat removal via DRACS and

RVACS. The system is designed to be able to operate with one division continuously out of service when a design basis event occurs. The DCIS operates at low voltage as the fail-safe shutdown systems are designed to operate without electricity.

8. Plant Layout Arrangement

(a) Reactor Building

The Reactor Building is a cylindrical building made of reinforced concrete floors and walls. Roof trusses and their supporting columns are made of structural steel. The Reactor Building Structure houses the Primary Reactor System, reactor support and safety systems. The refueling floor of the Reactor Auxiliary Building Structure includes the refueling and fuel handling systems and the overhead crane. The design pressure and temperature of the Reactor Building will be established from maximum calculated pressures and temperatures resulting from postulated design basis events including sodium fires. The reactor building and the primary systems including the concrete silo will have a design life of at least 60 years.

(b) Balance of Plant

The steam turbine is a non-reheat, air cooled, single shaft, single casing turbine with separate HP/IP/LP sections. steam turbine with a single flow high pressure turbine and combined intermediate and low-pressure turbine. The Condensate and Feedwater System collects water from the main turbine and auxiliaries after available thermal energy in the water has been extracted, conditions it, and returns it to the steam generator at design temperature and pressure. The turbine generator auxiliary systems provide supportive services to the turbine generator via cooling, sealing, lubricating, and control functions to sustain the operation and assure the maximum efficiency of the turbine generator.

9. Testing Conducted for Design Verification and Validation

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

10. Design and Licensing Status

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

11. Fuel Cycle Approach

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

12. Waste Management and Disposal Plans

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

13. Development Milestones

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

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

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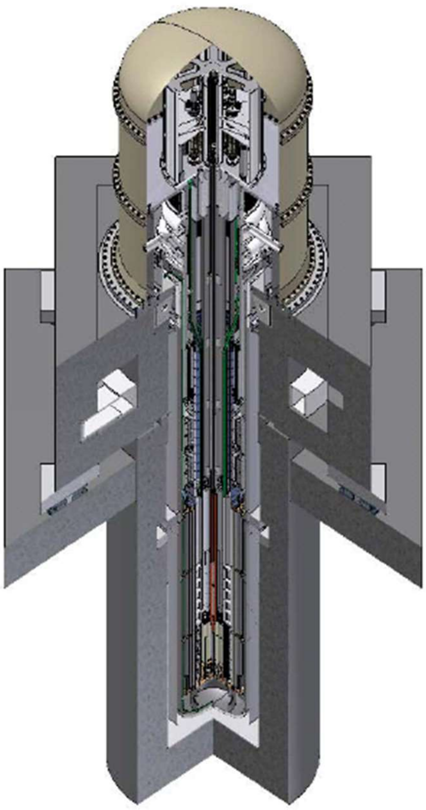


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Toshiba Energy Systems & Solutions Corporation, Japan
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant	Sodium
Thermal/electrical capacity, MW(t)/MW(e)	30 / 10
Primary circulation	Forced convection
NSSS operating pressure (primary/secondary), MPa	Non pressurized
Core inlet/outlet coolant temperature (°C)	355 / 510
Fuel type/assembly array	Metal fuel (U-Zr alloy) enriched uranium
Number of fuel assemblies in the core	18
Fuel enrichment (%)	< 20%
Refuelling cycle (months)	N/A
Core discharge burnup (GWd/ton)	34
Reactivity control mechanism	Axially movable reflectors / fixed absorber
Approach to safety systems	Hybrid passive and active
Design life (years)	60
Plant footprint (m^2)	157 000
RPV height/diameter (m)	24 / 3.5
RPV weight (metric ton)	–
Seismic design (SSE)	Seismic isolator
Fuel cycle requirements/approach	Either one-through scheme or closed fuel scheme is applicable
Distinguishing features	Core lifetime of ~30 years without on-site refuelling, passive walkaway safety
Design status	Detailed design

1. Introduction

The 4S (super-safe, small and simple) is a small sodium-cooled pool-type fast reactor with metal fuel. Being developed as distributed energy source for multi-purpose applications, the 4S offers two outputs of 30 MW(t) or 10 MW(e) and 135 MW(t) or 50 MW(e), respectively. These energy outputs are selected from the demand analyses. The 4S is a non-breeder fast reactor. 4S reactor cores are designed to have a lifetime of 30 years for the 30 MW(t) core and 10 years for the 135 MW(t) core. The plant electric power can be controlled by the water–steam system, which makes the reactor applicable for a load follow operation mode.

2. Target Application

The 4S is designed for electricity supply to remote areas, mining sites as well as for non-electric applications. The plant can be configured to deliver hydrogen and oxygen using the process of high temperature electrolysis.

This process can be performed without producing environmentally disadvantageous by-products, such as carbon dioxide. Two kinds of systems for non-electric applications can be incorporated in the 4S:
- Seawater desalination system: the 50 MW(e) 4S plant can produce fresh water at a rate of 168 000 m^3/day;
- Hydrogen and oxygen production system: hydrogen production rate for the 10 MW(e) and 50 MW(e) 4S is 3000 Nm^3/h and 15 000 Nm^3/h respectively.

Combinations of these systems and the turbine generator system as balance of plant (BOP), including the capacity of each system, would be determined to meet the actual needs at any particular site.

3. Design Philosophy

The 4S reactor is an integral pool type with all the primary components installed inside the reactor vessel (RV). The 4S design is optimized to achieve improvement of public acceptance and safety, minimization of fuel cost and O&M cost, use of uranium fuel with enrichment less than 20%, adequate fuel burn-up and reduction in core size.

4. Main Design Features

(a) Power Conversion

The steam turbine generator is used for converting nuclear power to electricity. The nuclear steam supply system (NSSS) consists of the primary cooling system, the intermediate heat transport system and the water/steam system. The intermediate heat transport system has an EM pump, piping and a steam generator (SG). The SG is a helical coil type with wire-meshed double-wall tube to prevent a sodium-water reaction in the event of the tube failure.

(b) Reactor Core

The core and fuel are designed to eliminate the need for refuelling during approximately 30 years for the 10 MW(e)-4S and to make all reactivity temperature coefficients negative. Metal fuel, which has an excellent thermal conductivity, is applied. The core can be operated by axially moving reflectors installed outside of the core, upward from the bottom. No reloading or shuffling of fuel is required during the whole core lifetime. The fuel element consists of fuel slugs of U-Zr alloy, bonding sodium, cladding tube, and plugs at both ends. Coolant inlet modules located beneath the fuel subassembly provide a lower shielding for the reactor internal structures including the core support plate and air in the reactor vessel auxiliary cooling system (RVACS).

(c) Reactivity Control

The reactivity control during normal operation is by the axial movement of reflectors and using fixed absorbers. The movable reflector surrounding the core gradually moves, compensating the burnup reactivity loss over the 30 years lifetime. Therefore, the reactivity control is unnecessary at the reactor core side and this is an important factor to simplify the reactor operation. The transient overpower is prevented by the limitation of high-speed reactivity insertion by adopting the very low speed driving system.

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

The primary sodium circulates from the EM pumps downward, driven by its pump pressure, and flows through radial shielding assemblies located in the region between the RV and the cylindrical dividing wall. The coolant flow changes its direction at the bottom of the RV and then goes upward, mainly into the fuel subassemblies and partly into the movable reflectors. The coolant flow is distributed appropriately to fuel subassemblies of each type and to the movable reflectors. Here, the core barrel separates the core and the reflector regions. Heat produced in the core is transferred to the coolant while it flows through the fuel pin bundles. The reflectors are also cooled so that the temperature becomes sufficiently low and the temperature distribution is flattened to maintain integrity through the plant life time. The coolant gathers at the hot plenum after flowing through the fuel subassemblies and the reflectors. The heated primary sodium then goes into the IHX to transfer heat to the secondary sodium. During normal operation, the primary system is enclosed inside the RV; sodium coolant is circulated by two EM pump units arranged in series. The heat generated in the reactor is transferred to the secondary sodium via the IHX located at the upper region in the RV. The secondary sodium is circulated by one EM pump unit. The heat is transferred to the water-steam system via heat transfer tubes in the SG. The heated water/steam is circulated by the feedwater pump.

(f) Secondary System

The secondary sodium loop acts as an intermediate heat transport system and consists of the EM pump, piping, dump tank, and the SG. The secondary sodium coolant heated in the IHX flows inside the piping to the SG where heat is transferred to water/steam to be supplied to the steam turbine generator.

(g) Steam Generator

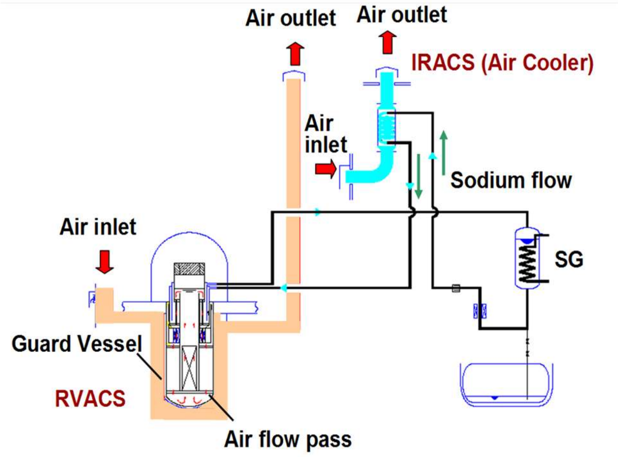
The 4S adopts a once through type double-wall tube SG with failure detection systems. The heat transfer tube of the SG is a double-wall type. Between the inner and outer tube, wire meshes are installed, which are filled with helium, to detect one side tube failure prior to failure of the other side tube.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

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The philosophy of the 4S safety concepts is to put an emphasis on simplicity achieved using passive and inherent safe features as a major part of the defence in depth (DiD) strategy. In addition to the inherent safety features, there are two independent systems for reactor shutdown. The primary shutdown system provides for a drop of several sectors of the reflector, and the back-up shutdown system provides for insertion of the ultimate shutdown rod from a fully out position at the core centre. The reflectors and the shutdown rod are fallen by gravity on scram. Both the reflector and shutdown rod are each capable of enough negative reactivity to shut down the reactor.

(b) Decay Heat Removal System / Reactor Cooling Philosophy



Engineering safety system

The water/steam system is available for normal shutdown heat removal. The decay heat of the core is transferred to water/steam system via the intermediate heat transport system by forced convection and is finally removed from a condenser. For decay heat removal during water steam system is not available upon accidents, two independent passive systems are provided; the RVACS and the intermediate reactor auxiliary cooling system (IRACS). The RVACS is completely passive and removes decay heat from the surfaces of the guard vessel (GV) using natural circulation of air. There is no valve, vane, or damper in the flow path of the air; therefore, the RVACS is always in operation, even when the reactor operates at rated power. Two stacks are provided to obtain a sufficient draft. The IRACS removes decay heat by air cooler which is arranged in series with the secondary sodium loop. Heat is removed by forced sodium and air circulation at the IRACS when electric power is available. In addition, the IRACS can also remove the required amount of heat solely through natural circulation of both air and sodium during loss of power events.

(c) Spent Fuel Cooling Safety Approach / System

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

(d) Containment System

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

(e) Chemical Control

4S does not have chemical control system.

6. Plant Safety and Operational Performances

The 4S is designed to operate safely without active involvement of the plant operators. The design features to support such operation include: (1) burn-up reactivity swing automatically compensated by the fine motion reflectors, (2) no need in reloading and shuffling of fuel in the course of 30 years for the 10 MW(e)-4S, (3) reduction in maintenance requirements achieved by adopting static devices and (4) reduction of in-service inspections achieved by taking advantage of the non-pressurized systems of sodium-cooled reactor and by applying a continuous monitoring process based on leak-before-break detection concept to ensure safety.

7. Instrumentation and Control System

The instrument and control system consists of safety related and non-safety related systems. The safety-related systems include the reactor protection system (RPS), the engineering safety feature actuation system (ESFAS) and the remote shutdown system (RSS). These systems have the safety class 1E instruments. The RPS is plant protection system to initiate reactor trip at abnormal plant operation condition.

8. Plant Layout Arrangement

The plant layout of the 4S is optimized to meet various functional needs; the requirements for safety; radiation zoning, piping and cabling; construction requirements; and access and security considerations.

(a) Reactor Building

The 4S is a land-based nuclear power station with the reactor building embedded underground for security considerations and to enhance protection against extreme external events. The reactor building including the concrete silo can be used for more than 60 years.

(b) Balance of Plant

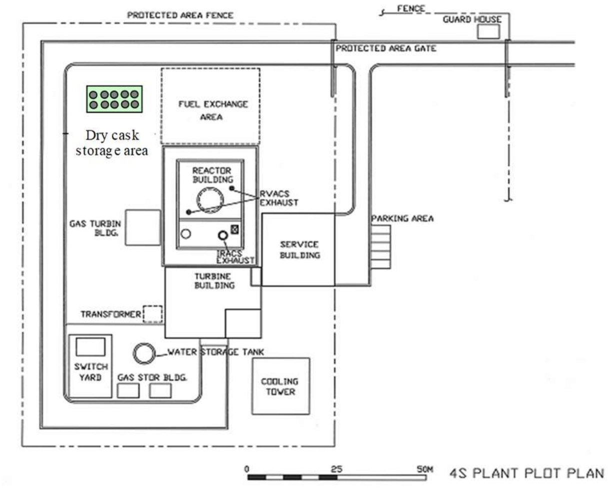
The BOP including a steam turbine system is located at ground level.

i. Turbine Generator Building

The 4S plant consists of one reactor and one turbine generator system. Superheated steam is supplied from the steam generator to the turbine.

ii. Electric Power Systems

These systems include the plant main generator (PMG), the main power transformer and the generator circuit breaker (GCB), diesel power generator and batteries. The grid is also connected to the unit auxiliary transformer (UAT). The two class 1E buses are separated from each other and separated from the non-class 1E electric system. Each class 1E system is provided with a separate emergency diesel generator and batteries.



Plant Layout

9. Testing Conducted for Design Verification and Validation

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

- A critical experiment for the nuclear design method of a reflector-controlled core with metallic fuel.
- The pressure drop for a fuel subassembly has been confirmed by fuel hydraulic testing.
- The scale test of the reflector drive mechanism with fine movement has been performed.
- Heat transfer characteristics between the vessel and the airflow of RVACS have been confirmed.
- The heat transfer characteristics of the steam generator have been confirmed by sodium testing.
- Manufacturability investigations, electromagnetic oscillation testing and sodium flow testing of the EM pump completed.

10. Design and Licensing Status

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

11. Fuel Cycle Approach

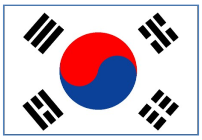
The 4S reactor can be applied to either once-through fuel cycle scheme or closed fuel cycle scheme. It mainly depends on user country's fuel cycle policy. In the case of once-through fuel cycle scheme, spent fuel after 30 years' operation is cooled in the reactor vessel for one year and temporary stored in dry cask for the 10MWe-4S. Then, it is eventually shipped to a permanent repository.

12. Waste Management and Disposal Plan

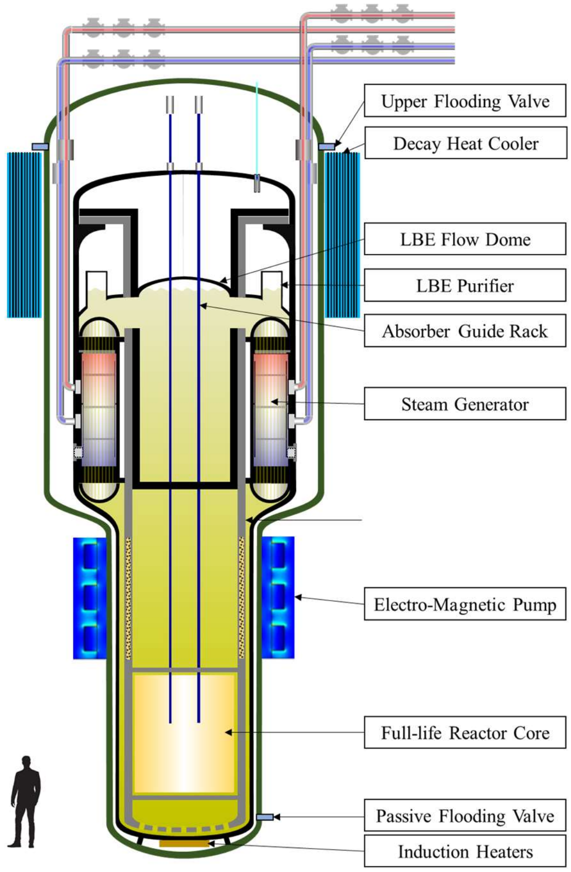
Decommissioning at the end-of-life was evaluated such as sodium deposition and reactor vessel deposition can be done by following decommissioning method of EBR-II and LWR plant in the US. Sodium will be disposed by following experiment of EBR-II. The RV without fuel and sodium will be filled with concrete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

13. Development Milestones

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



MicroURANUS (UNIST, Republic of Korea)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Ulsan National Institute of Science and Technology (UNIST), Republic of Korea
Reactor type	Lead-bismuth cooled Reactor
Coolant/moderator	Pb-Bi (45-55% Wt.%) eutectic alloy
Thermal/electrical capacity, MW(t)/MW(e)	60 / 20
Primary circulation	Electromagnetic Pump (EMP)
NSSS Operating Pressure (primary/secondary), MPa	0.1 / 6~12
Core Inlet/Outlet Coolant Temperature (°C)	250 / 350
Fuel type/assembly array	UO_2 / Monolithic Inverted Core
Number of fuel assemblies in the core	85
Fuel enrichment (%)	3 radial zones; 8, 10, 12
Core Discharge Burnup (GWd/ton)	60
Refuelling Cycle (effective full power years)	No refuelling (30)
Reactivity control mechanism	Control rods and shutdown rods
Approach to safety systems	Passive
Design life (years)	40
Plant footprint (m^2)	N/A
RV height/diameter (m)	11.0 / 2.4
RV weight (metric ton)	200
Seismic Design (SSE)	3D seismic isolation system
Fuel cycle requirements / Approach	No HLW/PyroGreen Recycling
Distinguishing features	Ex-vessel emergency water cooling system
Design status	Conceptual design

1. Introduction

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

2. Target Application

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

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

MicroURANUS has evolved from URANUS-40 design with hexagonal-lattice using LEU oxide fuels. The total 85 pellet assemblies, 13 control assemblies and 6 ultimate shutdown assemblies. Three enrichment zones are used to reduce the radial power peaking and to increase internal breeding gain correspondingly small reactivity swing. The fissile enrichment adopted for the inner enrichment zone is lower than that of the outer fuel zone. The MicroURANUS core has an active height of 2.0 m and equivalent diameter of 1.6 m that generates 60 MW(t) with the average coolant temperature rise of 100ºC from the core inlet to the core exit.

(c) Reactivity Control

The MicroURANUS has a very small reactivity swing over thirty effective full power years, requiring almost no control rod movements if operated steadily at full power. Reactivity can be controlled by using 13 assemblies for versatile navigation of merchant ships. Rapid power increases can be tolerated by using PCI-resistant fuels. To meet regulatory requirements for the reactor module transportation before and after commercial operations, additional reactivity suppression is made by using the ultimate shutdown assembly.

(d) Reactor Vessel and Internals

The reactor vessel of MicroURANUS is approximately 11.0 m in height and 2.4 m in diameter. The reactor vessel is encased in the containment vessel with an annular space used for in-service inspections and maintenance. The reactor barrel is a cylindrical shell that provides both structural supports for reactor core and internals while serving as LBE flow divider, neutron reflector and thermal insulator.

(e) Reactor Coolant System

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

(f) Secondary System

Currently a comparative study is in progress for the secondary systems, between traditional superheated steam cycle or supercritical CO_2 cycle. Criteria for the down-selection process include technical maturity, transient responses, system footprint as well as maintainability.

(g) Steam Generator

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

Reactor scram systems can be actively actuated with independent emergency power supplies. During normal shutdown and reactor trip conditions, decay heat is removed by the steam dump through atmospheric dump valve(ADV) of the secondary steam system when traditional steam system is chosen as the secondary system. On the other hand, the critical CO_2 system is chosen, decay heat is removed by valves the reactor auxiliary cooling system that circulates coolant to transport heat from the reactor outer wall to external coolers. Under the beyond design base accident conditions, a dedicated ex-vessel emergency water cooling system is passively activated to remove any heat beyond the capacity of the reactor auxiliary cooling system. If the reactor vessel is heated above limits, the annular space between reactor vessel and containment vessel is flooded with shielding water through valves. Steam produced from the water-cooling of the external surface of reactor vessel goes to dedicated coolers where the steam is condensed into water before returning to the shield water tank by gravity.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

During normal shutdown and reactor trip conditions, decay heat is removed by the steam dump through the atmospheric dump valve of the secondary steam system when traditional steam system is chosen as the secondary system. On the other hand, the critical CO_2 system is chosen, decay heat is removed by valves the reactor auxiliary cooling system that circulates coolant to transport heat from the reactor outer wall to external coolers.

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

(c) Emergency Core Cooling System

Under the design base accidents, the passive reactor auxiliary cooling system in combination with heat exchangers submerged in the reactor building pool(shield tank) removes the decay heat from the core, in addition to this, MicroURANUS has another option that the active AFWS in combination with ADV can remove the decay heat from the core by dumping steam through ADV.

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

(d) Containment System

The containment vessel and vacuum system forms the containment system. The containment system enclosed by the reactor building pool that is also functioning as the shield water tank. This containment system is physically protected by external shield from external attacks including aircraft crashes. The containment vacuum system is replacing the complicated insulation of the conventional nuclear power plants and can also be used to detect any leakage from the RCS utilizing the radiation detectors in the vacuum system in a short period. The containment system also includes a filtered vent system for venting gas and/or steam when the pressure in the containment increase the pressure limit under the accident conditions.

6. Plant Safety and Operational Performances

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

7. Instrumentation and Control Systems

The conceptual design of digital hybrid instrumentation and control systems with analogue backup system will be completed by the end of 2022.

8. Plant Layout Arrangement

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

9. Design and Licensing Status

A conceptual design of MicroURANUS is to be completed in 2022. Currently, the activities for Front End Engineering Design (FEED) are carried out for design, optimization, modelling and experimental validations. Pre-application engagement with the regulatory body will be expected to start in 2025.

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

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

12. Development Milestones

2023 ~ 2024	Engineering design and license document preparation
2025 ~ 2027	Pre-application review and finalize the design
2028 ~ 2030	Application of the Design Certification and the construction permit of the prototype reactor for maritime use of MicroURANUS

LFR-AS-200
(*new*cleo srl, Italy)

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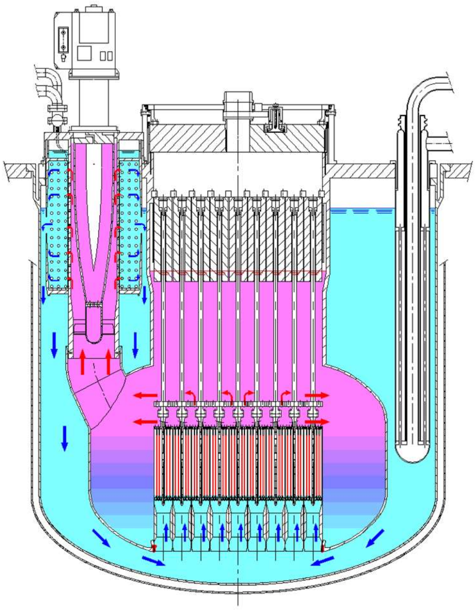
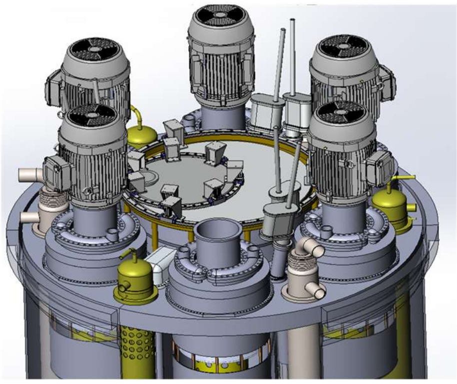




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*new*cleo srl, Italy originally developed by Hydromine S.àr.l., Luxembourg
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant	Lead
Thermal/electrical capacity, MW(t)/MW(e)	480 / 200
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	0.1 / 18 MPa(abs)
Core inlet/outlet coolant temperature (ºC)	420 / 530
Fuel type/assembly array	MOX / hexagonal
Number of fuel assemblies in the core	61
Fuel enrichment (%)	19% avg / 23,2% max in Pu
Refuelling cycle (months)	16 months (1/5)
Core discharge burnup (GWd/ton)	100 GWd/ton
Reactivity control mechanism	Ex-core, reversed-flag type B_4C rods, rotating B_4C Rods
Approach to safety systems	Active + passive
Design life (years)	60
Plant footprint (m^2)	1 100, nuclear island with fuel storage
RPV height/diameter (m)	6.2 / 6
RPV weight (metric ton)	62
Seismic design (SSE)	0.3 g
Fuel cycle requirements/approach	Stockpiled MOX from reprocessing
Distinguishing features	Active + passive walkaway safety; No intermediate loops; Simple, compact primary system: ≤ 1 m^3/MWe; Compact reactor building.
Design status	Conceptual design

1. Introduction

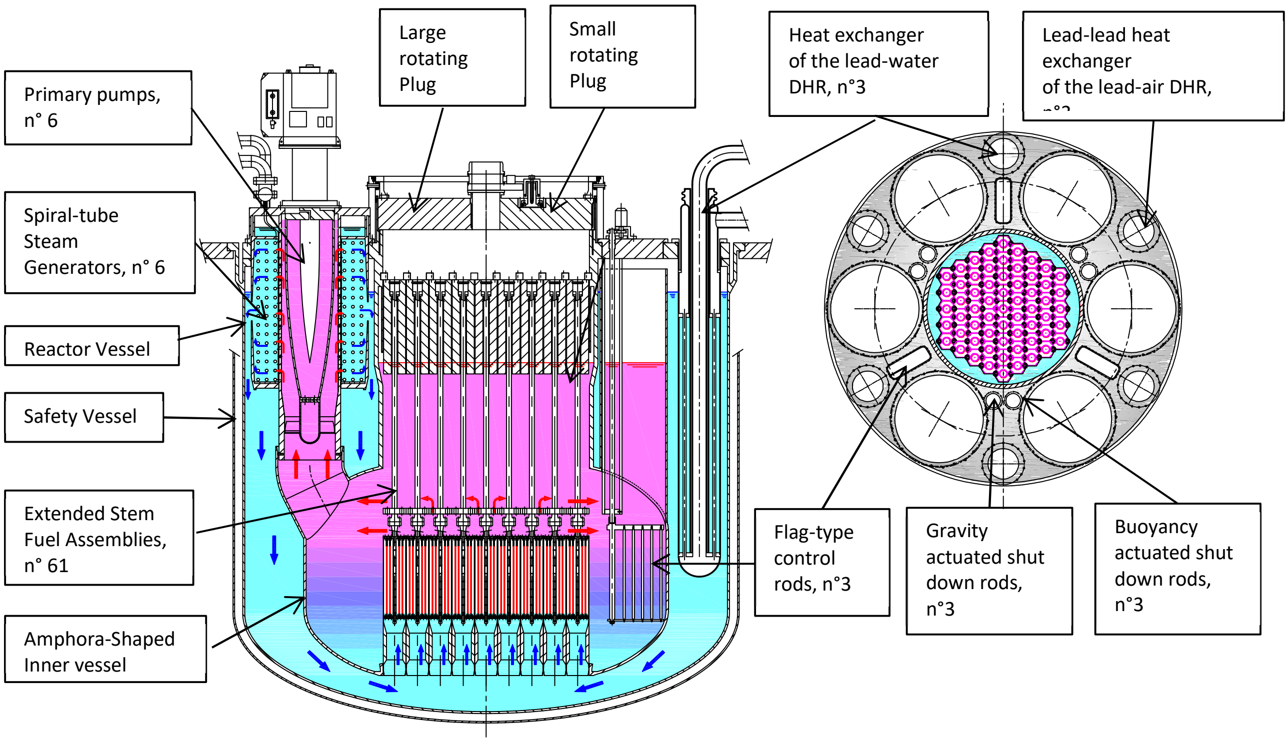
The LFR-AS-200 is an innovative reactor cooled by molten lead; AS stands for Amphora-Shaped, referring to the shape of the inner vessel and 200 is the electrical power in MW. The embodied innovations exploit the lead properties and enhance the potential for future deployment, owing to plant simplification and compactness, while behaving passively safe.

2. Target Application

The absence of intermediate loops, the primary system specific volume of less than 1 m^3/MWe and the compact reactor building are key factors for competitive cost per kWh. Market application is energy production with use of stockpiled Pu and perspective recycle of minor actinides without burden of long-lived transuranics in the waste. The breeding ratio is 0.9 without blanket assemblies and can be reduced with core design adaptation where required.

3. Design Philosophy

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



Scheme of the LFR-AS-200 and peculiar features

4. Main Design Features

(a) Power Conversion

Superheated water steam is generated in once-through steam generators partially immersed in primary coolant.

(b) Reactor Core

The core consists of 61 wrapped, hexagonal FAs, each containing 390 fuel pins laid out on triangular pitch.
Power shaping or flattening has been achieved through the use of zones with three different levels of Pu-enrichment. The FAs, the weight of which is supported by buoyancy, present stems extended up to above the free level, i.e. in gas space. Their heads can be interconnected, and the outer heads fixed also to the section of ASIV facing the cover gas, by means of cams, which are integral part of each head. The result is a self-sustaining core anchored to the inner profile of the ASIV, that acts as the core barrel in gas space.
The stem of the FAs is peculiar to this novel design, because the FAs heads are directly accessible for handling with an ex-vessel refuelling machine operating in gas space under visual control in conjunction with rotating plugs of classical design. The FA foot is free from mechanical supports (no core grid of classical design such as the Diagrid of SPX1) except for the radial touching with adjacent FAs, which are brought into contact to create a packed bundle, bounded on the lower end by the contoured edge of the bottom port of the ASIV.

(c) Reactivity Control

Core reactivity is controlled in normal operation by ex-core rods, installed in the lead pool between the core and the ASIV. Buoyancy-driven and gravity driven rods are used as shutdown systems. Ultimate reactor antireactivity insertion is ensured by core expanders placed on the FAs stems. In case of an ULOOP accident the strong negative feedback due to radial core expansion – magnified by the expansors placed atop the active region – quickly reduces the core power to a level that is comparable with the power of the DHR system.

(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

The LFR-AS-200 is a pool-type reactor that uses forced convection by 6 pumps for the lead coolant primary circulation. The vertical axial-flow pump is integrated inside the SG; the pump rests on, and is connected to, the upper support plate of the SG by means of a flange which closes the pump's shaft penetration through the reactor roof and supports the variable-speed electric motor of the pump. The pump is characterized by a short, large diameter, tapered hollow shaft containing lead brought in rotation by the shaft itself, in order to increase the mechanical inertia of the pump. There are no in-lead pump bearings.

(f) Secondary System

The secondary system is based on the Rankine cycle with superheated steam. The only peculiarity is related to the use of lead as the coolant in the primary system. To avoid lead freezing inside the SG, a prudential high SG inlet temperature is set to 340ºC; this temperature is higher than the melting point of Pb, at 327ºC.

The turbine is made of one high pressure stage and two low pressure stages, with a deaerator fed by steam from the outlet of the high-pressure stage. The turbo-generator set operates at 3000 rpm. A hot water storage is provided in order to reduce the amount of steam bled from the low-pressure turbine and allow a temporary operation at 110% Pn.

(g) Steam Generator

The Spiral-Tube Steam Generator (STSG) is an innovative SG conceived for compactness and because it offers several advantages in term of reactor cost, safety and reactor operability and simplicity of the lead flow path (Table 1). The SG tube bundle, partially raised above the lead-free level of the cold collector, is composed of a stack of spiral-wound tubes. The inlet and outlet ends of each tube are connected to the feed water header and steam header, respectively, both arranged above the reactor roof to eliminate, in case of their failure, the risk of large water/steam release inside the reactor vessel. The tube spirals, one spiral for each tube, are arranged one above the other and equally spaced. The SG is fed from the bottom. Hot lead flows radially through the perforated inner shell and, once past the tube spirals, flows into the cold collector through a circumferential window located just below the lead's free-level.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Lead has excellent cooling properties and its nuclear properties (i.e., its low tendency to absorb neutrons or to slow them down) enable it to readily sustain the high neutron energies needed in a fast reactor, while offering the reactor designer great flexibility. Lead has a very high boiling point, namely 1737 °C. As a result, the problem of boiling coolant is for all practical purposes eliminated. As a coolant operating at atmospheric pressure, the loss of coolant accident (LOCA) can be virtually eliminated by use of an appropriately designed guard vessel. The LFR-AS-200 exploits lead properties for actuation of passive shutdown and passive decay heat removal systems, which do not need power sources, operator intervention and logics. Compatibility of lead with air and water allows diversification of the heat sinks: stored water and atmospheric air.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

DHR is performed by means of two diverse, redundant systems, each consisting of three identical loops, each loop rated 2.5 MW. Two loops are sufficient to remove the decay heat. The loops of the first system are filled with lead. Each loop consists of a lead-lead dip cooler and of a lead-air cooler with interconnecting piping, and is passively operated and also passively actuated thanks to the thermal expansion of the cold branch of the loop, which actuates the louvers of the air cooler when its temperature exceeds 400°C.

(c) Spent Fuel Cooling Safety Approach / System

The plant is provided a handling flask to transfer the spent fuel from the reactor to the spent fuel water pool. It is equipped with cooling systems to allow FA cooling during transfer. Cooling is performed in argon forced circulation. Argon natural circulation will back up forced circulation in case of loss of power.

Spent fuel in "dry" conditions is under consideration.

(d) Containment System

The secondary confinement is provided with a concrete containment, external-missile-proof. The dimension of the containment is kept small because of the low potential energy stored in the coolant (which operates at atmospheric pressure) and of the small inventory of water/steam of the secondary circuit. Moreover, a safety vessel eliminates any loss of coolant accident (LOCA), in the event of a failure of the reactor vessel. The "practical elimination" of "core melting" will reduce the design requirements on containment, in particular in term of cooling systems

(e) Chemical Control

An oxygen control system is implemented to avoid the formation of lead oxides and at the same time the preserve the passivation of steel structures and components. Several systems, already successfully tested on experimental loops, are under investigation for implementation in LFR-AS-200.

6. Plant Safety and Operational Performances

The safety of the LFR-AS-200 is based on the properties of lead and on the specific design features. One of the most important characteristics of lead as a coolant is its chemical inertness. Lead is a coolant that does not undergo violent chemical reactions, which could possibly lead to high energy release in the event of accident conditions. The LFR-AS-200 is designed to operate safely in priority reactor mode i.e. at constant power in the range 20% and full power. Possibility of reactor load following mode is being investigated, but not implemented. Load following mode in the range -10% +10% of Pn is possible, by means of adjusting the amount of spilled steam from the low-pressure body of the turbine and using stored hot water as balance heat sink and source. The reactor will operate with an almost stable ΔTcore in the operating range; this imposes the operation of the MPs at variable speed. Core inlet temperature is maintained constant by control of the feedwater flow rate. Steam temperature is also maintained constant by the control rods.

7. Instrumentation and Control System

Signals from in-core instrumentation could be made available to both control and surveillance systems. Moreover in-core instrumentation will remain mostly operational even during refuelling.

8. Plant Layout Arrangement

The reactor building, the spent fuel building and the new fuel building are located on a common basement. The control room is located above the fuel building. The reactor building extends approximately from 9 m below grade up to 18 m above grade. The turbine generator building is located at ground level. In addition to the single-module configuration, two additional arrangements are studied at conceptual level: - a two-modules configuration with a common turbine generator of 400 MW(e) and - a four-modules configuration with a common turbine generator of 800 MW(e). A common basement for reactor buildings and fuel buildings is foreseen also in case of multi modules configuration. Economics is expected at turbine generator level but also by the reduction of the number of spent fuel and new fuel buildings.

9. Testing Conducted for Design Verification and Validation

Testing phase not started, yet. A substantial amount of experimental assessments related to LFR-AS-200 technology will be made for LFR-AS-30 development.

10. Design and Licensing Status

Licensing process has not started, yet. Newcleo is investing in the design of a smaller scale reactor (LFR-AS-30) as a FOAK demonstrator, whose licensing is intended to ease the safety assessment of LFR-AS-200.

11. Fuel Cycle Approach

Being a fast reactor, the core of the LFR-AS-200 can be nearly self-sufficient in Pu (breeding ratio~0,9) or, with some modifications, can also be transformed in Pu burner (breeding ratio~0,5). A Pu breeder cycle is not a main objective of the project, given the surfeit of Pu available worldwide.

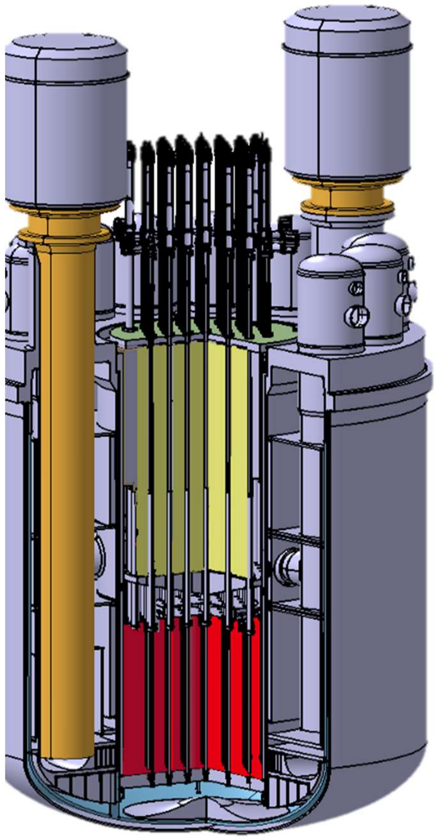
12. Waste Management and Disposal Plan

Uranium, plutonium and minor actinides will be recycled in the LFR-AS-200. Residual fuel waste will be transported to geological repositories.

13. Development Milestones

2014	Completion of pre-conceptual design (by Hydromine Nuclear Energy S.àR.L)
2019	Completion of the conceptual design
2021	Hydromine Nuclear Energy S.àR.L. is incorporated in *new*cleo for a fast development program. Design restarted after an idle phase

SVBR (JSC AKME Engineering, Russian Federation)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	JSC Institute for Physics and Power Engineering and JSC EDB Gidropress, Russian Federation
Reactor type	Liquid metal cooled fast reactor
Coolant/moderator	Lead-bismuth eutectic alloy
Thermal/electrical capacity, MW(t)/MW(e)	280 / 100
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Low pressure
Core Inlet/Outlet Coolant Temperature (°C)	340 / 485
Fuel type/assembly array	UO_2 / hexagonal
Number of fuel assemblies in the core	61
Fuel enrichment (%)	< 19.3
Core Discharge Burnup (GWd/ton)	60 (average)
Refuelling Cycle (years)	7 – 8
Reactivity control mechanism	Control rod drive mechanism
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	150 000
RPV height/diameter (m)	8.2 / 4.53
RPV weight (metric ton)	280, without core and coolant
Seismic Design (SSE)	0.5g
Fuel Cycle Approach/Requirements	On the first stage will use mastered UO_2 fuel with postponed reprocessing. In the more distant future transition to closed fuel cycle with self-supplying mode
Distinguishing features	Integral monoblock primary circuit where reactor, steam generators, pumps are installed in one vessel
Design status	Detailed design for potential construction in 2025

1. Introduction

The SVBR-100 is a multipurpose small modular fast reactor lead–bismuth (LBE) cooled with an equivalent electric power of 100 MW. In the Russian Federation, lead–bismuth cooled reactor technology has been used in several nuclear submarines (NSs). The SVBR technology, according to its basic parameters and salient technical characteristics, is claimed as a Generation IV nuclear reactor. The development of SVBR-100 is based on the experience gathered in the design and operation of several LBE facilities on NSs, which allows:

- Use of mastered LBE technology;
- Use of almost all basic components, units and equipment devices of the reactor installation, which are verified by operational experience in LBE;
- Capability to master primary and secondary circuits;
- Use of existing fuel infrastructure;
- Ensuring the corrosion resistance of structural materials;
- Controlling the LBE quality and the mass transfer processes in the reactor circuit;
- Ensuring the radiation safety of personnel carrying out work with equipment contaminated with the ^{210}Po radionuclide; and
- Multiple LBE freezing and unfreezing in the reactor facility.

2. Target Application

The possibility of multi-purpose application of modular nuclear power plants (NPP) of different capacities (100 – 600 MW(e)) based on SVBR-100 creates the conditions to satisfy the requirements of consumers in a new sector of regional and small-scale atomic energy industry: 1) creation of regional NPP and nuclear co-generation plant (NCGP) of low and medium capacity, 2) utilization as part of floating NPPs, 3) renovation of NPP units. The standard reactor modules of 100 MW(e) can be used for multipurpose, e.g.:

- Modular NPP of small, medium or large power;
- Regional nuclear heating and electricity generating plant of 200-600 MW(e) which are located not far from the cities;
- Refurbish the NPPs with expired reactors' lifetime; and
- Nuclear desalination systems.

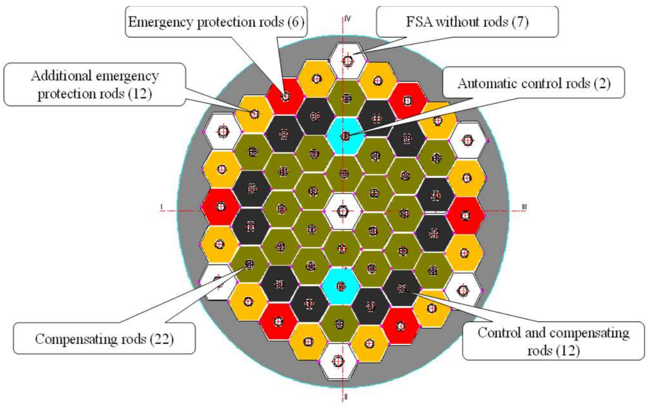
3. Design Philosophy

SVBR-100 is designed as a multipurpose modular integral lead-bismuth cooled small power fast reactor to generate an equivalent electricity of 100 MW(e). The design is based on more than 80 reactor-years operational experience of LBE cooled reactors for submarine propulsion application. Its main features include:

- Enhanced inherent self-protection and passive safety and significant simplification of the design of the reactor as well as entire NPP;
- Possibility to operate with different type of fuel in different fuel cycles (period of operation without refuelling: not less than 7-8 years);
- Compact design and maximum factory readiness of the reactor and its transportability, include railway;
- Possibility of creation of module based structured NPP with power multiplying by adding the reactors.

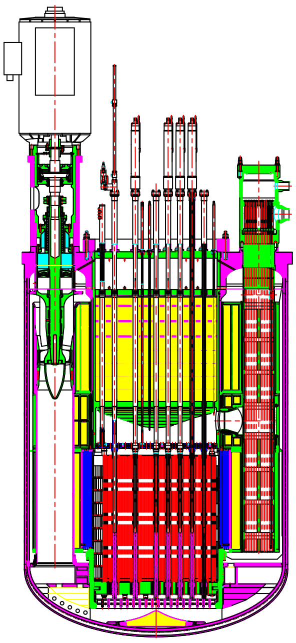
4. Main Design Features

(a) Reactor Core



The SVBR-100 reactor core operates without any partial refuelling. The fresh fuel is loaded as a single cartridge while the spent nuclear fuel is unloaded cassette by cassette. The core configuration allows for a lower power density compared with the nuclear submarines using LBE reactors. This design has the capability to utilize various fuel cycles. The first stage will be the typical uranium oxide fuel leading to a core breeding ratio (CBR) of 0.83; MOX fuel can also be used, leading to a CBR just about 1, which provides fuel self-sufficient mode in the closed fuel cycle. Using UO_2 as the starting fuel, the closed fuel cycle can be realized in 15 years. Nitride uranium and uranium plutonium fuel can also be used to improve safety and fuel cycle characteristics. The SVBR-100 reactor pursues resistance to nuclear fissile material proliferation by using uranium with enrichment below 20% while using uranium oxide fuel in the initial core. The reactor is designed to operate for eight years without core refuelling.

(b) Reactor Coolant System



The entire primary equipment circuit of SVBR-100 is contained within a robust single reactor vessel. A protective casing surrounds the single-unit reactor vessel. The reactor transfers generated heat into a two-circuit heat-removal system and SG with forced multiple-circulation secondary coolant system. Natural circulation of coolant in the reactor heat removal circuits is sufficient to passively cool down the reactor and prevent superheating of the core. The coolant system includes mass-exchangers, gas mixture ejectors, sensors of oxygen activity in LBE; its function is to maintain the LBE quality, inhibiting structural materials corrosion. The circuits for primary coolant circulation (the main and the auxiliary one), are entirely realized by components of the in-vessel components, without using pipelines and valves. Within the main circulation circuit (MCC), the coolant flows according to the following scheme. Being heated in the core, the coolant flows to the inlet of the medium part of the inter-tube chamber of twelve SG modules connected in parallel to each other. Then coolant is divided into two flows. One flow moves upwards in the inter-tube chamber and enters the peripheral buffer chamber with a free surface level of the "cold" coolant. Another flow moves downwards and enters the outlet chamber out of which it goes to the channels into in-vessel radiation shielding. Coolant flows upwards through in-vessel radiation shielding and cools it, and then it enters the peripheral buffer chamber as well. Out of the peripheral buffer chamber, the coolant flows over the downcomer circular channel along the RMB vessel via the inlet chamber to the MCP suction. Out of the MCP the coolant flows over the two channels installed in the mono-block of the lower zone of in-vessel radiation shielding into the distributing chamber, from which main part of flow goes to the reactor inlet chamber, thus closing the MCC circuit. Very small part of the coolant moves upwards via the gap near RMB vessel wall, cooling it and goes into the peripheral buffer chamber.

(c) Secondary System

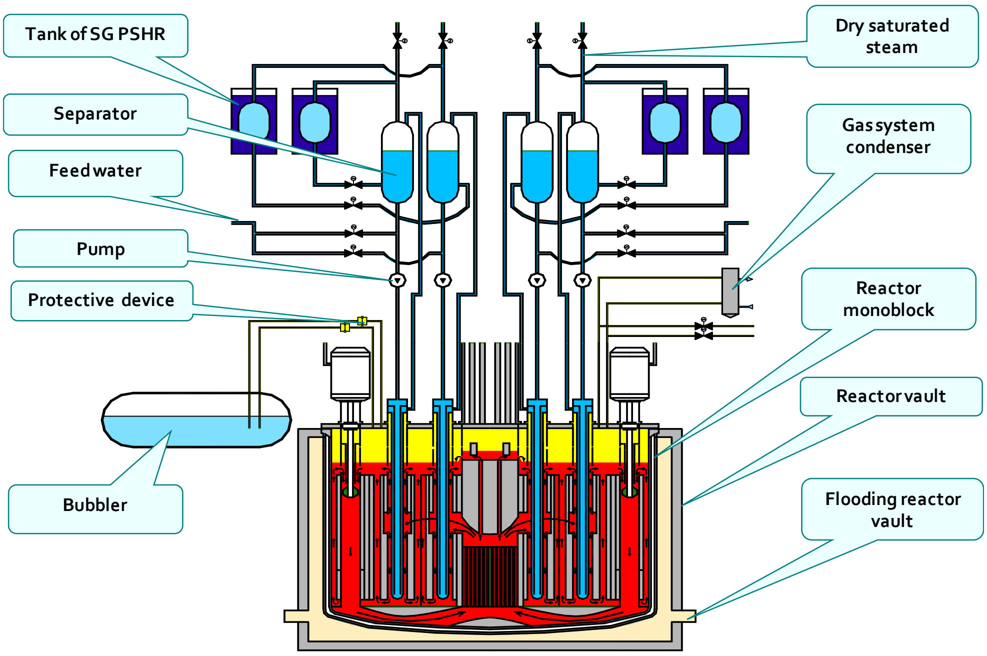
The secondary system includes: SG modules, feedwater and steam pipelines, separators and autonomous cooling condensers. The basic equipment of SVBR-100 is installed in an 11.5 m high tight box-containment. In the lower part of each box, there is a concrete well to be flooded by water in beyond the design accidents that involve failures of all four PHRS via SGs. The reactor monoblock is installed inside concrete well and is fastened on the head ring of roof. In the upper part of the box there is the reactor equipment, including four steam separators and four cooling condensers immersed in the water tanks PHRS. The high elevation of the separators has been selected in order to guarantee the coolant natural circulation in the secondary circuit in cool down mode. The gas system condensers are installed in the upper part of the box in the separate concrete compartment.

5. Safety Features

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

- First, this is potential energy contained in coolant. At atmospheric pressure, LBE does not store potential energy, which in an event of accident occurrence can cause destruction of defence barriers, core damage and disastrous release of radioactivity. For other reactor types and coolants, there are potential energy of coolant compression and potential chemical energy of coolant's interaction with structural materials (zirconium) (for water coolant), and with water and air (for sodium coolant);
- Potential energy is a natural property of coolant and cannot be changed by any technical solutions;
- Further, it is the integral structure of the reactor facility that completely eliminates pipelines and valves with radioactive coolant and eliminates the possibility of coolant leak;
- Finally, this is a fast neutron reactor, in which there are no poisoning effects, low burnup reactivity margin, low value of negative temperature reactivity effect, and negative void reactivity effect. Efficiency of the strongest absorbing rod does not exceed 0.5$, that being coupled with technical performance of the control and protection system exclude an opportunity of prompt neutrons criticality in the reactor;
- Elimination of radioactivity release into the environment is insured by the system of disposed defence-in-depth barriers.

Those type RFs assure their high resistance not only in events of single failures of the equipment and personnel errors but in events of intentional malicious actions when all special safety systems operating in a standby mode can be intentionally disabled. At LBE cooled reactors such catastrophic accidents as Chernobyl or Fukushima disasters as well as fires similar to that occurred at Monju NPP are physically impossible or can be easily localized with a purpose to prevent population's exposure to irradiation beyond the NPP site (LOHS type accidents). This is extremely viable for radiophobia elimination and realization of NPP construction in developing countries where the level of terroristic threat is high.



6. Instrumentation and Control Systems

The main principles of the design include:

- Distributed I&C system with several level of hierarchy and defence in depth;
- Soft control of NPP technological systems;
- Availability of the large screen and reserve zone at main control room (MCR);
- Principles of diversity, reliability, physical separation and others, providing high level of functional reliability, including protection against common cause failures;
- Well-developed diagnostic functions; and
- Self-diagnostic of I&C programmable devices.

Design specificities are:

- Control of coolant flow rate by changing rotation speed of MCP depending on reactor power for maintaining constant coolant heat up;
- Full scope diagnostic system of NPP;
- Providing I&C operability during 7-8 years of continuous NPP operation; and
- New tasks of neutron flux monitoring while core refuelling;
- Highly reliable reactor control at start up and operation;
- Load-follow operation in the deep range (100–50–100%).

7. Plant Layout Arrangement

General view of pilot plant is shown in Figure.

8. Design and Licensing Status

The Rosatom Scientific and Technical Council convened on 15 June 2006 approved the development of the technical design of experimental industrial power unit based on the SVBR-100. Siting licence is received to current time at Dimitrovgrad, in the region of Ulyanovsk. Key reactor and reactor core research and development works have begun.

9. Fuel Cycle Approach

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

10. Waste Management and Disposal Plan

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

11. Economic Characteristics

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

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

12. Development Milestones

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

SEALER-55 (LeadCold, Sweden)

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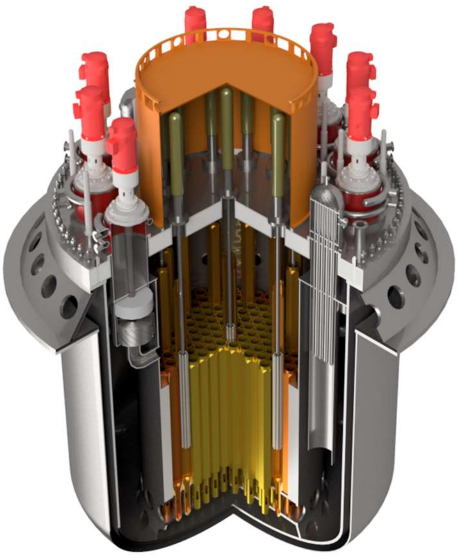


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	LeadCold, Sweden
Reactor type	Lead-cooled SMR
Coolant	Lead
Thermal/electrical capacity, MW(t)/MW(e)	140 MWt / 55 MWe
Primary circulation	Forced
NSSS operating pressure (primary/secondary), MPa	Non-pressurised
Core inlet/outlet coolant temperature (°C)	420 / 550
Fuel type/assembly array	UN/Hex-can
Number of fuel assemblies in the core	84
Fuel enrichment (%)	12
Refuelling cycle (months)	300
Core discharge burnup (GWd/ton)	60
Reactivity control mechanism	B_4C, WB_2
Approach to safety systems	Passive
Design life (years)	28
Plant footprint (m^2)	600 / 20 000 (single unit/fence)
RPV height/diameter (m)	5.5 / 4.8
RPV weight (metric ton)	20
Seismic design (SSE)	None
Fuel cycle requirements / approach	TBD
Distinguishing features	Nuclear battery
Design status	Conceptual design

1. Introduction

SEALER-55 is designed by LeadCold for on-grid, commercial power production. It is intended to be fabricated in series of more than 100 units in an automated factory.

2. Target Application

The primary market for SEALER-55 is existing nuclear power sites, where several units can be clustered to replace the capacity of existing light water reactors. Other markets to be considered are large consumers of high temperature steam, such as hydrogen and bio-oil/bio-char plants. Remote and marine applications (mining, shipping) may also be considered.

3. Design Philosophy

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

- Elimination of on-site fuel cycle operations, by designing a long-life core
- 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) Power Conversion

Rankin cycle.

(b) Reactor Core

The core of SEALER-55 features 84 fuel assemblies containing 169 UN or (U,Hf)N rods, two B4C control rod assemblies, ten shut-down rod assemblies using WB2/B4C absorbers and 66 ZrO2 reflector assemblies. The active height is 1.3 m. The rod diameter, pitch and channel length have been optimised using an analytical approach published in Annals of Nuclear Energy.

(c) Reactivity Control

Reactivity control is accomplished by two B4C control rod assemblies and ten WB2 shut-down rod assemblies. The latter are inserted passively by gravity.

(d) Reactor Pressure Vessel and Internals

The primary vessel is manufactured from SS316 with an overlay weld of alumina forming austenite (AFA), for corrosion protection. The wall thickness is 40 mm, diameter 4.8 m, and height of the cylindrical section 4.3 meters.

(e) Reactor Coolant System

During normal operation conditions, forced circulation of the lead coolant is provided by variable speed reactor coolant pumps. Eight steam generators transfer the heat to the secondary system.

(f) Secondary System

Rankine steam cycle at 165 bar and T = 350-530°C.

(g) Steam Generator

Spiral tube, where the tube is a compound of Alloy 690 on the steam side and Fe-10Cr-4Al on the primary side.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Ten shut-down rod assemblies are parked above the core during nominal operation. These are divided into two diversified banks, each being capable of taking the core sub-critical by more than 1$. The first consists of high density WB2 absorber pellet rods (96% B-10 enriched), which will be inserted by gravity. The second bank consists of 96% B-10 enriched B4C pellets, which will be inserted by a spring-load mechanism.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

In case the secondary system is unavailable for decay heat removal, dip-coolers will be passively actuated, permitting removal of decay heat by gravitationally assisted insertion of water at room temperature and ambient pressure for a period of at least 200 hours. The dip-coolers are designed to ensure that primary system temperatures remain such that restart of the reactor can take place after inspection. In case that dip-coolers fail to actuate, 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. However, primary vessel creep may occur, and restart of the reactors is not foreseen.

(c) Spent Fuel Cooling Safety Approach / System

The fuel resides in the primary system during the entire life of the plant. Following shut-down of the plant, the fuel will cool in-situ until the primary system is removed from site.

(d) Containment System

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

(e) Chemical Control

An oxygen control system consisting of oxygen sensors and oxygen pumps is ensuring that the concentration of oxygen in the primary coolant is such that lead oxide precipitation will not occur, and that a sufficient amount of oxygen is available for self-healing of any damage that may be inflicted on alumina forming steel surfaces. All surfaces exposed to lead consist of alumina forming steels (ferritic, austenitic or martensitic) which have been proven to be highly corrosion tolerant.

6. Plant Safety and Operational Performances

SEALER-55 is designed to provide a passively safe, secure and reliable power source on-grid or industrial site applications. The reactor is able to produce 55 MW of electric power for 25 full power years without reloading nor reshuffling of its UN fuel. A capacity factor of 95% is foreseen, permitting preventive maintenance exchange of pump and steam generator modules with a frequency of 1/year. No fuel reload is foreseen for the entire life of the reactor (25 full power years). The application of alumina alloyed steels provides corrosion protection that is deemed sufficient over the life of the reactor. Safety analysis shows that as designed, SEALER can survive unprotected loss of flow, loss of heat sink and transient overpower accidents with no consequences for fuel and clad integrity. Moreover, the source term is sufficiently small that a full release of volatile fission products into the coolant at the End of Life (EOL) does not require permanent relocation from housing residing beyond 1.0 km from the point of release.

7. Instrumentation and Control System

The primary system is instrumented in order to measure neutron flux, temperature, oxygen concentration and hot leg lead free level position.

8. Plant Layout Arrangement

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

9. Testing Conducted for Design Verification and Validation

An electrically heated prototype of SEALER-55 will be built and operated in Oskarshamn, Sweden for the purpose of validating its safety concept, operational and maintenance procedures, as well as materials performance. The prototype is designed with a power of 3 MW, produced in 7 heated rod assemblies with 37 rods each. The height of the prototype is 1:1 with respect to the SEALER-55, in order to validate residual heat removal capability by dip-coolers, whereas the diameter of the vessel is scaled by 1:2.5. The prototype is currently under engineering design, with the intent to have it in operation by 2024.

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 a combination of IAEA guidelines for research reactors, and existing regulation for light water reactors, when the latter might be applicable. Moreover, the Swedish government has given the task to SSM to develop a framework for licensing of SMRs.

11. Fuel Cycle Approach

The uranium nitride fuel of SEALER-55 will be fabricated by direct ammonolysis of 12% enriched UF_6, followed by spark plasma sintering of pellets.

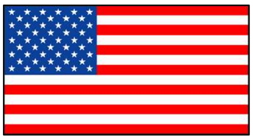
The spent fuel of SEALER-55 reactors may either by disposed in a geological repository, or recycled. In the former case, disposal of entire cores in frozen lead, conversion of irradiated UN to UO_2 pellets or direct disposal of UN fuel rod assemblies will be evaluated.

12. Waste Management and Disposal Plan

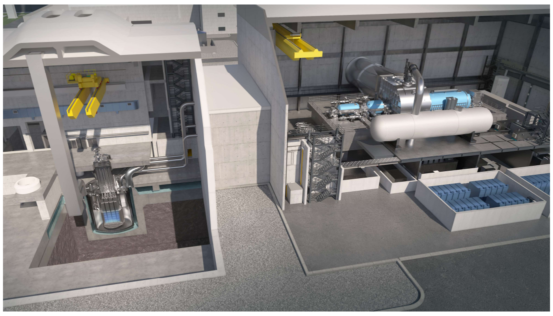
See fuel cycle approach.

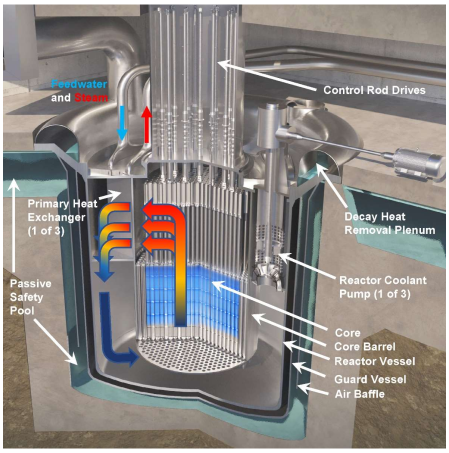
13. Development Milestones

2018	Concept design of SEALER-UK funded by UK BEIS
2020	SUNRISE project, including design of a demonstration unit funded by SSF.
2021	LeadCold and Uniper form joint venture SMR AB
2022	The SOLSTICE project including design, construction and operation of the electrically heated mock-up SEALER-E in Oskarshamn is funded by the Swedish Energy Agency.
2024	The SEALER-E electrical mockup is taken into operation
2030	The SEALER-D nuclear demonstration unit is taken into operation



Westinghouse Lead Fast Reactor (Westinghouse Electric Company, USA)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse Electric Company, LLC, U.S.A.
Reactor type	Pool-type, liquid metal cooled fast reactor
Coolant/moderator	Lead / fast spectrum
Thermal/electrical capacity, MW(t)/MW(e) Load following (MWe)	950 / ~450 (Net Avg.), 300 - 600 with constant core power
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Nearly atmospheric / 33
Core Inlet/Outlet Coolant Temperature (°C)	390 / 650 (target, pending material qualification)
Fuel type/assembly array	Oxide, with provision for transition to Nitride
Number of fuel assemblies in the core	325
Fuel enrichment (%)	≤ 19.75%
Refueling Interval (full power years)	8 - 15 (depending on fuel)
Core discharge burnup (GWd/ton)	up to 100
Reactivity control mechanisms	Control and shutdown rods; thermal/pressure activated, non-rod-based passive shutdown
Approach to safety systems	Passive: IAEA passive safety category B goal
Design life (years)	60 (components), 100(structures)
Plant footprint (m^2)	~40 000 w/ air-cooled condensers
RPV height/diameter (m)	Approx. 9.0 / 6.7
RPV weight (metric ton)	~215 (Assembly of RV, GV, flange, and passive cooling passages)
Seismic Design (SSE)	The same as typical LWRs
Fuel cycle requirements/approach	Once-through fuel cycle with single-batch long-life core design. Closed cycle possible.
Distinguishing features	Compact configuration with hybrid microchannel heat exchangers; non-reactor-based load follow
Design status	Conceptual design

1. Introduction

The Westinghouse lead fast reactor (LFR) is a medium-output, modular construction plant harnessing a lead-cooled, fast spectrum core operating at high temperatures in a pool configuration reactor. The simplicity of its safety systems, high-efficiency advanced ultra-supercritical (AUSC) water balance of plant (BOP), compact reactor building, absence of high-pressure containment, streamlined modular construction, and integrated non-reactor-based load following capability results in unparalleled economic potential, allowing the Westinghouse LFR to supply clean competitive energy even in the most challenging market conditions.

2. Target Application

The Westinghouse LFR is designed to be a versatile plant, with baseload electricity production and load leveling as the primary design focus, but with the capability to fulfill a range of non-electric applications such as process heat, desalination, and hydrogen production needs according to market demand. Its output is sufficiently small to integrate lower-capacity grids while also being substantial enough to be used in standard baseload plant applications. High temperatures permit extremely high BOP efficiencies and use for many process heat applications. Furthermore, a lower cost per MW(e) permits further electrical boosting of process temperatures while still being competitive at any temperature. Similarly, high temperature and BOP efficiency, combined with projected low plant cost, permit the use of hybrid heat/electricity methods providing the potential for cost-effective hydrogen generation. Integrated thermal energy storage using low-cost materials, coupled to standard BOP equipment, allows for non-reactor-based load follow to complement non-dispatchable energy forms while maximizing energy production. These capabilities could allow for the increased use of renewable technologies, making nuclear power and renewables complementary.

3. Design Philosophy

The Westinghouse LFR was designed to harness the outstanding safety, neutronic, and thermal characteristics of molten lead coolant to simplify safety, reduce overall plant size, and maximize the BOP efficiency. This, when coupled to fuel cycle flexibility resulting from operation in fast neutron spectrum, and non-reactor-based load follow, drives towards the ultimate goals of being economically competitive against any competing energy form in free markets while maintaining mission flexibility to customers worldwide.

4. Main Design Features

a) Balance of Plant (Power Conversion)

While previous variants of this technology aimed to take advantage of an advanced supercritical CO_2 power cycle, the unique operating conditions and needs of the LFR, along with advances in steam cycles (so-called advanced ultra-supercritical technology), has resulted in a technology change “back” to a water-based cycle. While this does result in larger turbomachinery, it permits shrinking of the primary heat exchangers and their connections on the LFR’s highly compacted vessel layout. Along with other LFR-specific benefits and reduced capital cost, it has allowed a significant reduction in the LFR’s vessel size and unprecedented power density. Furthermore, due to the aforementioned cycle developments, over 47% net efficiency is predicted using an air-cooled condenser (ACC).

b) Reactor Core

The core employs a conventional fuel assembly configuration, featuring solid fuel in cylindrical cladding, arranged in a hexagonal array. A staged approach to deployment is adopted for fuel system materials, featuring higher-maturity materials for LFR technology demonstration followed by higher performance materials as they become available for use. Specifically, higher technology readiness oxide fuels (UO_2 and MOX) in 15-15Ti-type austenitic steel cladding, such as D9, are considered for near-term use, while advanced fuel (uranium nitride being the focus) in an enhanced-performance cladding are envisaged to support an increase in burnup and operational temperature in follow-on LFR units. Consistent with LFR’s goal to ensure fuel and fuel cycle flexibility, fuel transitions can easily be accommodated as the various core designs only differ in geometric characteristics inside of the assembly duct, with assembly footprint and core layout being the same regardless of the fuel adopted. The core design uses ^{235}U fuel with enrichment <20% while maintaining the option to burn Pu-containing fuel through its MOX version.

c) Reactor Coolant System

The Westinghouse LFR features a novel reactor design configuration utilizing high power density hybrid microchannel-type primary heat exchangers (PHE) integral to the upper part of the core barrel. The compactness of the PHE design reduces the overall volume of the RV. This arrangement allows the reactor coolant pump (RCP) impellers to be placed in the lower temperature coolant discharging from the PHEs, reducing service temperature of the rotating components and related material design challenges. The figure above depicts the primary coolant flow path. After exiting from the PHE, primary coolant at cold pool temperature flows through the RCPs and is sent to the core, where it is heated and discharged to the upper plenum. The primary coolant then flows radially through the PHE and returns to the RCP inlet plenum. The configuration and resulting flow pattern also ensure the entire RV is in contact with lower temperature coolant, easing material requirements for this component.

d) Primary Heat Exchangers

Located inside the reactor vessel and integrated into the core barrel internal structure are the three primary heat exchangers (PHEs) to transfer heat to the secondary side working fluid; supercritical H_2O at 330 bars. With no welds in the main body, very small H_2O channels within diffusion bonded plates, and fluid headers located outside of the RV/contained area, a robust structure capable of maintaining extreme pressure differentials is created. When combined with the lack of exothermic reaction between primary lead coolant and BOP fluid, these elements allow PHE’s placement directly into the RV pool with limited risk of a significant RV pressurization event, eliminating the need for an intermediate heat transport loop present in other advanced reactor technologies, resulting in a more compact and cost competitive plant.

e) Non-Reactor-based Load Levelling

Westinghouse is currently developing thermal energy storage systems capable of providing load-levelling for thermal power plants. The storage system maximizes economic advantage by being integrated with the same turbine and generator as would be used for power generation, managing supply fluctuations produced by renewable sources by storing heat energy when electricity demand is low and selling produced electricity from that stored heat when it is high, all while maintaining the reactor core at full power. This is accomplished through manipulation of feedwater and turbine extraction flows to either increase or decrease the mass flow and steam conditions through the turbine.

At present, it is predicted to allow non-reactor-based load follow continuously between 300 and 600 MWe (Net) with round trip efficiencies competitive with battery solutions. In addition to ongoing work to enable coupled electricity + heat, hydrogen generation, and integrated desalination, it should be noted that the same systems and components used in the energy storage system may be integrated with solar-thermal boosting..

5. Safety Features

The Westinghouse LFR harnesses the inherent favorable safety characteristics of the lead coolant to simplify the reactor design and lower plant cost while allowing for the highest level of safety.

a) Robust, Inherently Safe Design Characteristics

The following characteristics of the design enhance its safety inherently:

- Thermophysical properties of lead, including its high boiling point (1745°C); atmospheric pressure operation; lack of exothermic chemical reaction with water and air; ability to retain some key fission products; shielding capability; high thermal conductivity; and, when combined with a pool-type primary system configuration, high thermal inertia
- Lead's excellent neutronic properties for operation in fast neutron spectrum allows the fuel rod lattice to be opened relative to sodium fast reactors, resulting in a minimum neutronic penalty while providing a significant enhancement in natural circulation capability during accidents, due to the associated reduction in core pressure drop
- Integral, pool-type configuration of the primary system eliminates primary line break, thus eliminating loss of coolant concerns by design
- Favorable reactivity feedback typical of liquid metal fast reactors
- Robust, microchannel hybrid PHE reduces chance of secondary break and substantially reduces its severity
- Underground placement of components important to safety.

b) Passive Heat Removal

The LFR harnesses its high temperature capability in order to use radiation heat transfer between the RV and GV to remove reactor core decay heat during a safety event. The GV is submerged in a pool of water sufficient to remove heat from the reactor for several days, subsequently transitioning to natural circulation air-cooling once water is depleted. Being radiation heat transfer-driven, this system does not require any actuation signal (i.e., it is always on) and does not result in significant parasitic losses during normal plant operation as the corresponding RV temperature, on the order of 400°C, is too low to result in effective radiation heat transfer. Instead, during a safety event the increase in RV temperature accelerates radiation heat transfer which, because of its function of temperature to the 4^{th} power, becomes effective in removing RCS heat only when it is needed.

c) Pressurization Events

In addition to not having any source of credible RV pressurization events originating from the BOP, no BOP plenums, piping, or headers will be located within the primary containment (represented by the RV, GV and common cover), thus removing the potential for large leaks in nuclear-related areas. The limiting credible break size in the contained area is reduced to the microchannels in the diffusion bonded block of the PHE. While small in terms of leak size, the RV pressure resulting from a microchannel break could become substantial over time without mitigation. Due to the target of IAEA Passive Safety Category B, no isolation valves are credited for use in PHE leaks (although non-safety isolation valves will be present). Instead, condensing chambers are paired with a passive, pressure-activated reactor shutdown mechanism and tailored power conversion system design which limit releasable inventory, ensure RV integrity, and eliminate the need for a traditional, high-pressure containment.

Reactivity management and shutdown. A reliable, diverse, redundant reactivity control and shutdown system comprised of control and shutdown assemblies ensures reactivity management and shutdown capabilities for all operating and anticipated events. This includes the possibility for a non-rod based, passive means to shut down the core based both on over-temperature and over-pressure transients.

d) Spent Fuel Approach

Because of, primarily, adoption of a longer active length relative to traditional fast reactors, the LFR core has a low power density (despite the high-power density of the overall nuclear system). Among many other benefits, this low power density has commensurate reductions in linear decay heat power. Recognizing the long, unshuffled, once-through fuel cycles possible with the LFR and their intrinsic capacity factor benefit, this low duty condition supports adoption of a direct-to-cask refueling scheme, with continuing work being conducted to prove its safety and commercial efficacy.

Utilizing Westinghouse's latest cask technology, this system will allow for direct lifting of used fuel assemblies into a cask following a brief waiting (cooling-off) period, estimated at less than 90 days. Given the infrequency of refueling, especially in anticipation of further advanced fuel types (such as Uranium Nitride), the many benefits in plant and operational simplicity, and reduced capital cost, were deemed worthwhile to pursue in exchange for extended, although infrequent, outages.

6. Instrumentation and Control Systems

A design goal for the development of the Westinghouse LFR's plant safety systems is to not rely on signals from the instrumentation and control (I&C) system. As a result, most components, systems, and software used to control the plant will be commercial grade.

7. Plant Layout Arrangement

As previously noted, the Westinghouse LFR design results in an ultra-compact nuclear system such that mass (often a challenge for lead-cooled reactors) is not such as to require non-standard support provisions. The GV and RV will be suspended from concrete support pads into a safety pool. The safety pool into which the GV is submerged will reside in the lowest levels of the plant, with no adjacent rooms or cavities to form leakage paths. All of these areas, as well as an area located above the reactor platform, will be located underground. At grade elevation will be an impact shield sufficient to provide protection from external threats. All components and systems with safety significance (outside of the safety pool itself) will be part of the fabricated RV/GV assembly and attached condensing tanks (located in safety pool), thus truly providing a 'nuclear island.'

The use of an AUSC system on the LFR reduces the size of in-vessel primary heat exchangers and their connected lines significantly and eliminates the need for a large condenser to be located in the turbine building (below the turbine). This shrinks the RV and turbine building and greatly eases the structural requirements of the turbine building, as the turbine can be placed much closer to grade. This arrangement results in a compact nuclear/turbine island with significant, large-component erection (of the ACCs) performed outside of the plant, allowing for more parallel construction activities and reduced construction duration.

8. Design and Licensing Status (Design and Testing Status)

Westinghouse has established an international team of partners dedicated to successfully delivering a high performance commercially viable plant. Between ten and twenty state-of-the art test facilities in the U.S., U.K. and Europe are currently being used or constructed in order to demonstrate key LFR's materials, systems, components and phenomena. These include a full-height passive heat removal facility, a versatile lead loop facility to test components such as the fuel bundle and PHE at prototypic conditions, and multiple material testing facilities.

9. Development Milestones

Relatively conventional materials have already been extensively tested in liquid lead and/or in fast spectrum, e.g., SS316 for low-DPA structural components and D9-type steel for fuel rod cladding. Testing of more advanced materials, and subsequently of individual components at higher temperatures, will (and is being) performed in a controlled environment to qualify them for use in evolved designs. LFR serves to answer the call of future energy markets, allowing a multitude of missions, fuel cycles, and operating strategies to be adopted in various locations and nations around the world.

10. Development Milestones

2020s – 2025	Design development and licencing process Additional testing is performed to validate enhancements to the plant performance.
2025 – 2026	Start of construction of the first of a kind (FOAK) LFR demonstration plant
2030	FOAK LFR demonstration plant to generate ≤300 MW(e)
Beyond 2030s	A higher output ~450 MW(e) FOAK plant representative of the commercial fleet will be licensed and deployed.

PART IV. MOLTEN SALT SMALL MODULAR REACTORS

IMSR400 (Terrestrial Energy Inc., Canada)

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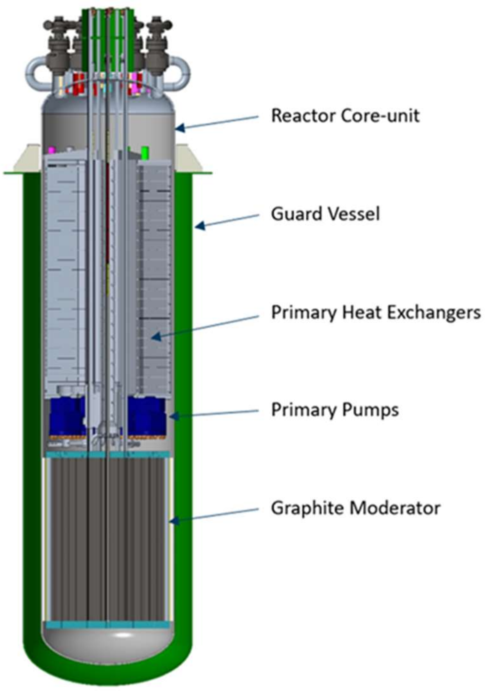


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Terrestrial Energy Inc., Canada
Reactor type	Molten salt reactor
Coolant/moderator	Fluoride Fuel Salt/graphite
Thermal/electrical capacity, MW(t)/MW(e)	440 / 195 per operating Core-unit; two Core-unit configuration
Primary circulation	Forced
NSSS Operating Pressure (primary/secondary), MPa	<0.4 (hydrostatic)
Core Inlet/Outlet Coolant Temperature (°C)	620 / 700
Fuel type/assembly array	Molten salt fuel
Power conversion process	Steam generator supplying super-heated steam to turbine
Fuel enrichment (%)	<5%
Refuelling cycle (months)	84; before Core-unit replacement
Reactivity control mechanism	Short-term: negative temperature coefficient Long-term: online fuel addition
Approach to safety systems	Passive
Design life (years)	56
Plant footprint (m^2)	45 000
RPV height/diameter (m)	18.0/4.1
RPV weight (metric ton)	300
Seismic design (SSE)	0.3g
Fuel cycle requirements / approach	Once-through, low enriched uranium fluoride eutectic mix Fuel Salt
Distinguishing features	Core-unit replaced completely as a single unit every 7 years; Fuel salt is reused from one Core-unit to the next
Design status	Detailed design

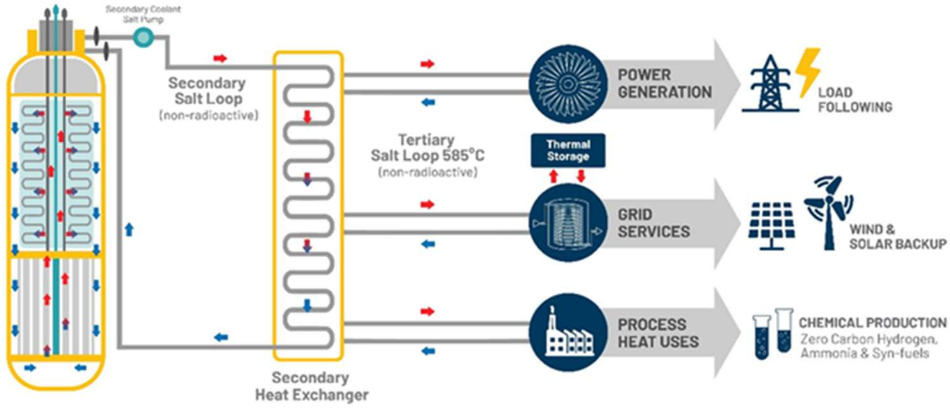
1. Introduction

The integral molten salt reactor (IMSR®) is a 442 megawatt-thermal per Core-unit (IMSR400) small modular molten salt fuelled, graphite moderated, thermal spectrum reactor. Terrestrial Energy has developed a two-unit configuration which can deliver 884 MW(t)/390 MW(e). The IMSR® integral nuclear reactor design. features a completely sealed reactor vessel with integrated pumps, heat exchangers and shutdown rods all mounted inside a single vessel; the IMSR® Core-unit. The sealed Core-unit is replaced completely at the end of its useful service life (nominally 7 years). This allows factory production levels of quality control and economy, while avoiding any need to open and service the reactor vessel at the power plant site. The IMSR400 achieves the highest levels of inherent safety as there is no dependence on operator intervention, powered mechanical components, coolant injection or their support systems such as electricity supply or instrument air in dealing with upset conditions.

2. Target Application

The IMSR® plant is designed to serve electric power grids and industrial heat and power application as represented in the schematic.

3. Design Philosophy



The underlying molten salt technology of the IMSR® design is molten salt reactor technology that was the product of successful and extensive research programs at the Oak Ridge National Laboratory (ORNL) in the 1950's, 1960's and 1970's Terrestrial Energy's technology development focuses on building the foundations of cost and safety with ORNL's design as a starting point. Three technical programs are tightly integrated – R&D, Engineering, Safety Analysis – over the entire project life-cycle design, construction, commissioning, operations, and decommissioning. These programs and the attendant processes and outputs (documentation in accordance with Terrestrial Energy's Management System) are the feedstock for licensing submissions and to the Cost Engineering program as well).

4. Main Design Features

(a) Power Conversion

The solar salt loop, which is pumped from the nuclear island to a separate building outside the Nuclear Facility, either supplies a steam generator that generates superheated steam for power generation or is used to drive process heat applications. The steam circuit powers a conventional, off-the-shelf industrial steam turbine for power generation and/or industrial steam production, depending on the required application. Alternately, some or all the hot molten solar salt may connect directly to a process heat application.

(b) Reactor Core

The Core-unit is manufactured in a controlled factory environment and then brought to the reactor plant site where, following final assembly, it is lowered into a surrounding guard vessel located in a below grade reactor silo. The IMSR® Fuel Salt is a thermally stable fluid with excellent coolant heat transfer properties, and high intrinsic radionuclide retention properties. As shown in the above figure, a secondary coolant salt loop, filled also with a fluoride salt (but without fuel), transfers heat from the Core-unit Fuel Salt via primary heat exchangers to a third industrial solar salt loop.

(c) Reactivity Control

Reactor control is assured through negative temperature feedback made possible by the neutronic behaviour of the molten salt fuel in the reactor core. This negative temperature feedback avoids overheating by assuring criticality control, without the need for the activation of the engineered control system. Molten salt fuel does not degrade by heat or radiation, which gives a high-power limit to the salt fuel. Although shutdown rods are integrated into the IMSR® Core-unit, these are for the operational control, and not needed for safety. These shutdown rods will shut down the reactor upon loss of forced circulation and will also insert upon loss of power.

(d) Reactor Vessel and Internals

The Reactor Vessel houses the graphite core, the integral primary pumps and heat exchangers, and the thimbles that receive the shutdown mechanism rods.

(e) Fuel Characteristics

The IMSR® is a liquid-fuel reactor – there are no solid fuel elements in the reactor core. The fuel, in the form of uranium tetrafluoride (UF_4), is infused with a primary coolant forming a near-eutectic salt mixture. The primary salt mixture consists of low-cost fluoride salts and avoids the use of either lithium or beryllium. The benefit of this primary coolant is that it minimizes the production of tritium. The fuel- primary coolant salt mix is pumped through a critical, graphite moderated (thermal spectrum) core, and then through the integral heat exchangers to transfer its heat to the secondary coolant salt loop. Overall, the IMSR400 Fuel Salt provides the basis for a less complex reactor configuration with many safety attributes.

(f) Fission Product Management

The first barrier for fission product release is the fluoride Fuel Salt. Under the operation redox potential, the majority of elements produced as a result of fission (fission products) will form their respective stable fluorides, trifluorides, and iodides, which are soluble in the Fuel Salt. The Core-unit/Reactor Vessel represents the second barrier. Since the Core-unit operates at low pressure the probability of explosive release is extremely low. Additional successive barriers are the Guard Vessel, the Containment, and the Reactor Auxiliary Building

(RAB).

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 previously mentioned – *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 figure below.

Control	Cool	Contain
IMSR® has a strong negative reactivity coefficient of temperature for inherent reactivity control. Passive shutdown that does not require the traditional use of control rods.	IMSR® design assures heat dissipation in all circumstances as IMSR® Fuel Salt is molten and serves also as the coolant. This uniquely enables convective cooling of the nuclear fuel. IMSR® design is small and operates at 700ºC. Small high temperature reactors have greater inherent cooling capabilities.	IMSR® Fuel Salt forms strong ionic chemical bonds with many fission products to uniquely contain radioactivity within the salt and the reactor. No chemical driving forces are present, in part, as zirconium metal-water reactions are not possible. No physical driving forces as IMSR® operates at atmospheric pressure.

(b) Decay Heat Removal System

The decay heat from the fuel in the reactor core is removed by the Internal Reactor Vessel Auxiliary Cooling System (IRVACS). IRVACS is an air coolant based fully passive system with no operator intervention required and does not rely on any instrumentation for its operation.

(c) Emergency Core Cooling System

The IMSR400 is designed with passive heat removal design features; therefore, an Emergency Core Cooling System is not required for safety.

(d) Containment System

The 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 (FSSTs), Gas Holding Tanks (GHTs), Hot Cell, and connecting piping.

(e) Chemical Control

Chemical control is achieved by ensuring certain predetermined purity of materials (fuel-bearing fluoride salt mixture, graphite and alloy) prior to operations and establishing and maintaining the predetermined redox potential window inside the Core-unit as well as the prevention of material contamination during operations.

(f) Spent Fuel Cooling Safety Approach / System

The irradiated Fuel Salt will reside in Fuel Salt Storage Tanks 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 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. Due to its molten Fuel Salt characteristics, the IMSR400 "core damage frequency" safety goal of less than 10^{-5} per reactor year is expressed in terms of loss of reactor vessel integrity. The large release frequency is less than 10^{-6} 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 128 outage days planned for Core-unit replacement (every 7 years) and other maintenance.

7. Instrumentation and Control System

The control safety function for reactor power for the IMSR400 is inherently part of the reactor's fundamental physics characteristics and does not require an engineered active system to maintain control of power in any design basis accident scenario. A plant investment protection system and a plant control system provide all control functions.

8. Plant Layout Arrangement

The IMSR® Plant consists of two distinct facilities: a Nuclear Facility (NF) and a Thermal and Electric Facility (TEF) shown in the figure below. Acting together, they form the IMSR® Plant, an 884 MW(t) nuclear powered cogeneration facility. The TEF output can be configured in any combination of thermal and electric power output. In the figure, the NF shows two Reactor Auxiliary Buildings (RABs), R1 and R2, each housing one operating Core-unit within its own Containment system. A common Control Building is located between the two RABs.

9. Testing Conducted for Design Verification and Validation

TEI 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 Slat 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

Terrestrial Energy completed Phase 1 of the Canadian Nuclear Safety Commission's (CNSC) Vendor Design Review (VDR) in November 2017, successfully meeting CNSC's requirements for the IMSR® design. In October 2018, the IMSR® entered Phase 2 pre-licensing VDR with the CNSC. Phase 2 VDR involves a detailed follow-up on Phase 1 VDR activities, and an assessment of the IMSR® power plant design's ability to meet CNSC's expectation in all its 19 design focus areas. VDR Phase 2 submissions have been completed as of early 2022 and the CNSC is preparing the final report for issuance. Regulatory interface with US NRC commenced in 2019 and is progressing toward first topical reports in late 2022. Terrestrial Energy's Postulated Initiating Events (PIE) methodology was subject of joint review by CNSC and US NRC, and the Regulators' final report was published in 2022.

11. Fuel Cycle Approach

The IMSR400 fuel consists of low enriched uranium fluoride, in a eutectic mix with other fluoride salts and utilizes a once-through, low enriched uranium fuel cycle. During Core-unit's operation, makeup Fuel Salt is added online. The development of the fuel supply chain is in progress with key suppliers for uranium conversion and enrichment services, and for transport and packaging. At the end of its useful life, the spent fuel is stored onsite until it can be moved offsite to a final waste management facility.

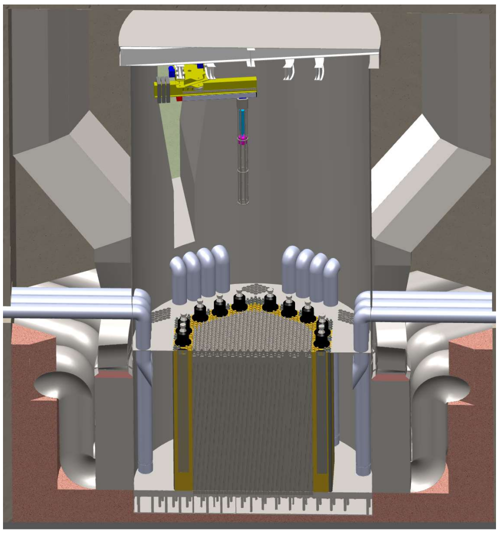
12. Waste Management and Disposal Plan

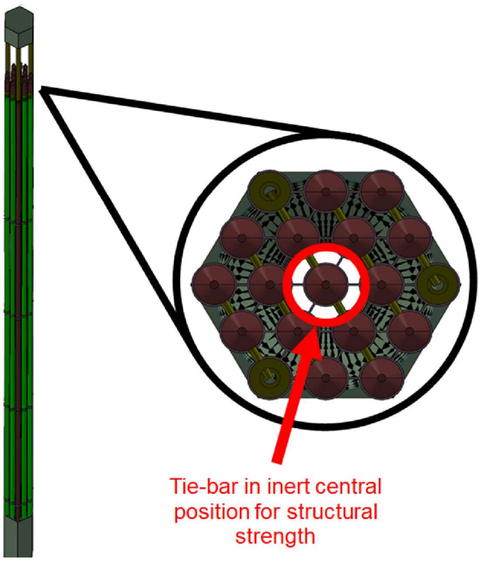
The used fuel from each Core-unit at its end of life is recycled/used in the next Core-unit through the plant's life cycle. After the end of the last Core-unit's life, the spent fuel is stored onsite until it is ready for transportation to and storage in an offsite final high-level waste repository facility.

13. Development Milestones

2015	Conceptual design completed
2016	Start of basic engineering phase
2017	Completion of CNSC pre-licensing Phase 1 Vendor Design Review
2018	Commenced CNSC pre-licensing Phase 2 Vendor Design Review
2019	Commenced irradiation test program on graphite
2020	Strategic Innovation Fund (SIF) Award from Canadian Government
2022	Basic engineering complete; planned - CNSC VDR Phase 2 completion
2026	Planned - secure necessary licenses
2027	Planned - commence construction of a first full-scale IMSR
2031	Planned - first IMSR plant in-service

SSR-W (Moltex Energy, Canada)





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 MgCl/NaCl No moderator
Thermal/electrical capacity, MW(t)/MW(e)	750 / 300 continuous as baseload 750 / 900 as 8-hour peaking plant
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Atmospheric
Core Inlet/Outlet Coolant Temperature (°C)	575/625
Fuel type/assembly array	Molten salt fuel within vented fuel tubes
Power conversion process	Molten salt storage tanks feed into steam turbines
Fuel enrichment (%)	N/A (uses recycled spent fuel)
Refuelling cycle (months)	N/A (on-load refuelling, 10 days)
Core Discharge Burnup (GWd/ton)	~100
Reactivity control mechanism	Negative temperature reactivity coef Boron carbide shutdown assemblies
Approach to safety systems	Eliminate hazard, passive engineering,
Design life (years)	60
Plant footprint (m^2)	22 500
RPV height/diameter (m)	14 / 7
RPV weight (metric ton)	50
Seismic design (SSE)	Yes
Fuel cycle requirements / approach	Spent CANDU fuel is recycled through WATSS to create SSR-W fuel
Distinguishing features	Molten salt fuel in conventional fuel Assemblies. Thermal energy storage.
Design status	Conceptual design

1. Introduction

The Stable Salt Reactor - Wasteburner (SSR-W) is unique in its use of molten salt fuel replacing solid pellets in conventional fuel assemblies. This brings the major advantages of safe molten salts without the technical hurdles of managing a mobile liquid fuel. The reactor is fuelled with very low purity, reactor-grade plutonium 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 reactor a low-cost peaking power plant rather than being restricted to baseload operation. This same system permits the entire steam cycle to be identical to that of coal-fired power stations and for it to be operated completely independently of the nuclear plant. The steam cycle is therefore not subject to nuclear regulations.

2. Target Application

The SSR-W is designed for countries with significant stocks of spent nuclear fuel. The reactor burns all of the higher actinide component of that fuel, leaving a waste stream of relatively short-lived fission products only, as it can recycle actinides from its own spent fuel as well as from spent CANDU 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;
- modular design, with a single 300 MWe reactor assembled from road-transportable, factory-produced modules;
- a fuel assembly form that is compatible with IAEA safeguards procedures used in reactors today.

4. Main Design Features

(a) Power Conversion

A molten salt to steam boiler is proposed to generate steam. The turbine is currently being specified for a 300 MWe plant similar to modern fossil fuel plants. There can be a larger turbine of up to 900 MWe installed, to be coupled with the solar salt storage or multiple smaller turbines. This will depend on the local electricity grid needs and economics.

(b) Reactor Core

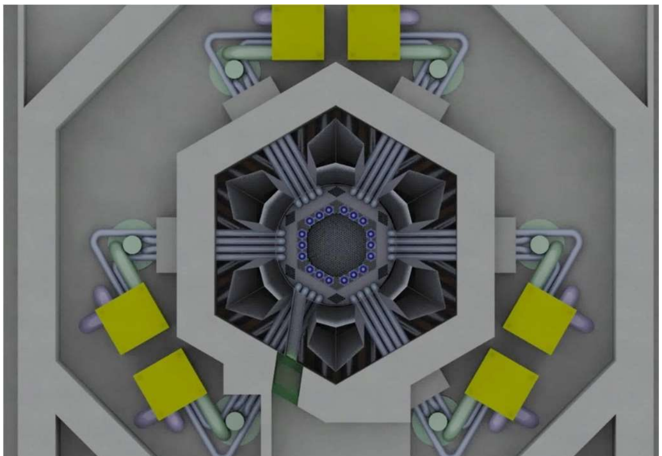
The reactor core's horizontal cross section is a circle mostly filled concentric hexagonal rings. The innermost rings of hexagons in the core are fuel assemblies, with shutdown assemblies spaced out along the periphery. The gaps between these shutdown assemblies are filled in with permanent reflector 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 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 activity through radial neutron leakage.

(d) Reactor Vessel and Internals



SSR-W Coolant Loop

The reactor coolant loops take heat from the fuel assemblies and transfer this heat to the thermal storage system. This is achieved through six coolant salt pumps, which feed six primary heat exchangers by drawing coolant from the tank periphery through the suction pipes. This coolant is then discharged to the lower plenum via the downcomers and jet nozzles. The tube connections to the heat exchangers pass in angled paths through the concrete biological shield, to prevent a shine path (neutron or gamma radiation escaping past shielding).

In normal conditions, the flow is forced by pumps to the lower plenum (below the diagrid) and sucked from the upper plenum (coolant volume above the height of fuel), while also having a portion of the coolant bypass the suction pipes and flow freely down the tank wall to the lower plenum.

(e) Fuel Characteristics

The fuel salt is a mixture of 45 mol% potassium chloride (KCl) and 55 mol% actinide trichlorides (UCl3, PuCl3, etc.) and lanthanide trichlorides. It is redox stabilised to render it non-corrosive to steel by inclusion of metal zirconium in each tube, which maintains the salt in a strongly reducing state incapable of dissolving chromium from steel.

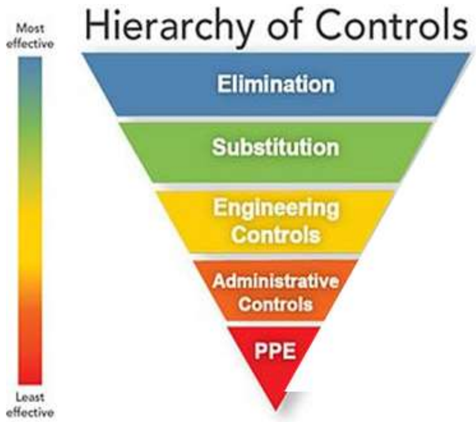
(f) Fission Product Management

The volatile fission products traditionally of concern, such as cesium and iodine, are salts and remain trapped in the fuel. Noble gas radioactive isotopes are held up by the fuel pin vents, which hold it until the gas contains acceptably low radiation levels. Those gases are then filtered by the reactor atmosphere conditioning system.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The design philosophy is to follow the internationally accepted principle of the risk mitigation pyramid. Therefore, the focus is to use inherent characteristics to eliminate hazards and to use engineering safety features to provide additional confidence and backup to the inherent physical characteristics.



Hierarchy of Controls

(b) Decay Heat Removal System

Natural convection of the primary coolant salt would continue in the event of a reactor shutdown or pump failure. Heat would then be transferred to the tank walls, where a finned air duct to atmosphere around the tank walls would take decay heat away indefinitely in an accident scenario.

(c) Emergency Core Cooling System

There is no safety injection system as the coolant has a very high boiling point and there is no credible leakage path to allow the core to be uncovered.

(d) Containment System

There are no significant internal pressures within the SSR-W. The primary containment for the reactor is the tube wall. The secondary containment is the coolant salt, which absorbs fission products in the event of tube failure. The third is the tank itself. The fourth is the concrete structure. Above the tank is an argon space, which has a stainless steel liner surrounded by a ~1 m concrete wall serving as the biological shield. The reactor building walls are ~300 mm thick reinforced concrete. This serves as both a building structure and a shield.

(e) Chemical Control

The reactor coolant salt chemistry is maintained in a reducing state through the use of an electrolytic cell (the redox control unit). The flow of coolant salt for the reactor coolant redox control unit is normally provided by the leak-off flow that is generated in the reactor coolant loop vacuum break line when the reactor coolant pumps are in normal operation.

(f) Spent Fuel Cooling Safety Approach / System

Several positions along the periphery of the reactor allow the fuel assemblies to cool down after being removed from the core. Afterward, they can be brought back into the WATSS process to allow them to be recycled into new fuel for the reactor.

6. Plant Safety and Operational Performances

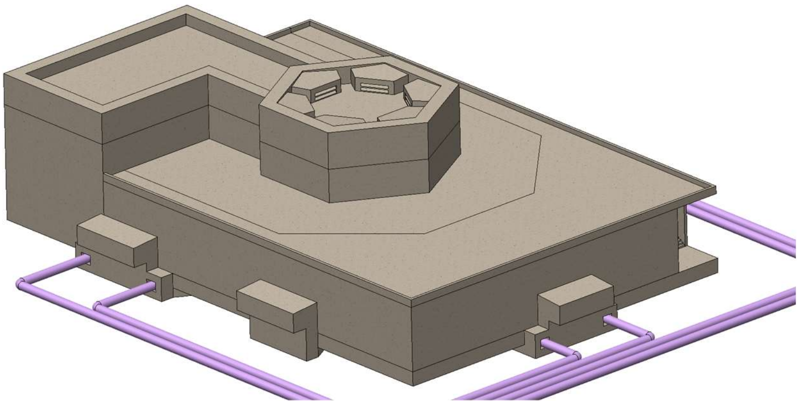
The design intent is that the release of radioactivity in excess of the agreed operating discharge authorization is expected to be less than 10^-6 per year. The ramp rates of the plant will be driven by the steam side, not the nuclear side.

7. Instrumentation and Control System

There is a separate protection and monitoring system. The protection system is hardwired for a four-channel coding system. The control system is a supervisory control and data acquisition (SCADA) system. The monitoring system is a distributed data acquisition and monitoring system with standard monitoring software.

8. Plant Layout Arrangement

The reactor biological shield (RBS) has a hexagonal cross section. The RBS is built up on the reactor basemat and both can effectively be considered as different parts of the same civil structure. There is a lining on the inside of the RBS Vessel that forms part of the reactor containment. The reactor tank is in the lower volume of the RBS vessel; the upper volume encloses the containment gas space above the reactor coolant level and provides a boundary for this gas space. The reactor tank is enclosed by the concrete RBS vessel, which provides protection against radiation from the reactor core and coolant.



SSR-W Civil Structure

9. Testing Conducted for Design Verification and Validation

The SSR-W and WATSS design is sufficiently developed to support an R&D plan that is now being executed. The novel componentry of the SSR-W includes the primary heat exchanger, the fuel assembly's gas venting facility, emergency heat removal system heat exchangers, and the fuel handling infrastructure. A medium-scale (tens of kgs of molten salt) thermal hydraulic loop is under construction and will demonstrate the phenomena that need to be accurately modelled by the appropriate software. A small-scale (<1 kg of molten salt) experimental rig has been built and operated to provide the first validation data for heat transfer coefficients necessary to modelling the thermal hydraulic performance of the Moltex core.

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. A successful Phase 2 review will demonstrate that there are no fundamental barriers to licensing the design in Canada.

11. Fuel Cycle Approach

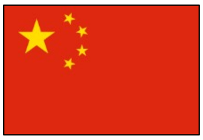
Current CANDU reactors run on the once-through natural uranium fuel cycle, which involves a single pass of the fuel through the reactor. Moltex plans to produce the fuel for the 300 MWe SSR-W (SSR-W300) from used CANDU fuel stocks using the WATSS process. WATSS separates out the fission products into 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 CANDU fuel, while generating useful power. Other radionuclides present in CANDU fuel are not consumed and remain as waste. Most of these radionuclides will be in a salt waste form that can be safely disposed of in the proposed Canadian deep geological repository.

13. Development Milestones

2014	UK patent granted for use of unpumped molten salt fuel in any reactor.
2014	Independent capital cost estimate complete.
2015	Pre- conceptual design complete and key claims validated by the UK's National Nuclear Laboratories.
2017	Conceptual design completed and CNSC VDR commenced.
2018	Master patent on static fuelled molten salt reactors granted in major geographies. Several other patents progressing through PCT process.
2018	Moltex successful in UK Government Advanced Modular Reactor Competition.
2018	Moltex signs agreement with NB Power to advance the development of the first commercial demonstration SSR-W in Canada.
2020	Moltex signs ONWARDs agreement with Ontario Power Generation.
2021	Moltex receives funding from the Government of Canada's Strategic Innovation Fund.
2021	Moltex completes CNSC VDR Phase 1.



smTMSR-400 (CAS/SINAP, China)

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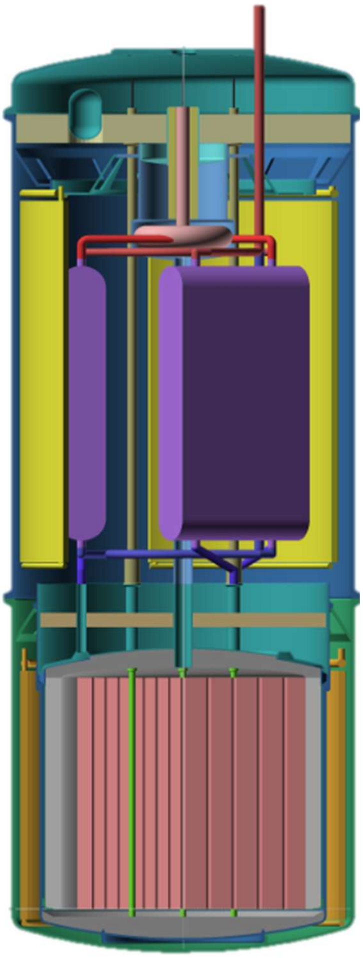


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	CAS/SINAP, China
Reactor type	Molten salt reactor
Coolant/moderator	LiF-BeF_2-ZrF_4-ThF_4-UF_4 fuel salt / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	400 / 168
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Near ambient
Core Inlet/Outlet Coolant Temperature (°C)	650 / 700
Fuel type/assembly array	Molten salt fuel
Power conversion process	helium/air/CO2 Brayton cycle
Fuel enrichment (%)	19.75
Refuelling cycle (months)	120 (batch-reprocessing off-line)
Core Discharge Burnup (GWd/ton)	~300
Reactivity control mechanism	Control rods, Online fuel addition, drain off fuel salt
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	40 000
RPV height/diameter (m)	~10 / 3.8
RPV weight (metric ton)	120
Seismic design (SSE)	0.3g
Fuel cycle requirements / approach	Th+U loading initial, LEU addition online, FP gas removal online, Batch-reprocessing offline
Distinguishing features	Replaceable Reactor module (~10 years); Passive safety Near 40% power contributed by thorium
Design status	Pre-conceptual design

1. Introduction

A three-step development route has been proposed by Shanghai Institute of Applied Physics, Chinese Academy of Science, (SINAP, CAS), to realize thorium-uranium breeding in molten salt reactor by the middle of this century. smTMSR-400 is a 400MW(t) / 168 MW(e) small modular Thorium Molten Salt Demonstration Reactor that forms part of the third step. It will demonstrate large-scale power produced based on the thorium fuel cycle and verify the conversion properties. Most off-line pyro-processing techniques, such as, fluoride volatility, vacuum distillation and electrochemical reduction, will be used in fuel cycle of smTMSR-400. Demonstration at this stage focuses on the R&D efforts and can reduce the risk and cost. The option also enable the gradual establishment of the thorium fuel and MSR supply chain system. smTMSR-400 will demonstrate the highly intrinsic safety and engineering reliability of molten salt reactor, such as shutdown by negative temperature feedback and draining off fuel salt, passive residual heat removal systems. The aim of safety design is to avoid any off-site emergencies and therefore increase the siting flexibility. smTMSR-400 uses small and modular designs to reduce and simplify the R&D challenge and difficulty of MSR. Modular design will make MSR more flexible to deploy at different places with various demands. Modular manufacture and assembly can improve the quality of equipment and enhance safety and economy. Modular construction can also reduce the financing cost and risk.

2. Target Application

smTMSR-400 is designed as a thorium convertor and in situ burner driven by low enriched uranium. The fuel cost is expected to be cheaper than pure uranium fuel mode. After offline batch reprocessing, thorium and uranium can be recycled into a new reactor in order to minimum the spent fuel mass as well as enhance the neutron economy.

It will be applied as a high temperature heat source, which not only can be used for electricity generation, but also can satisfy the energy diversified demands, such as, seawater desalination, heat supply, supercritical steam supply for industry demands and hydrogen production, etc.

3. Design Philosophy

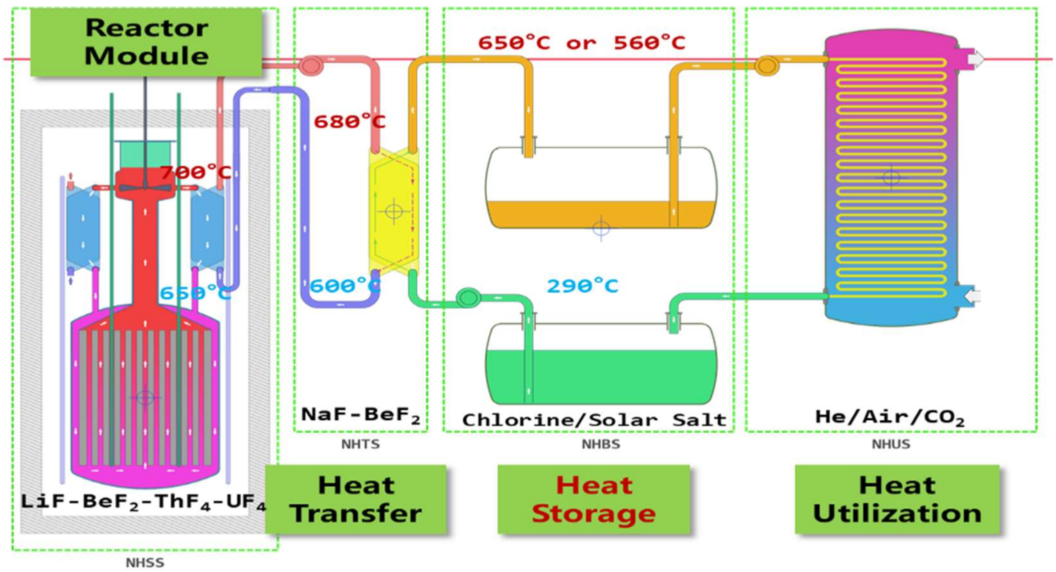
The design philosophy of smTMSR-400 includes:

- Using mature technologies and experiences accumulated from TMSR-LF1 project and potential technologies to be developed in the next years.
- The primary loop is designed as a compact reactor module and can be replaced every 8-10 years to solve the long-term irradiation problems of materials.
- -Online fuel addition and offline batch reprocessing mode is used. The power contribution from thorium should be more than 40%. The spent fuel can be used for offline pyro-processing and recycled for three times.
- -Passive safety design, consist of negative temperature feedback, passive fuel salt discharge, passive residual heat removal systems, and radioactive nuclides retention by salt coagulation.
- Heat storage system is applied for comprehensive energy utilization and peak-shift of electrical demand.

4. Main Design Features

(a) Power Conversion

The heat can be applied for electricity generation by helium/air/CO2 Brayton power system based on their technological maturity and application scenarios. Also, it can be applied for seawater desalination, heat supply and steam supply.



(b) Reactor Core

Reactor module is designed as a compact loop structure with reactor core, three salt/salt heat exchangers, one centrifugal pump and connecting pipelines. Reactor core is filled with LiF-BeF_2-ZrF_4-ThF_4-UF_4 fuel salt and hundreds of prismatic graphite blocks. ^{235}U is 19.75% enriched. Three salt/salt plate heat exchangers are arranged upon the reactor core side by side for compact layout. The size of reactor module is about ~3.8 m in diameter and ~10 m in height, which is suitable for railway transportation. Reactor module is packed in a safety vessel for additional radioactive confinement barrier. The reactor vessel and metal internals in touch with fuel salt are made by nickel-based alloy.

The fuel salt is heated up to 700°C by fission energy in the active core (fuel salt channels in graphite moderated core), then flows through the upper plenum, upper pipeline, driven by the pump, and then to the three plate heat exchangers. After heat transfer, the fuel salt is cooled down to 650°C, then flow through the down-comer, lower plenum and back to the fuel salt channel.

(c) Reactivity Control

Six control rods are used to control the reactivity change covering from subcritical status, power lift condition, xenon poison, operation disturbance, and excess reactivity. Most of control rods are used to adjust the power lift condition due to the negative temperature feedback.

(d) Reactor Vessel and Internals

The reactor vessel is made of nickel base alloy Hastelloy-N, which shows good mechanical behaviour and corrosion resistance under high temperature. The internals include the support plates of core, the barrel, the arc plate of the upper and lower chamber, the tubes of control rods, and so on. Composite material will be also used in smTMSR-400.

(e) Fuel Characteristics

The fuel salt consists of LiF-BeF_2-ZrF_4-ThF_4-UF_4. LiF-BeF_2 is used to lower the melting point and keep a good neutronic and thermal-hydraulic properties. In order to reduce the neutron absorption and tritium production, Li-7 is required to be enriched more than 99.99%. ZrF_4 is used to prevent the precipitation of uranium and thorium in the case of accidental introduction of oxygen impurities. UF_4 with 19.75% enrichment is the start-up fuel, and ThF_4 is the breeding fuel.

(f) Fission Product Management

During the operation, the fission gases are online swept through bubbling method to the off gas system, where they will decay to a low radioactive level and then emit into the atmosphere. After 10 year operation, soluble fission products will be discharged into the drain tank and decay for several years, then will be sent to the offline pyro-processing facility for separation and permanent disposal.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

smTMSR-400 contains two sets of control rod system with different driving mechanisms, which will automatically go down in accidents. Low excess reactivity will also make it possible to shut down with the negative temperature feedback alone. After that, fuel salt will be drained off for long-term shutdown.

(b) Decay Heat Removal System

smTMSR-400 is designed with two passive decay heat removal systems (DHRS). Natural circulation will form in the reactor module in case of force circulation failure. When the nuclear heat supply system fails, two kinds of DHRSs will ensure long-term cooling of fuel salt. One is located around the safety vessel in the silo (DHRS1), another is in the fuel salt drain tank directly installed below the reactor vessel (DHRS2). DHRS1 is operated under normal accidents to cool the safety vessel and reactor vessel. DHRS2 is triggered after the freeze-valve opens passively by increased accident temperatures or by active heating.

(c) Containment System

There are four containment barriers in smTMSR-400. The first one is the reactor module made by high-temperature-resistant and corrosion-resistant nickel-based alloy, the freeze valve function prevents the first barrier from exceeding its temperature limits and thus to keep its integrity under accidents. The second one is the safety vessel, which will ensure gas tightness and contain the leaked salt under beyond design basic accidents. The third one is the fuel salt itself, which has a large degree of retention on some important radioactive elements. The solidification of fuel salt will prevent further leakage. The final one is the underground construction, which can effectively prevent the spread of radioactive materials in an accident and can resist natural disasters and terrorist attacks.

(d) Chemical Control

Online chemical control system with addition of metal Be or ZrF_4 is designed to control the oxidation-reduction potential of fuel salt.

(e) Spent Fuel Cooling Safety Approach / System

-The spent fuel salt will stay in the drain tank for several years with continuously cooling.

6. Plant Safety and Operational Performances

There is no core meltdown accident in smTMSR-400, the fuel salt itself is at liquid state during the normal operation. There are no significant safety risks, such as high pressure, violent chemical reaction or explosion, which may lead to the large early release of radionuclides. The high temperature margin, the large thermal capacity of fuel salt and graphite, the passive decay heat removal ability and the low excess reactivity guarantee the integrity of containment boundary. Potential less leak of gas or salt in the case of pipe or valve corrosion can be limited by multi-physical barriers. It is expected the eliminate the possibility of massive release of radioactivity. The capacity factor can reach 95% due to the ability of online fuel loading.

7. Plant Layout Arrangement

The reactor module, fuel management and gas removal system are under the ground. The heat storage system and heat utilization system are above the ground. Power expansion can be realized by employing multiple reactor units. The power plant can adopt single unit, two units and different units at one site, and realizes different power output to meet the energy needs of different application sites.

8. Testing Conducted for Design Verification and Validation

To be developed.

9. Design and Licensing Status

The design is in a pre-conceptual stage of development and licensing activities have not yet been undertaken.

10. Fuel Cycle Approach

The fuel cycle in smTMSR-400 belongs to a semi-enclosed type. During the operation, fuel will feed online and some of fission gases are removed and treated in the off-gas system. After each 10 year, the spent fuel salt will be transported to the pyro-processing facility for recovering the uranium, thorium and carrier salt. The remained TRU and FPs will be minimized and be permanently disposed.

11. Waste Management and Disposal Plan

A pilot stage pyro-processing facility will be established in 2030s.The spent fuel in smTMSR-400 will be sent to the facility for fuel salt recovery and to minimize the radioactive inventory for disposal.

12. Development Milestones

2018-2021	Pre-conceptual design phase and technology research.
2022-2024	Conceptual design phase and technology validation
2025-2027	Engineering design phase.
2028-	Projected deployment (start of construction) time.

Compact Molten Salt Reactor (Seaborg Technologies, Denmark)

CMSR POWER BARGE

CMSR

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Seaborg Technologies ApS, Denmark
Reactor type	Molten salt reactor / thermal spectrum
Coolant/moderator	Fluoride fuel salt / NaOH salt (patented moderator)
Thermal/electrical capacity, MW(t)/MW(e)	250 / 100 per CMSR
Primary circulation	Forced circulation
Operating Pressure (MPa) (primary/secondary/NSSS)	0.45 / 1.05 / 18
Core Inlet/Outlet Coolant Temperature (°C)	600 / 670
Fuel type/assembly array	HALEU / Molten salt fuel
Power conversion process	Superheated steam driven turbine / generator (Rankine Cycle)
Fuel enrichment (%)	HALEU/different scenarios considered
Refuelling cycle (months)	No online refuelling (One Fuel Cycle: 144 months / 12 years)
Core Discharge Burnup (GWd/ton)	Pending Fuel Enrichment Selection
Reactivity control mechanism	Negative temperature coefficients, regulating and safety rods, fuel salt draining
Approach to safety	Automatic Shutdown, Passive Cooling
Design life (years)	12 per CMSR / 24 per Power Barge
Plant footprint (m^2)	200 - 800 MWe – 5 000 – 14 000 m^2 Depending on the Power Barge model.
RPV height/diameter (m)	5.5 / 2.5
RPV weight (metric ton)	50 – 60
Seismic design (SSE)	Under review
Fuel cycle requirements / approach	HALEU one cycle Off-Site reprocessing
Distinguishing features	CMSR integrated into a floating non-self-propelled Power Barge; new CMSR units installed after 12 years. CMSR Power Barge operates for 24 years. Moduar Nature provides output from 200-800 MWe Liquid fuel which is the primary coolant, and a liquid moderator.
Design status	Conceptual design

1. Introduction

The Seaborg Compact Molten Salt Reactor (CMSR) is an advanced, small, and modular molten salt reactor characterized by using a liquid fluoride molten salt fuel and a liquid patented molten hydroxide moderator. The CMSR produces approximately 250 MWth (MW-thermal) through the fission of High-Assay Low-Enriched Uranium (HALEU) in a thermal neutron spectrum. The liquid moderator, which does not accumulate irradiation damage, enables the CMSR an operating life of 12 years, during which no refuelling is required. The fuel and moderator salts have high melting points. As a result, the CMSR needs no pressurization above what is needed to circulate the salts. The CMSR has unique inherent safety features. The CMSR is deployed on modular and standardized non-self-propelled Power Barges, scalable from 200 MWe to 800MWe, and with an operating life of 24 years.

2. Target Application

The CMSR Power Barge is designed to generate power for the electrical grid and for industrial applications such as hydrogen electrolysis. Furthermore, the CMSR Power Barge can supply high temperature heat for non-electrical applications such as district heating, water desalination, and industrial processes.

3. Design Philosophy

The CMSR design philosophy is focussed on implementing inherent and passive safety features as well as fully automated shutdown. The design ensures high proliferation resistance. The design is aimed to reduce complexity and ensure increased manufacturability and modularity. Proven technologies are used as far as possible and innovative technologies are applied only where necessary to facilitate the design goals. The modular and compact nature facilitates the integration of the CMSR into the CMSR Power Barge. The philosophy of the CMSR Power Barge is to combine the CMSR design with the construction and high-quality, matured capabilities of shipyards, thus providing extensive scalability.

4. Main Design Features

(a) Power Conversion

Each CMSR liquid fuel fluoride molten salt loop transfers heat generated in the core to a secondary liquid heat-transfer fluoride salt via the primary heat exchanger. The secondary salt circuit then transfers heat to the Power Conversion system via a steam generator, where live superheated steam at up to 18 MPa and 565ºC is produced. The steam produced from the two generators drives a single conventional condensing steam turbine with exhaust positioned axially into the seawater-cooled condenser. Electrical output from the connected generator allows delivery of 200MWe net from each Power Module through a common electrical interface to the high-voltage onshore electrical transmission grid. Through this arrangement, the Power Barges can provide between 200 and 800MWe net to the grid depending on the number of modules installed.

(b) Reactor Core

The reactor core of the CMSR consists of an upper plenum, lower plenum, and fuel tubes which contain and guide the molten fuel salt through the active core, whilst also maintaining separation between the fuel salt and moderator salt. The CMSR consists of 250 fuel tubes that are arranged in a 60° triangular tube pitch pattern with control rod guide tubes inserted in between. Most of the heat is produced in the fuel salt and is the primary heat carrier from the active core to the heat exchanger.

(c) Reactivity Control

In the CMSR design there are two types of control rods used: Regulating Rods for compensating for short-term reactivity changes, and Safety Rods, for extinguishing the nuclear chain reaction promptly when needed. The design also contains a highly reliable, redundant and supplementary method for extinguishing the nuclear chain reaction achieved through draining of the Fuel Salt from the reactor to a Fuel Salt Drain tank.

(d) Reactor Vessel and Internals

The reactor vessel of the CMSR has a height of 5.5 meters and a diameter of 2.5 meters excluding shielding and thermal insulation, the reactor vessel lid has the penetration for the control rods. Inside the reactor vessel, the 2m active core contains the fuel salt in the tubes.

(e) Fuel Characteristics

The fissile component of the fuel is HALEU. The enrichment level is chosen to enable an operating lifetime of 12 years without refuelling. The fuel consists of NaF, KF, and UF4, with partial reduction of UF4 to UF3 to limit degradation of the structural materials. Partial reduction is maintained throughout the operating lifetime by the chemical control system. Most fission products form soluble fluoride species which are retained in the fuel for the lifetime of operation. A fraction of these fission products is insoluble, mostly as noble metals and volatile species such as noble gases. Noble metals are isolated within the Fuel Salt System. Volatile species are continuously removed from the primary circuit during operation through the Fuel Salt Off-gas System; this stream is treated and conditioning through different operation units.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The approach and configuration of engineering safety system relies on a tailored defence in depth, with a top-down focus on the three fundamental safety functions. In addition to the engineering systems, the inherent features of the CMSR are considered with respect to these three fundamental safety functions, assessed, and cascaded down at the defence in depth and plant level safety architecture. The inherent features considered in the conceptual phase and supported by R&D but not limited, are: 1.- Control of reactivity; a) use of gravity to reach a subcritical configuration, b) negative temperature coefficient, c) moderator. 2.- Cool by removal of heat from the reactor and from the fuel/waste storage; a) high melting point, b) volumetric heat capacity, c) passive systems. 3.- Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental Retention of fission products; a) atmospheric pressure operation, b) hydrogen risk elimination.

(b) Decay Heat Removal System

The CMSR Power Barge has a main line and a diverse line for decay heat removal. Both rely on the principle of draining the fuel salt from the reactor and primary circuit into a subcritical geometry fuel salt drain tank using gravity and other passive means. Once the fuel salt is in the drain tanks, the decay heat can be removed through the main active line. If this main active line fails or during a Station Black-out (SBO), a diverse passive line is required to extract the decay heat. This system enables cooling for 72h without active systems/external action.

(c) Emergency Core Cooling System

Considering the atmospheric operating conditions and ability to drain the fuel salts, the concept of ECCS is covered and implemented via the decay heat removal systems.

(d) Containment System

A containment enclosure system (known as the "safety enclosure" in the marine industry) prevents releases to the atmosphere. The containment enclosure is a leak-tight structure and comprises of non-accessible (reactor and drain tank compartments) and accessible compartments (equipment compartments).

(e) Chemical Control System

Chemical control systems are implemented for each of the molten salt circuits. The fuel salt chemical control system maintains partial reduction of UF4 to UF3 and continuously removes fission products from the primary circuit to maintain the redox potential that is necessary to minimize uniform corrosion during normal operation and transients. The moderator salt chemical control system limits corrosion in the moderator circuit using proprietary technology. The secondary salt chemical control system ensures that the secondary coolant chemistry remains within operational design limits.

(f) Spent Fuel Cooling Safety Approach / System

Each reactor in the CMSR Power Barge operates 12 years without refuelling with the drain tanks described in section 5-b fulfilling the role of a spent fuel cooling system after the 12 years of operation and until decommissioning. The fuel salt is allowed to solidify in the drain tanks and passive means of decay heat removal are employed in a similar fashion as dry cask storage for conventional nuclear spent fuel.

6. Plant Safety and Operational Performances

The unique nature of the molten fuel salt drastically reduces the Core Damage Frequency and Large Early Release Frequency. The high boiling point of the molten salt does not require the high pressures typically associated with LWRs. The on-line noble fission products are removed via helium carrier gas and purification. Additionally, as the fuel cools and solidifies, non-noble fission products are retained in the salt, minimizing the release to the environment. Experiments are currently being designed to validate these claims. The 12-year CMSR lifecycle reduces the Aging Management requirements and presents an opportunity to implement a novel maintenance approach. Based on the principles of Reliability Centred Maintenance, efficient scheduling of maintenance activities allows a single CMSR to achieve a Capacity Factor of 90% including a reserve of several weeks for unplanned outages. The 24 years cycle of operation is achieved with the 2nd set of CMSRs.

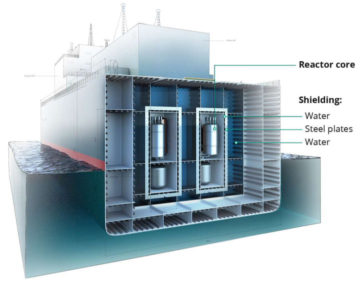
7. Instrumentation and Control System

The Seaborg CMSR Power Barge I&C systems provide automated control, monitoring, and protection functions of the CMSR Power Barge. The Seaborg CMSR I&C architecture reflects the modular nature of the CMSR Power Barge and is designed to ensure separation of safety and non-safety systems. Each CMSR Reactor shall have its own protection system which shall monitor critical parameters and if required shall automatically initiate reactor trip and relevant safety functions to achieve and maintain the CMSR systems in a safe state.

8. Plant Layout Arrangement

The CMSR Power Barge consists of one or more power modules connected to an aft end module (accommodation, offices, conference rooms, main control room, supplementary control room, emergency power, and barge utilities) and a fore end module, where the High Voltage substation shall be positioned. The entire hull is designed with a complete double hull, which protects the safety enclosure, the CMSR auxiliary systems, and the emergency diesel generators, diesel tanks, and switchboards. The CMSR Power Barge is not self-propelled and shall be equipped with suitable towing and mooring facilities.

200MWe Power Module of the CMSR Power Barge



Cross-sectional view

9. Testing Conducted for Design Verification and Validation

In house laboratories verify corrosion control processes in the fuel and moderator salts, including flow loops to enable properties to be verified in a relevant environment. Flow loops and additional custom designed apparatus are used to validate simulation methodologies such as thermal hydraulics. Partnerships established with major institutions enhance verification and validation capabilities, both experimental and modelling. Collaborative projects established to determine: a) physicochemical properties of the fuel salt, b) thermal scattering kernels measured for solid and those for molten NaOH. Irradiation experiments for verification and validation of materials and salts properties are also being established. In addition, in-house robust capabilities to model and simulate in parallel the design are constantly enhanced.

10. Design and Licensing Status

The CMSR is at a conceptual design stage with an emphasis on a research and development approach through the development of test facilities and loops to support licensing of a first of a kind CMSR, simplify design and reduce regulatory risk. The licensing of a first of a kind CMSR focuses on a standard design assessment based on a technology-inclusive international framework and complemented by the ongoing development of risk-informed performance-based approaches. Preliminary engagement is being evaluated with several national regulators in parallel to support international deployment of the CMSR.

11. Fuel Cycle Approach

Strict atmosphere control is always maintained once the fuel has been manufactured to prevent the ingress of impurities such as oxygen and moisture. Fuel is loaded into the drain tanks prior to operation. From here, it is heated to the operating temperature and transferred to the primary circuit. The fuel is enriched to a level which enables operation for 12 years without refuelling. At the end of lifetime, the fuel is returned to the drain tanks where they are cooled. After 24 years of operation, the Power Barge is returned to a specialized dedicated decommissioning facility where further decay heat removal of the drain tanks is ensured followed by reprocessing of the fuel.

12. Waste Management and Disposal Plan

Waste management and disposal approach is flexible and can be adapted to fit national polices of the host state and contractual arrangements with a “greenfield” philosophy. Industry practices are applied to manage and classify waste with higher level wastes (CMSR, spent fuel prior to reprocessing, etc.) predominantly aimed for storage/disposal in dedicated facilities abroad with reprocessing of the spent fuel favored, and lower-level wastes aimed for storage/disposal in the host state country or abroad provided adequate additional provisions.

13. Development Milestones

2020	CMSR Feasibility Statement received by ABS
2022	Barge Approval in Principle by ABS
2024	End of Concept Verification
2026	End of Design
2028	Delivery of first Barge

Copenhagen Atomics Waste Burner (Copenhagen Atomics, Denmark)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Copenhagen Atomics, Denmark
Reactor type	Molten salt reactor
Coolant/moderator	Fuel salt/heavy water
Thermal/electrical capacity, MW(t)/MW(e)	100 / NaN
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.05 – 0.25 / 0.1 – 0.25
Core Inlet/Outlet Coolant Temperature (ºC)	600 / 650 - 700
Fuel type/assembly array	^{7}LiF-ThF_4-$(TRU)F_3$ / none
Power conversion process	Heat source
Fuel enrichment (%)	TRU or RGPu
Refuelling cycle (months)	NaN
Core Discharge Burnup (GWd/ton)	900 – 1000
Reactivity control mechanism	Heavy water level adjustment
Approach to safety systems	Passive
Design life (years)	5 years for the reactor vessel, minimum 50 years for the surrounding building, and infinite life for the salts and heavy water.
Plant footprint (m^2)	2 500
RPV height/diameter (m)	12 / 2.4
RPV weight (metric ton)	40
Seismic design (SSE)	NaN
Fuel cycle requirements / approach	Spent fuel initiated / conversion to Th-U cycle
Distinguishing features	Liquid moderator, Low fissile inventory, and Potential for breeding
Design status	Detailed design / Equipment manufacturing in progress

1. Introduction

The Copenhagen Atomics Waste Burner is an autonomous small single module 100 MW(t) heavy water moderated, fluoride salt based, thermal spectrum, molten salt reactor the size of a 40 feet shipping container. The reactor doesn't require refuelling, human intervention, or maintenance for its 5-year design life and operates completely autonomously with passive decay heat removal. The salts and heavy moderator have, in principle, infinite 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. In this way the Copenhagen Atomics Waste Burner employs three barriers to nature.

2. Target Application

Copenhagen Atomics will design, licence, build, operate, and decommission all Copenhagen Atomics Waste Burners. Each reactor delivers heat as a service, in the form of up to 560°C nitrate salt, to the customer. In order to facilitate early mass deployment, Copenhagen Atomics plans to initially sell to customers that need double digit GW thermal capacity to produce liquid products, such as ammonia production or desalination of 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 burning of spent nuclear fuel transuranics, kickstarting 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, and a lot of the infrastructure needed for operation such as salt purification and data acquisition.

4. Main Design Features

(a) Power Conversion Unit

The Copenhagen Atomics Waste Burner delivers heat as a service and is operated by Copenhagen Atomics. The customer of the heat can choose to couple the reactor heat output with a power conversion system.

(b) Reactor Core

The Copenhagen Atomics Waste Burner uses $^{7}LiF\text{-}ThF_4\text{-}(TRU)F_3$ or $^{7}LiF\text{-}ThF_4\text{-}RGPuF_3$ kickstarter fuel salt, $^{7}LiF\text{-}ThF_4$ blanket salt, and unpressurized room temperature heavy water moderator. Bred uranium from the blanket is transferred to the fuel salt online.

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 air conditioner, consuming a small fraction of the produced heat to maintain the moderator temperature. Employing a blanket salt allows for a compact core while maintaining a low neutron leakage. The reactor core design enables self-sustained breeding in thermal spectrum with the thorium fuel cycle, if fission products are removed online and composite core construction material is employed. Current versions of the Copenhagen Atomics Waste Burner prototype's core uses stainless steel core construction material and fission product separation limited to volatile gases.

(c) Reactivity Control

During normal operation the liquid moderator allows core reactivity control through a simple heavy water level adjustment. Gradual lowering of fissile inventory from transuranic burning period can be compensated by gradually increasing the heavy water level. Gradual build-up of fissile inventory from breeding when predominantly running on thorium cycle can be compensated by gradually lowering the heavy water level.

The highly negative temperature reactivity feedback of the Copenhagen Atomics Waste Burner is relied upon as a second and independent reactivity control mechanism.

The fuel salt, blanket salt, and heavy water moderator are all respectively being continuously and passively drained from the core at a high rate and actively pumped and cooled before going back into the core. If power is shut off or any of the pumps tripped the reactor will shut down, due to the lack of fuel, moderator, and or reflector blanket.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The entire reactor operation and shutdown is designed to be completely autonomous with decay heat removed passively. Safety is ensured through inherent safety instead of operator action.

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

The fuel salt is pumped through the dump tank so that in the case of a pump trip the fuel salt drains quickly to the dump tank, where the decay can conduct through the tank bottom and through the primary radiation shielding to the surroundings. This enables completely passive decay heat removal but also means that there's always a parasitic heat loss through the same heat path during normal operation. This is an economic trade-off for enhanced passive safety.

(c) Containment System

The reactor containment consists of three physical barriers to nature. The first barrier is the fuel salt and fission product trap wetted structure and components. The second barrier is the leak tight 40 feet size vessel that surrounds the first barrier, reactor core, reactor coolant salt, and other components. The third barrier is an exterior onsite constructed leak tight containment that also serves as radiation shielding. Another path is through the reactor coolant and secondary coolant salt which is served to the customer. Here the three heat exchangers serve as three physical barriers, first the fuel salt to reactor coolant salt heat exchanger, second the reactor coolant salt to secondary coolant salt, both of these are located inside the 40 feet sized vessel, and third the heat exchanger or boiler on the customer side.

(d) Spent Fuel Cooling Safety Approach / System

If online fission product removal is employed then the fuel salt can be transferred to a new reactor unit, at the end of life or after failure of one unit. If only online volatile fission product removal is employed then intermediate salt cleaning will be necessary before transfer to a new unit. Thus, due to the continuous use of the fuel there is no spent fuel. Fission product traps are envisioned to be left in the used reactor vessel, where their decay heat can be passively removed through the normal decay heat removal system for a cool down period before decommissioning. In the case of a failure of the salt wetted barrier where the fuel salt can't be transferred through the normal means then it would be left for a cool down period before decommissioning, where the fuel would be separated, cleaned, and reused.

6. Plant Safety and Operational Performances

The Copenhagen Atomics Waste Burner's safety is not dependent on the use of the heat that is delivered to the customer. The reactor doesn't require refuelling, human intervention, or maintenance for its 5-year design life and operates completely autonomously with passive decay heat removal. A plant is envisaged to have dozens of reactor units, placed together and inaccessible during operation, besides for swapping of reactor vessels under observation. Satisfactory safety statistics for commercial operation will be determined from reactor prototype operation.

7. Instrumentation and Control System

Copenhagen Atomics makes/writes all of our own electronics and software for the Copenhagen Atomics Waste Burner. The system uses redundant consumer grade electronics running consensus steering for autonomous operation of the reactor throughout its 5-year lifetime. All instrumentation and control systems are sealed inside the 40 feet sized reactor vessel 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 2500m^2 per 100MW(t) reactor. 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 two dozen pumped molten salt loops, four dozen static molten salt test systems, including some with fertile LiF-ThF_4 salt and some with LiF-ThF_4-PuF_3 salt test slated for 2022-2023, two 1MW non-fission prototype reactors are under construction to run with water and FLiNaK respectively, and multi ton scale fertile salt production is under construction. Both pumped molten salt loops with fuel salt and 1MW non-fission prototype reactors with fertile salt are slated for 2023-2024.

10. Design and Licensing Status

Copenhagen Atomics has started licensing for production of multi digit tonnes of thorium and natural uranium for testing of non-fission prototype reactors. Copenhagen Atomics is interacting with nuclear safety authorities on testing of a 1MW prototype reactor. Copenhagen Atomics does not intend to engage in a commercial licensing process for a 100MW(t) reactor before a prototype reactor has been tested.

11. Fuel Cycle Approach

The Copenhagen Atomics Waste Burner uses ^{7}LiF-ThF_4-(TRU)F_3 or ^{7}LiF-ThF_4-RGPuF_3 fuel salt with 3% mol (TRU)F_3 or RGPuF_3 kickstarter fuel salt and ^{7}LiF-ThF_4 blanket salt. The reactor can transition to a breeder reactor using the heavy water level for coarse reactivity adjustment. The reactor core design enables self-sustained breeding in the thermal spectrum and thorium fuel cycle if fission products are removed online and C/C composite core construction material is employed. The Copenhagen Atomics Waste Burner needs an initial load of roughly 100-200 kg of TRU or RGPu and doesn't require any refuelling for its 5-year operation.

The Copenhagen Atomics Waste Burner design can also start on roughly 1 ton 5% enriched uranium using a ^{7}LiF-UF_4 fuel salt and ^{7}LiF-ThF_4 blanket salt, and can achieve a conversion factor of around one within three years of operation. After 5-year operation the burnup is roughly 200 GWd/tU and most of the fissile uranium in the fuel salt is ^{233}U that was produced in the blanket. The fuel and blanket salt can be reused in subsequent reactors without adding makeup fuel.

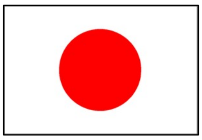
12. Waste Management and Disposal Plan

The Copenhagen Atomics Waste Burner can transition to breeding within the first three years of operations with online removal of fission products and C/C 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, infinite lifetime and will be reused in subsequent reactors, while the reactor vessel is the consumable and fission products are left as a by-product.

After operation the reactor vessel and fission product trap are left to cool down before decommissioning. For decommissioning it’s envisaged to separate fission products for vitrification or other uses and remelting of the vessel for non-nuclear reuse with low remaining activity and non-metals separated in the slag.

13. Development Milestones

2015	Copenhagen Atomics was founded in 2015 by a group of passionate engineers and scientists meeting up since 2013 and based on an open-source model, where results and findings are shared with the thorium molten salt reactor community.
2015	First simulations of neutron economy and online fission product removal.
2016	First static molten salt test.
2017	First pressure driven circulation molten salt loop.
2018	First pumped circulation molten salt loop.
2019	Award two Danish research grants and start of collaboration with Alfa Laval.
2020	First public funding round and first fertile salt test.
2022	Completion of non-fission prototype of a 1MW(t) demonstration reactor.
2025	First test of a 1 MW(t) demonstration reactor.
2028	First test of a 100 MW(t) commercial reactor.



FUJI (International Thorium Molten-Salt Forum, Japan)

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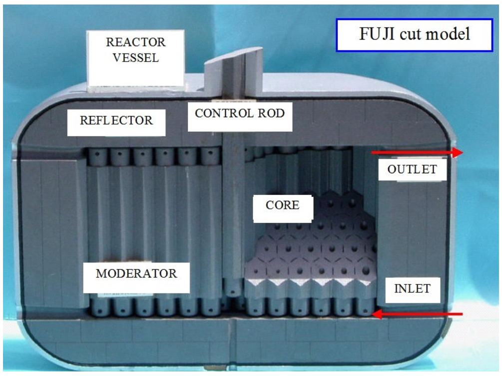
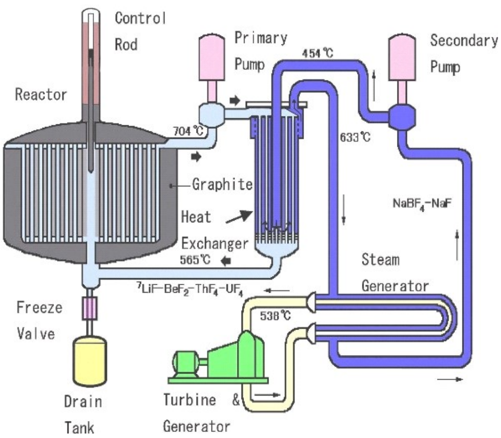
MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ITMSF, Japan
Reactor type	Molten salt reactor
Coolant/moderator	Molten fluoride/graphite
Thermal/electrical capacity, MW(t)/MW(e)	450 / 200
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.5 / 0.5
Core Inlet/Outlet Coolant Temperature (ºC)	565 / 704
Fuel type/assembly array	Molten salt with Th and U
Number of fuel assemblies in the core	N/A
Fuel enrichment (%)	2.0 (0.24%^{233}U + 12.0%Th). Pu or LEU can be used
Core Discharge Burnup (GWd/ton)	No mechanical limit for burnup
Refuelling Cycle (months)	Continuous operation possible
Reactivity control mechanism	Control rod, or pump speed, or fuel concentration
Approach to safety systems	Passive
Design life (years)	30
Plant footprint (m^2)	<5000 (RB+SGB+TGB)
RPV height/diameter (m)	5.40 / 5.34 (inner)
RPV weight (metric ton)	60 (made of Hastelloy N)
Seismic Design (SSE)	Same as LWRs
Fuel cycle requirements / Approach	Self-sustaining at FUJI-U3. No online reprocessing, except gaseous FP removal. Spent fuel salt is reprocessed
Distinguishing features	Self-sustaining at FUJI-U3. Spent fuel salt is reprocessed at off-site facility.
Design status	3 experimental MSRs were built. Detailed design not started



MSR-FUJI cut model



Schematic diagram of MSR-FUJI

1. Introduction

The Molten Salt Reactor (MSR) was originally developed at Oak Ridge National Laboratory (ORNL) in 1960s, and three experimental MSRs were constructed. One of them was operated for 4 years without severe problems. Thus, it is verified that the MSR technology is feasible. MSR-FUJI was developed since the 1980s by a Japanese group (International Thorium Molten-Salt Forum: ITMSF), based on the ORNL's results to deploy it in the world.

Molten salt is stable and inert at high temperature and can be used at very low pressure. Since core meltdown or steam/hydrogen explosion is impossible, high safety can be achieved. MSR-FUJI is size-flexible as from 25 MW(e) to 1000 MW(e). But, a latest and typical design (FUJI-U3) is 200 MW(e), which can be categorized as small-sized reactors with modular designs (SMR). The thermal output of FUJI-U3 is 450 MW(t) and thus a 44% thermal efficiency can be attained. In addition, the simple core structure and high fuel efficiency should facilitate a favourable economic performance.

Molten fuel salt can contain thorium (Th) as fertile material and ^{233}U as fissile material, and the FUJI-U3 design can attain a self-sustaining fuel cycle with a conversion factor of 1.0. Since MSR-FUJI applies the Th-cycle, generation of plutonium (Pu) and minor actinide (MA) is very small compared with Light Water Reactors (LWR). Furthermore, it can consume Pu, and can thus contribute to reduce the proliferation risk caused by Pu from LWR spent fuel. It can also be used to transmute long-lived MA to shorter ones.

2. Target Application

MSR-FUJI can be applied not only to electricity generation, but also to transmutation of Pu and/or MA. Besides these purposes, it can be used as a heat source for water supply by desalination of seawater or for hydrogen production, utilizing its high exit temperature of 704°C.

3. Design Philosophy

The design philosophy of MSR-FUJI is to achieve a high level of safety, good economic performance, contributing to non-proliferation, and to achieve fuel cycle flexibility.

MSR-FUJI is based on the ORNL's results, and has been optimized as a small sized plant and further simplified by removing the online reprocessing facility. Based on the operating experience at three experimental MSRs in ORNL, it has been verified that MSR-FUJI is feasible. The steam generator (SG) is however a major unverified component but it can be developed based on Fast Breeder Reactor (FBR) experience and the recent supercritical power station technology.

MSR-FUJI adopts a passive safety system to improve the safety, reliability as well as the economics. Molten fuel salt can be drained to a sub-critical drain tank through a freeze valve. Since gaseous fission products (FP) are always removed from molten fuel salt, the risk at accidents is minimized. MSR-FUJI is operated at very low pressure (0.5 MPa), and a thick reactor vessel and pipes are not required. There are no fuel assemblies or complex core internal structure, with the only component of graphite moderator within a reactor vessel. Based on these design principles, in-factory fabrication would be simple.

4. Main Design Features

(a) Power Conversion

The nuclear steam supply system (NSSS) consists of a reactor core, pipes, pumps, a heat exchanger (HX), and a steam generator (SG), which supplies steam to a turbine/generator (T/G). The schematic diagram of MSR-FUJI shows only one loop, but a loop can be redundant depending on a plant size or a need for flexibility.

MSR-FUJI is designed to produce an exit temperature of 704°C in molten fuel salt, and its heat is transferred to the secondary salt through a HX. Then, its heat produces 538°C supercritical steam of 252 kg/cm^2 at a SG, and generates electricity by a supercritical T/G. Owing to its high temperature, MSR-FUJI can achieve 44% thermal efficiency.

The primary loop (molten fuel salt loop) is operated with forced circulation by a centrifugal pump during normal operation. The system also has a natural circulation capability in emergency conditions.

(b) Reactor Core

The core structure is made of hexagonal shaped graphite moderator blocks. The blocks contain holes that serve as the flow paths of the molten fuel salt that flow upwards through the blocks circulated by the primary pump. The molten fuel salt then goes to a heat exchanger to transfer the heat to the secondary coolant salt.

The concentration of the fuel composition can be adjusted at any time through the fuel concentration adjustment system. Since there are no fuel assemblies in the core, refueling shutdown is not required, and continuous operation is possible. In order to achieve a core conversion factor of 1.0, it is recommended to refresh the fuel salt every 7 years. Periodic maintenance shutdown will be required as in any power plant

(c) Reactivity Control

Reactivity control for long-time operation can be performed anytime by a fuel concentration adjustment system. In normal daily operation, reactivity or power level can be controlled by core flow or by core temperature. Control rods are withdrawn in normal operation and are inserted by gravity in case of emergency shutdown.

(d) Reactor Vessel and Internals

The reactor vessel is made of Hastelloy N. Since the operating pressure is very low (0.5 MPa) a 'pressure vessel' is not required. The only one core internals is the graphite moderator blocks.

(e) Fuel Characteristics

The molten fuel salt is a liquid form of fluoride (LiF-BeF_2) with ThF_4 and a small amount of $^{233}UF_4$. A typical composition is LiF-BeF_2-ThF_4-$^{233}UF_4$ (71.76-16-12-0.24 mol%).

Molten fluoride can be used at very low pressure owing to its very high boiling temperature and very low vapor pressure. The melting temperature of the above fuel composition is 499°C. It can dissolve uranium (U) or Pu as fissile material so that low enriched uranium (LEU) or Pu can be used. Fuel assembly fabrication is not required for molten fuel salt, and radiation damage or fuel cladding failure does not occur.

5. Safety Features

(a) Engineered Safety System Approach Configuration

In case of pipe break, leaked molten salt is drained to an emergency drain tank without passing through a freeze valve. A pressurization accident is very unlikely owing to its low vapor pressure. Therefore, an emergency core cooling system (ECCS), containment cooling system (CCS), makeup water pools, and automatic depressurization system (ADS) are not required. In order to protect against a freeze accident in a molten fuel salt loop, a high temperature containment is equipped with heaters.

(b) Decay Heat Removal System

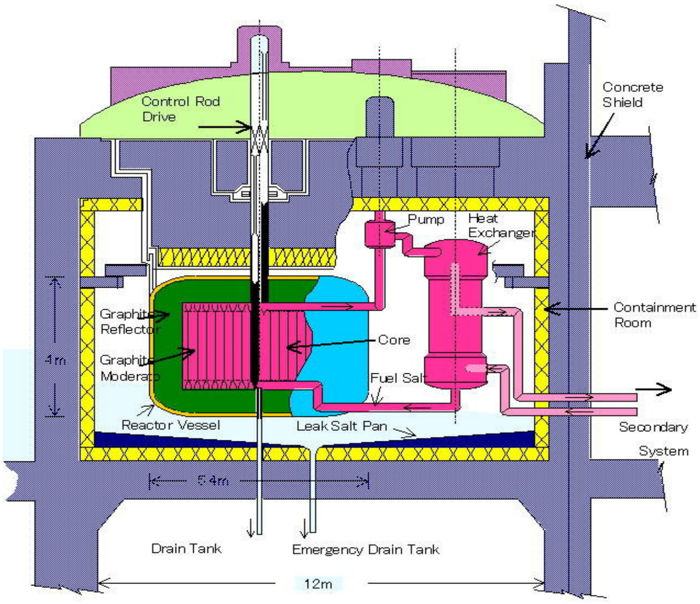
In normal shutdown condition, decay heat is transferred to a secondary loop and a steam-line loop, and disposed to the ultimate heat sink (seawater etc.). If all pumps in a primary or secondary loop stop, fuel salt is drained to a drain tank through a freeze valve. Decay heat at the drain tank is cooled by a passive heat removal system, and finally its heat is disposed to the outside environment through an air-cooled system that does not require electricity.

(c) Emergency Core Cooling System (ECCS)

As is explained above, redundant and diverse ECCS and makeup water pools are not required. This would simplify the plant, and eliminate concerns of failures in safety systems,

(d) Containment System

Since the risk of pressurization accidents is very unlikely, the containment size can be minimized. Although molten salt is not flammable, inert gas (N_2) is enclosed within a containment in order to maintain fuel salt purity in case of a pipe break accident. The MSR-FUJI design has 3 levels of containment. The 1st is the reactor vessel and pipes made of Hastelloy N. The 2nd is a high temperature containment composed of three layers, which contains a reactor vessel, pipes, and a heat exchanger. In order to avoid a freeze accident, this containment is equipped with heaters. The 3rd level is a reactor building composed of two layers. As explained above, a pressurization accident is very unlikely due the low vapor pressure. Therefore, a containment cooling system and makeup water pools are not required.



Vertical cross-section of primary system of MSR-FUJI

6. Plant Safety and Operational Performance

Overall safety is described above. In case of a station blackout (SBO: Loss of all AC electricity) the MSR-FUJI can be shut down and cooled without electricity. Core meltdown or steam/hydrogen explosion is physically excluded by design, and no ECCS is needed. Long-time and daily operations are described in Section-3(d). Based on those features, load following is performed without control rods.

7. Instrumentation and Control Systems

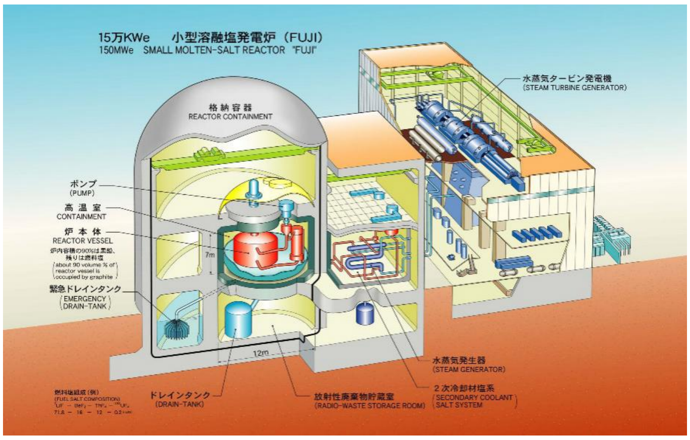
Instrumentation and control (I&C) systems in the MSR-FUJI design are the same as recent LWR designs. It must support operators in making decisions and efficiently operating the plant during plant start-up, shutdown, normal operation, surveillance testing, and accidental situations. It adopts the man-machine interface more useful, and expands the scope of automatic control.

8. Plant Layout Arrangement

Major buildings of MSR-FUJI are a reactor building, a SG building with a main control room, and a T/G building.

(a) Reactor Building

The reactor building contains a high temperature containment, drain tanks, a radio-waste storage, and other facilities required for the reactor. This reactor building is a cylindrical shape with a hemispherical dome, which is made of concrete with steel liner as its inner layer. The reactor building is founded on a common base-mat together with other buildings.



Bird's-eye view of MSR-FUJI

(b) Control Building

The main control room (MCR) is located at a SG building, which is next to a reactor building. The MCR is a key facility to cope with normal and emergency situations, so it is designed to ensure that plant personnel successfully perform the tasks according to the proper procedures.

(c) Balance Plant

Turbine Generator (T/G) contains the supercritical T/G to produce electricity. Also, it contains condensers for disposed steam, which use outside water (sea water etc.) for cooling. The electric power systems include the main generator, transformers, emergency diesel generators, and batteries, besides external power lines. In case of station blackout, it can be shut down and cooled without electricity.

9. Testing Conducted for Design verification and Validation

MSR-FUJI is based on 3 experimental reactor experience. But some of technologies such as a supercritical SG or remote maintenance equipment have to be established.

10. Design and Licensing Status

Preliminary designs for various applications have been completed [1]. Three experimental MSRs were constructed, and one of them was operated for 4 years without severe problems. Although the detailed design is not yet started, safety criteria and guidelines for MSR licensing are proposed with numerical results for major accident analysis [1].

11. Fuel Cycle Approach

As explained in Section-1 and 3, FUJI-U3 design can attain a self-sustaining fuel cycle. MSR-FUJI is simplified by removing the online reprocessing facility. Only gaseous fission products (FP) are always removed from molten fuel salt. Spent fuel salt is discharged, and reprocessed at the off-site pyro-reprocessing facility. Some usable actinides can be sent back to the reactor.

12. Waste Management and Disposal Plan

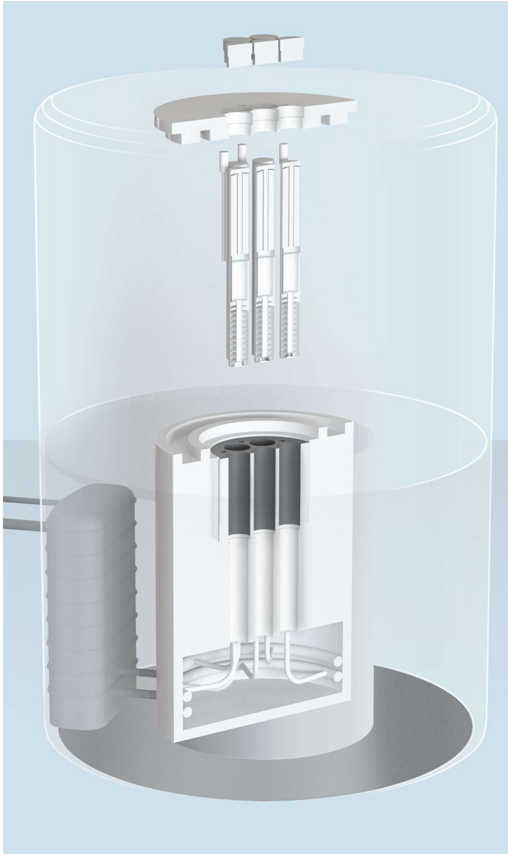
Actinides such as U/Pu/Th/MA separated at the off-site reprocessing facility are recycled to MSR. FPs and salt are stored at disposal facility. These facilities are to be developed in parallel with MSR deployment.

13. Development Milestones

1980's	Conceptual designs of MSR-FUJI have been started
1980's	Accelerator Molten-Salt Breeder (AMSB) design for a large production of fissile material (similar to Accelerator Driven System ADS)
Until 2008	Designs such as a pilot plant (mini-FUJI), a large-sized plant (super-FUJI), a Pu-fuelled plant (FUJI-Pu).
Recent	The latest SMR plant (FUJI-U3).

[1] Yoshioka, R., Kinoshita, M. "Liquid Fuel, Thermal Neutron Spectrum Reactors", Chapter-11 of the book "*Molten Salt Reactor and Thorium Energy*", Elsevier Inc., USA, 2017(1st edition), 2022(2nd edition).

THORIZON (THORIZON B.V., Netherlands)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	THORIZON, Netherlands
Reactor type	Modular core molten salt reactor
Coolant/moderator	Molten salt/graphite (no salt-graphite contact)
Thermal/electrical capacity	100-300 max MW_{th} 550 C steam/ 40-120 max MW_{el}
Primary circulation	Forced circulation
Primary system operating pressure	2 bar max at shutdown – 8 bar max under operation
Core inlet/outlet coolant temperature (ºC)	500 / 800
Fuel type/assembly array	Fissile-fertile material bearing molten salt in closed containment modules
Number of fuel assemblies/modules in the core	3-7 (current designs, can change)
Fuel	Plutonium-Thorium
Refuelling cycle	5 – 10 years target
Reactivity control mechanism	Control rods, burnable poison, core draining and strong negative temperature feedback
Approach to safety systems	Passive as much as possible
Design life (years)	full core replacement strategy allows life extension beyond 60 years)
Fuel cycle requirements/approach	LWR-Plutonium burning, Thorium breeding-burning
Distinguishing features	Nuclear safe core material replacement strategy, with external core module series production, continuous improvement by module updates, and fuel cycle flexibility
Design status	Conceptual design

1. Introduction

Thorizon develops a novel molten salt reactor design, based on a modular core approach, in which the core consists of multiple individually contained replaceable modules. When the modules are removed, the reactor has no primary circuit and no primary circuit components. The reactor has no fixed large reactor pressure vessel. The technology basis is flexible and allows larger and smaller systems, either by larger or smaller modules, or by more or less modules. The technology offers fuel cycle flexibility, with first goal to burn LWR-plutonium and commercialize, which can be followed by advanced versions on the same technology basis in which fuel cycles can be closed. Thorizon has finalised the conceptual design phase, has a positively evaluated patent, and successfully passed technical due diligences by third parties, and has secured the finances to enter the detailed design phase and execute a core material irradiation program.

2. Target Application

THORIZON offers a technology base for a multitude of different molten salt reactor designs. First goal is to maximize efficient LWR-Plutonium burning within the shortest route to deployment, to reduce the long-lived nuclear waste burden, and turn waste cost into energy generation income. The technology basis combined with the demonstration and commercialisation of Plutonium burners, could allow and support further expansion towards a closed Thorium cycle, or fast spectrum options.

3. Design Philosophy

The largest challenge in molten salt reactors is related to the degradation of core materials. Thorizon's design tackles this challenge by allowing convenient and safe replacement of core materials, while maintaining containment. This approach has led to a design that offers additional advantages, in terms of flexibility, plant lifetime, improvement implementation over the plant lifetime, cost, modularity/series production and practical management of spent fuel and degraded core materials. Another key aspect of the design approach is to avoid complexity, eliminate proliferation risk concerns, and make sure that the design safety and performance can be assessed by basic computational tools, facilitating validation and licensing.

4. Main Design Features

(a) Power Conversion

Thorizon systems produce 550°C supercritical steam in a tertiary circuit, connected to a molten-salt based secondary system by heat exchanger, connected to the primary loops of the modules, each with their own heat exchanger. The steam can be used directly in industrial processes or can be used to generate electricity in a steam turbine. The secondary-tertiary system is largely based on existing technology from concentrated solar power applications.

(b) Reactor Core

The reactor core consists of the upper sections of the modules, which when placed together form a critical zone. Each individual module is subcritical, unless placed in a critical configuration with other modules and moderator/reflector. For IP protection and export control reasons, Thorizon does not disclose information on design geometry and materials.

(c) Fuel Characteristics

Thorizon aims to adopt LWR-plutonium and Thorium as a fissile-fertile fuel mixture. Information on exact salt composition is regarded IP and export control sensitive.
Based on successful implementation and commercialisation, advanced modular core options can be considered, adopting different fuel cycles.

(d) Reactivity Control

Reactivity control is performed with burnable poison, control rods and passively via large negative temperature feedback. The latter is a specific benefit of molten salt reactors, and utilised in the Thorizon design. To achieve a long cycle, the initial over-reactivity is reduced by burnable absorber in replaceable core materials.
Reactivity is quickly reduced in case module pumps stop, as this leads to immediate core draining. Control rod insertion, also renders the reactor subcritical.

(e) Reactor Pressure Vessel and Internals

Thorizon has no reactor pressure vessel, but a radiation shield which also serves as external impact barrier, in which modules are placed that each have their own two containments. Each module contains its own pump and heat exchanger, and is closed.

(f) Reactor Coolant System

The primary circuit consists of individually contained modules, in which primary fissile-fertile material bearing molten salt is circulated, for each module individually. The number of modules can be varied, as can the size of the module, leading to different system powers. Each of the modules is connected to the same secondary cooling system.

(g) Secondary System

The secondary system is a central molten salt system, which connects to each module where the salt is heated, and connects to the heat exchanger of the steam generator where it is cooled down. Optionally the secondary system includes a heat storage system.

(h) Steam Generator

The secondary salt transfers the heat to a steam generator, 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.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Molten salt reactor technology offers passive safety features, which are maximally exploited by the Thorizon design. In addition, the low pressure in the primary system, which can be secured under all circumstances, reduces the stresses on primary components, and excludes having a significant force to release materials to the outside. Defence in depth is incorporated by adopting multiple containments.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

Thorizon systems adopt an active secondary (molten salt)-tertiary (water/steam) coolant system, and a passive (inert gas) core cavity cooling system.

Decay heat in a station blackout scenario is passively removed from core modules by gas flow of the core cavity coolant system along the modules, which has adequate capacity avoiding temperature escalation in the core modules. The system power, and component sizes, cooling surfaces and core cavity cooling system have been designed for that purpose. The core cavity cooling system also passively cools the outside of the modules and the moderator and reflector in between and around the modules, during operational conditions.
Considering this is worst case, adequate cooling can always be maintained via this system, or via the normal secondary-tertiary cooling system under normal or off-normal operating conditions.

(c) Spent Fuel Cooling Safety Approach / System

Temporary onsite storage of the core modules 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.

(d) Containment System

Defence in depth is managed by multiple barriers, taking credit for the absence of primary circuit pressure escalation scenario's under all circumstances:
- Molten salt matrix (solid or liquid)
- First and second containment for each core module
- First and second containment for the secondary coolant circuit
- Irradiation shield and external impact barrier surrounding the modular core
- Underpressurized building enclosure

(e) Chemical Control

Chemical control is passive, no elements are introduced or extracted from modules during operation. Chemical control methodology cannot be disclosed at this time, to protect IP.

6. Plant Safety and Operational Performances

Core damage frequency has not been determined yet. It is however to be noted that pressure escalation in the primary containments can be excluded, with a secondary containment to avoid release in case of primary containment failure.
Passive safety features, passive reactivity control and external impact barriers, largely reduce core damage and release probability to very low levels.
Station black-out scenario is managed by passive cooling as well, providing indefinite grace time.

7. Instrumentation and Control System

Plant detailed design is under development. Reactor control and shutdown systems have redundancy. Instrumentation relates to temperatures and pressure in primary, secondary and tertiary circuits.
Activity monitoring of secondary and tertiary system, and the core cavity cooling system, and of the space in between primary module containments.
Flux measurement in the moderator-reflector.
Chemical potential of the primary salt is foreseen to be monitored as well, as measure of primary salt corrosiveness and changes thereof.

8. Plant Layout Arrangement

Without modules present, the plant has no primary circuit or primary circuit components in the plant, and can therefore ne considered largely conventional. Primary system components are manufactured in series and fuelled outside the plant.
At this stage, the purification and conditioning of primary salt is excluded, hence the plant has no salt reprocessing equipment or facility on site, other than what is included inside modules.
Current design efforts focus on the primary system, core cavity cooling, reflector-moderator and radiation shielding, to be expanded first to the secondary and tertiary cooling and heat transfer systems. Enveloping building design and control room location and design is under development.

9. Testing Conducted for Design Verification and Validation

An irradiation program of core materials has been initiated. This program will provide material properties as function of irradiation damage and temperature, as a basis for safety and performance analyses, and to determine expected module lifetime.
Primary system thermalhydraulic calculations are to be validated with scaled mock-up and a test loop (both under development).
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 final design phase is finalized.
The design has been analysed by thermalhydraulic system code and deterministic and probabilistic core physics codes, reproduced by a qualified external party, confirming no safety showstoppers.

10. Design and Licensing Status

Interaction with Nuclear Safety Authority has started; Site selection is underway, not to be disclosed at this stage. License to build is targeted for 9 years from now, license to operate 14 years from now. It is Thorizon's aim to accelerate this schedule by co-development with industry as soon as possible, to initiate component manufacture, optimisation and qualification quickly.

The current phase is intended to be executed by a dedicated in house team of 15 people, supported by external consultants for specific subjects, for which the funding has been secured. Thorizon aims to establish partnerships and to raise additional funding, to support acceleration of the current schedule. Thorizon prefers not to share conceptual technical information in much detail, due to IP protection and export control considerations. The image shown on the first page is a high level, schematic visualisation, showing the module concept and approach, without disclosing technical detail.

11. Fuel Cycle Approach

Modules are replaced as a whole, and salt maximum burn-up is matched by core material maximum lifetime. A 5-10 year cycle is foreseen (with reactor at maximum power 100% of the time). Reactor criticality can match these cycle lengths with the fuel and core design foreseen, with material lifetime confirmation required by material test irradiations.

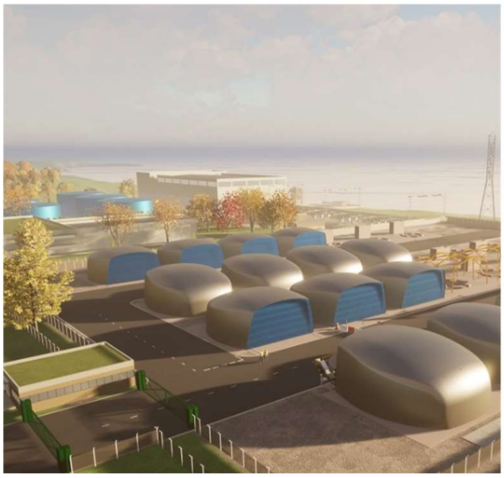
12. Waste Management and Disposal Plan

By design, Thorizon systems do not release liquid or solid radioactive waste during normal operation. After use and cooling, the modules can be transported in shielded casks of manageable dimensions, to be transported to waste treatment facility, where the salt is extracted for separation and purification and recycling, the core materials are cleaned and prepared for medium term storage. Economics and feasibility of ex-core sections (low-medium level waste) recycling has not yet been determined.

13. Development Milestones

2020	Conceptual design phase, technical due diligence successful, patent positively evaluated, investment secured
2022	Transition from phase 1 conceptual design to phase 2 final design
2025	Phase 2 final design finalised, all safety evaluations and PSA successful, license application submitted, core material irradiation test program finalised, site selected, salt experimental program initiated, start of phase 3
2030	Finalisation of Phase 3. Salt experimental program finalised, salt manufacturing ready, transport casks ready. Components qualified, site prepared, module non-nuclear prototype demonstration successful, license to build, Phase 4 construction starts.
2035	Construction finished, license to operate
>2035	Initial operation experience, commercialisation

Stable Salt Reactor - Uranium (SSR-U) (Moltex Energy, United Kingdom)



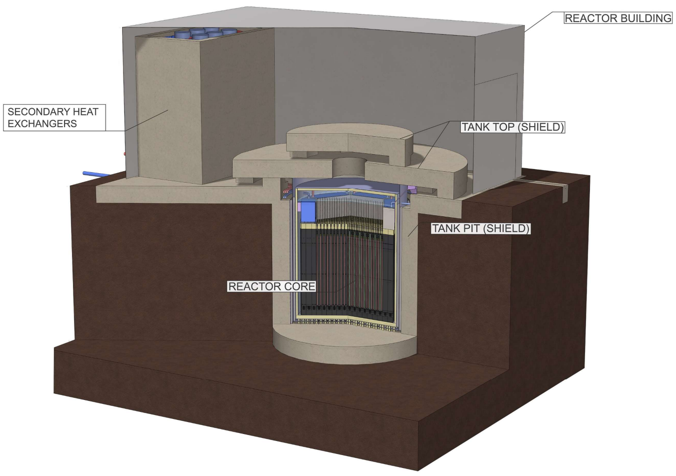
MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Moltex Energy, United Kingdom
Reactor type	Molten Salt Reactor
Coolant/moderator	Molten eutectic fluoride coolant salt / solid commercial grade graphite moderator
Thermal/electrical capacity, MW(t)/MW(e)	Single unit – 40 / 16, deployable in arrays
Primary circulation	Natural convection loop
NSSS Operating Pressure	Primary and secondary coolants atmospheric / unpressurised (0.1 / 0.1)
Core Inlet/Outlet Coolant	725 / 795
Fuel type/assembly	Molten salt fuel within vented fuel tubes
Power conversion process	Molten salt to steam to turbine
Fuel enrichment (%)	6
Refuelling cycle	240
Core Discharge Burnup (GWd/ton)	70
Reactivity control mechanism	Strong fuel temperature coefficient, liquid neutron absorber thermometer
Approach to safety	Hazard reduction, inherent/passive safety
Design life (years)	60
Plant footprint (m^2)	1 unit ~100, 32units (0.5 GW(e)) ~ 10,000
RPV height/diameter	7.5 / 6 (low pressure)
RPV weight (metric ton)	Empty ~100 / Filled ~500
Seismic design (SSE)	EUR standard - peak ground acceleration of 0.3g horizontally and vertically
Fuel cycle requirements / approach	Once-through fuel cycle. All fuel replaced together, 3 cycles total
Distinguishing features	Low pressure molten salt reactor system; ~20 years fuel cycle; ~700°C heat output; Thermal
Design status	Basic design

1. Introduction

Moltex has the technology to transform the energy landscape, ushering in a new age of clean, low-cost energy for all. Moltex's Stable Salt Reactor-Uranium (SSR-U) is an innovative modular molten salt reactor, with each reactor module providing 40 MW(th)/16 MW (e). Its simple design means that most components can be factory-produced and assembled on site, enabling a rapid roll-out of the technology at scale and pace.

The SSR-U's unique attributes include; ~ 20 years fuel cycle using Low Enriched Uranium (6%), high temperature heat output (700ºC) which enables thermal energy storage in molten salt and offers high quality process heat including thermo-chemical production of synthetic fuels.

The reactor is fuelled for a life of 16 full power



Schematic of an SSR-U Reactor single unit years at commissioning, with a further 2 fuelling cycles possible, giving the SSR-U one of the longest refuelling periods (comparable to HALEU reactors), and resulting in a total reactor lifetime of circa 60 years. The simplicity of the design, low pressure operation, inherent and passive safety features, in combination with the refuelling requirements result a low capital and low operating cost reactor.

2. Target Application

The reactor is flexible in its application, it is suitable for grid connection or standalone electricity and heat supply. The high temperature output can be used to provide economical thermal energy storage in molten salt, known as GridReserve®, which can then be despatched via normal steam and turbine generator arrangements to meet fluctuations in demand, for example due to the intermittency of renewable energy supplies.
It 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, for 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

Moltex' 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 CO2 emissions. Cost reduction is achieved through hazard reduction and elimination, and the use of inherent and passive features to deliver safety.
Moltex' 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 LEU, and its modular construction.

4. Main Design Features

(a) Power Conversion

Each SSR-U reactor provides output of 40 MW(t) as hot salt at 700ºC, which equates to ~ 16 MW (e). 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. 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 SSR-U reactor systems to operate with different sizes of GridReserve® means that different energy load profiles are readily accommodated.

(b) Reactor Core

The SSR-U is a fluoride salt reactor with separate fuel and coolant salts. The core comprises an array of fuel tubes in a graphite matrix which fills most of the tank. Each tube sits in a separate channel and molten salt primary coolant 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 is contained in vented fuel tubes. 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 has a fundamental impact on the concept of core damage. In a conventional reactor, core damage can result in the reactor moving to a substantially more hazardous state. In the SSR-U, similar scenarios are tolerable since even severe core damage moves the reactor to a safer state. Breach of the tube wall, for instance caused by the tube melting, 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 an increase in the probability of radiation exposure of workers or the public.

(c) Reactivity Control

In the SSR-U, the bulk of the excess reactivity present immediately after fuelling is neutralised through fixed burnable absorber in inserts within the graphite matrix. Any residual excess reactivity is neutralised by pellets of neutron absorber which are soluble in the coolant salt and 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 hot coolant salt exiting the top of the fuel tube channel and the stem extends through the graphite matrix towards the centre of the core. An increase in heat demand from the reactor causes a drop in primary coolant temperature and as a result the bulb of the thermometer cools, causing the neutron absorber to be withdrawn from the stem of the thermometer and increasing reactivity and power output. Conversely, when heat demand is reduced, the temperature of the primary coolant rises, heating the thermometer bulb, driving the neutron absorber through the stem, and thus, reducing reactivity and power

(d) Reactor Vessel and Internals

The reactor vessel for the SSR-U comprises an outer and inner tank, separated by a refractory layer. The outer tank constructed from a stainless steel will serve as the primary 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 700oC, and will be 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 concrete shield.

(e) Fuel Characteristics

The fuel is low-enriched uranium fluoride salt (6% enrichment), which is contained at low pressures by the patented vented fuel tube.

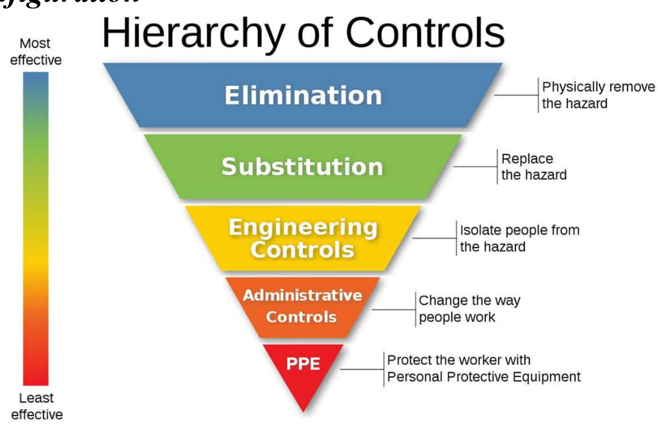
(f) Fission Product Management

Many of the fission gases are immediately captured as non-volatile salts and remain contained in the fuel salt within the fuel tube. The major gas releases are the Xe and Kr noble gasses and the great majority of the radioactive species decay within the tube and bubbler.

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.



Internationally recognised Hierachy of Controls

(b) Decay Heat Removal System

The decay heat removal system is completely passive and utilises natural convection with no need for mechanical components. Utilising an arrangement of perforated ducts, 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 be required for the SSR-U as the design ensures that the core is never exposed by the coolant, and decay heat is removed by convection as previously explained.

(d) Containment System

Containment is progressivley provided by the fuel tube, the coolant salt, the inner tank, the refractory insulation of the tank, the outer tank, and the concrete bilogical shield.

(e) Chemical Control

The SSR-U 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 metal-fluoride diluent. The redox buffer enables maintenance of redox potential and mitigation of potentially corrosive fission products generated through life. Trials to demonstrate suitability of fuel chemistry are underway. The primary and secondary coolant are both fluoride eutectics. Tritium production is mitigated by eliminating Lithium from the chemistry. Similarly, Beryllium is also eliminated to prevent highly corrosive and toxic environments. Both coolant salts are compatible with graphite and mix safely with the fuel salt. Corrosion of the reactor components by the coolant salts is managed using reducing agents. Some corrosion testing of candidate fuel mixtures has already been carried out with plant materials at representative reactor temperatures.

(f) Spent Fuel Cooling Safety Approach / System

Once the SSR-U's fuel is depleted, a cooldown operation occurs before the fuel is removed from the reactor. During cooldown, 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 SSR-U can ramp up and down in output, but Moltex considers the use of GridReserve® a better option for the provision of flexible electricity generation, with the turbine hall providing more responsive ramp rates. The SSR-U reactor system can be designed to operate at peak output throughout the ~ 20 year fuel cycle.

7. Instrumentation and Control System

Instrumentation is limited to flux monitors for controlled reactor commissioning, and temperature sensing to initiate the periodic injection of pellets of neutron absorber for reactivity management as detailed in (c).

8. Plant Layout Arrangement

The SSR-U is flexible and can be deployed in different array sizes as demanded by the application. An illustrative SSR-U plant layout arrangement would be an array of 32 reactors on the nuclear island, which will provide circa 1.2 GW(th) energy which can be converted into ~512 MW(e) 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.

Artists impression of an array of 32 modules configured in line with turbine hall, GridReserve® storage facility and sub-station

9. Testing Conducted for Design Verification and Validation

Current testing is focused on materials & corrosion, confirmation of thermo-physical properties of salts for use in modelling, and thermohydraulic tests. Moltex Energy 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 Moltex Energy with analysis of test samples including ICP-OES (for salt quality) and SEM (for materials performance).

10. Design and Licensing Status

Interaction with Nuclear Safety Authority to be launched in 2023. Site permit for FOAK to be developed, however a number of sites are under consideration and discussions are ongoing.

11. Fuel Cycle Approach

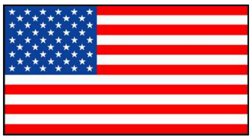
The SSR-U is fuelled for 16 full power years, with a further 2 fuel cycles possible. Interim storage of spent fuel will be in multi-purpose storage cannisters on site until site decommissioning. The spent fuel will then be conditioned for geological disposal or alternatively, recycled using another advanced nuclear technology under development by Moltex Energy, named WATSS (WAste To Stable Salt process).

12. Waste Management and Disposal Plan

Moltex are currently developing the design to be consistent with international good practice for waste management and spent fuel, aligned to 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 envisage minimal waste production during normal operation of the SSR-U. Argon is likely to be generated as a by-product of the process, however, it is unlikely to ever exceed 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 or recycled using the Waste To Stable Salt process -WATSS). Decommissioning will produce solid, liquid, and gaseous wastes, the largest being graphite. Moltex seek to recycle and re-use graphite wherever possible in new SSR-U reactors.

13. Development Milestones

Q4 - 2022	Concept Design Freeze for SSR-U
Q3 - 2023	Initiate regulatory asessment of the technology
Q4 - 2023	Preliminary Safety Report (PSR) Complete



KP-FHR (Kairos Power, United States of America)

Artist's rendering of the Hermes demonstration reactor facility to be constructed in Oak Ridge, TN

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Kairos Power, LLC, United States of America
Reactor type	Modular, pebble bed, high temperature, salt-cooled reactor
Coolant/moderator	Li_2BeF_4 (Flibe) / graphite
Thermal/electrical capacity, MW(t)/MW(e)	320 / 140
Primary circulation	Forced Circulation
NSSS Operating Pressure (primary/secondary) (MPa)	< 0.2
Core Inlet/Outlet Coolant Temperature (°C)	550 / 650
Fuel type/assembly array	TRISO particles in graphite pebble matrix / pebble bed
Power conversion process	Superheated steam cycle
Fuel enrichment (%)	19.75
Refuelling cycle (months)	Online refueling
Reactivity control mechanism	Control elements, boron
Approach to safety systems	Passive
Design life (years)	20 (vessel), 80 (plant)
RPV height/diameter (m)	7.2 / 3.9
Seismic design (SSE)	Target: contiguous USA
Fuel cycle requirements / approach	Once-through Uranium
Distinguishing features	Longer than 72-hour coping time for core cooling without AC or DC power, or operator action
Design status	Conceptual design

1. Introduction

Kairos Power is a mission-driven company singularly focused on its effort to commercialize advanced reactor technology in time to play a significant role in the fight against climate change. Kairos Power is disrupting the industry with rapid iterative development and vertical integration strategies to deliver a clean energy solution with robust safety at an affordable cost.

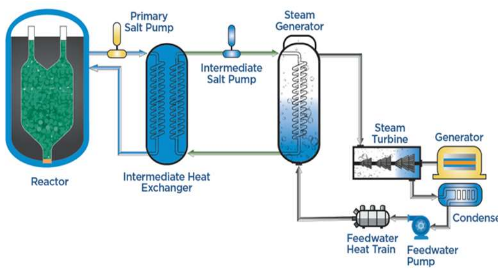
The Kairos Power fluoride salt-cooled high temperature reactor (KP-FHR) is a novel advanced reactor technology that leverages TRISO fuel in pebble form combined with a low-pressure fluoride salt coolant. The technology uses an efficient and flexible steam cycle to convert heat from fission into electricity.

2. Target Application

This advanced reactor technology is designed for high availability and performance with low maintenance and lifecycle costs, providing dispatchable power that improves grid resiliency and security. In combination with variable renewables, this technology can create a path to a truly clean energy system. The KP-FHR aims to be cost competitive with natural gas.

3. Design Philosophy

The fundamental design concept is the combination of Tri-structural Isotropic (TRISO) particle fuel coupled with molten fluoride salt coolant ($2LiF:BeF_2$, Flibe). This combination results in a high temperature, low-pressure reactor with robust, passive safety systems. In addition to robust, inherent safety, the design also reduces reliance on high-cost, nuclear grade components and structures and leverages conventional technologies to lower capital costs.



KP-FHR Power conversion system diagram

4. Main Design Features

(a) Power Conversion

The power conversion system will leverage prior technology demonstration of solar nitrate salt steam generators and conventional power conversion technologies for the balance of plant.

(b) Reactor Core

4.0-cm diameter spherical fuel and moderator pebbles forming the active region of the core. The core design utilizes a cylindrical geometry with a graphite side-reflector and bottom and top graphite structures. The core internal structures enable reactivity control and shutdown elements.

(c) Reactivity Control

Reactivity control for reactor maneuvering and non-accident events is provided by control elements that insert into the graphite reflector surrounding the pebble bed core. Reactivity control during accident events is provided by shutdown elements that insert directly into the pebble bed. The shutdown elements are gravity driven and are released by the reactor protection system. Both the shutdown and control elements consist of a composite structure of neutron absorber material made of natural B_4C in an inert gas with SS316H cladding. The shutdown and control elements fail safe (insert) on a loss of power. The number and the location of the shutdown elements are selected to provide sufficient shutdown margin at zero-power conditions. The shutdown system is designed with sufficient reactivity worth to shut down the core from hot full power assuming the failure of the highest worth control element. The reactivity control elements can be inserted for shutdown at a rate sufficient to assure that the design limits for the fission product barriers are not exceeded.

(d) Reactor Vessel and Internals

The KP-FHR reactor vessel operates at low pressure, so has thin-walled construction using 316H stainless steel that complies with ASME Section III, Division 5 requirements. The reactor internals include a core barrel and reflector support structure that position and retain the graphite reflector structure which maintains alignment of the structure to form the major flow paths and reactor core volume. As the pebbles are positively buoyant in the coolant, defueling occurs from the top of the core and online fuel addition from the bottom. The reactivity control and other components interface through the top head.

(e) Fuel Characteristics

Kairos Power's reactor uses fully ceramic fuel, which maintains structural integrity even at extremely high temperatures and will be undamaged to well above melting temperatures of conventional metallic reactor fuels.

(f) Fission Product Management

The fundamental safety strategy for the KP-FHR is rooted in the retention of fission products within the TRISO layers of the fuel particle design with additional retention in the flibe salt coolant. The TRISO layers are credited for providing a 'functional containment' for meeting design basis accident dose limits. The silicon carbide (SiC) coating on the TRISO particles is the primary fission product barrier, while the pyrolytic carbon layers and matrix act as secondary barriers for trapping or impeding the transport of fission products and protecting the integrity of the SiC layer.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The reactor system includes a pebble bed core, surrounded by a graphite reflector, contained within a cylindrical 316H stainless steel reactor vessel.

(b) Decay Heat Removal System

Decay heat removal during normal operations and non-accident events is provided by a normal shutdown cooling system that connects directly to the primary heat transport system. Decay heat removal during accident events is provided by a passive reactor vessel auxiliary cooling system (RVACS) located external to the reactor vessel. No coolant injection is required for inventory makeup nor decay heat removal functions under accident conditions. The RVACS relies on decay heat through thermal radiation and natural convective heat transfer utilizing a thermosyphon concept. The system evaporates water to provide heat removal and does not require inventory makeup for a minimum of 72 hours. System valves are fail-safe and do not rely on safety related electrical power or operator action.

(c) Emergency Core Cooling System

Kairos Power reactors have uniquely large safety margins based on the selected combination of fuel and coolant, which allows emergency cooling to be driven by fundamental physics rather than engineered systems.

(d) Containment System

Functional containment in the KP-FHR is provided by the robust intrinsic safety characteristics of the TRISO fuel and Flibe coolant to ensure that the health and safety of the public and workers are protected. Multiple additional barriers in the KP-FHR provide defense-in-depth.

(e) Chemical Control System

A Chemistry Control System (CCS) monitors the salt coolant to detect impurities, ensuring it stays within an established operating window.

(f) Spent Fuel Cooling Safety Approach / System

The zirconium cladding used in light-water reactor fuel contains substantial stored energy, which affects the safety significance of spent fuel cooling systems. KP-FHR fuel is fully ceramic, which simplifies cooling requirements for fuel in the pebble handling system and also in canister storage.

6. Plant Safety and Operational Performances

The KP-FHR leverages intrinsic safety characteristics of the fuel and coolant to achieve uniquely large safety margins while lowering capital costs and improving operating economics. The fuel in the KP-FHR is the TRISO particle fuel, which can withstand fuel particle temperatures up to 1600°C. The Flibe is chemically stable and is at low-pressure, with a boiling point of 1430°C, notably lower than 1600°C and yet functionally very high. The combination of extremely high-temperature-tolerant fuel and low-pressure, single-phase, chemically stable reactor coolant removes entire classes of potential fuel-damage scenarios, greatly simplifying the design and reducing the number of safety systems. The intrinsic low pressure of the reactor and associated piping, along with the functional containment provided by the TRISO fuel, enhances safety and eliminates the need for high-pressure containment structures.

7. Instrumentation and Control System

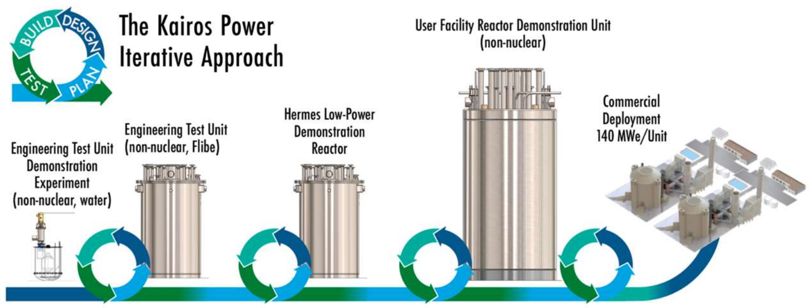
The Kairos Power Instrumentation and Control Systems takes advantage of the inherent safety features of the KP-FHR technology to simplify the reactor protection system configuration and automated actions. The system is sufficiently simplified to allow for deterministic analysis of all design basis events. The plant control system operates within the protection system established operating envelope as an industrial control system, maximizing automation and continuous health monitoring of the plant. The I&C Reactor Protection System (RPS) provides protection during steady state and transient power operation and includes the capability to manually or automatically trip the reactor and activate RVACS. The RPS is fully independent of the Plant Control System, which provides overall control during normal operation, startup or shutdown.

8. Testing Conducted for Design Verification and Validation

Kairos Power is committed to a culture that embraces iterative development and facilitates an integrated design philosophy, testing program, and licensing approach to mitigate technical, licensing, manufacturing, and construction risk while establishing cost certainty through iterative hardware demonstrations.
By mirroring SpaceX and its successful rapid spiral development, Kairos Power accelerates test cycles for innovation and optimization. Our rapid iterative approach leverages multiple design-build-test cycles with both nuclear and non-nuclear systems prior to the first commercial reactor. Major hardware iterations include:

Engineering Test Unit (ETU) – a non-nuclear system that will 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 is set to be commissioned in Summer 2022.

Hermes – a 35 MWth non-power reactor that will demonstrate Kairos Power's ability to produce affordable nuclear heat. Currently in the design phase, Hermes is scheduled to be operational in 2026.



Kairos Power's Rapid iterative approach to hardware development

U-Facility – a non-nuclear demonstration intended to test manufacturing and construction of primary reactor systems, serve as a training center and reduce O&M cost uncertainty

KP-X – a first-of-a-kind 140 MWe commercial KP-FHR operating at grid-scale

9. Design and Licensing Status

Kairos Power submitted the construction permit application (CPA) for the Hermes low-power demonstration reactor to the U.S. Nuclear Regulatory Commission (NRC) in October 2021 and it was accepted for review the following month. Prior to that, we conducted extensive pre-application engagement with the NRC. To date, Kairos Power has submitted 11 topical reports and several technical reports, many of which have been approved by NRC staff with the remainder expected soon. The Construction Permit Application for the Hermes low-power demonstration reactor is under review by the U.S. Nuclear Regulatory Commission. Scheduled to achieve criticality in 2026, Hermes is a reduced scale (35MWth) demonstration reactor that will prove Kairos Power's capability to deliver low-cost nuclear heat.

10. Fuel Cycle Approach

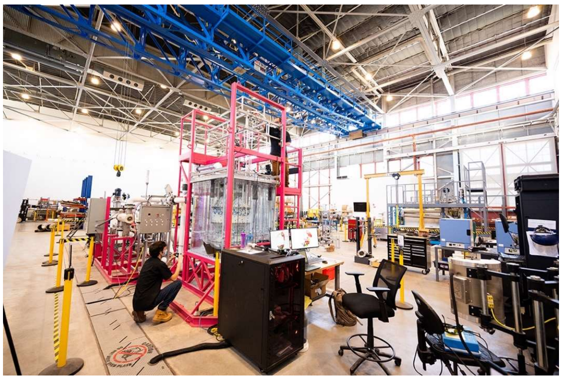
The KP-FHR operates on a continuous refueling cycle. Fuel pebbles are extracted from the core in the reactor vessel and inspected for burnup and integrity in a pebble handling system. Pebbles are either inserted back into the active core or directed to spent fuel storage. Fuel pebbles pass through the core six times on average before reaching full burnup.

11. Waste Management and Disposal Plan

The used TRISO pebble fuel and a small inventory of ‘greater than Class C’ waste is packaged in multi-purpose canisters for dry interim storage and subsequent off-site transportation for direct geologic disposal or recycling depending upon national policies. All remaining waste streams from operation and decommissioning qualify for low-level waste disposal.

Engineering Test Unit under construction

Salt Lab located in Alameda, CA

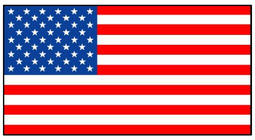


RAPID Lab located in Alameda, CA

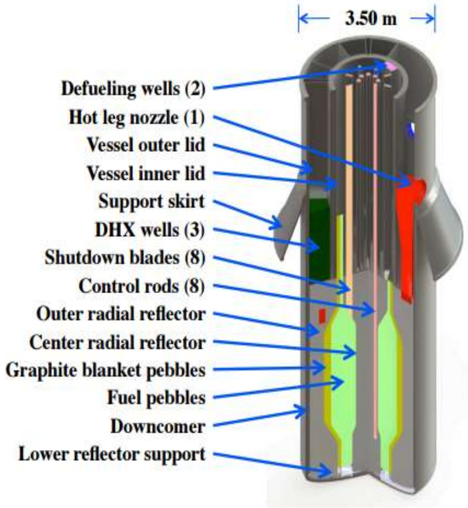
In-house manufacturing shop located in KP-Southwest

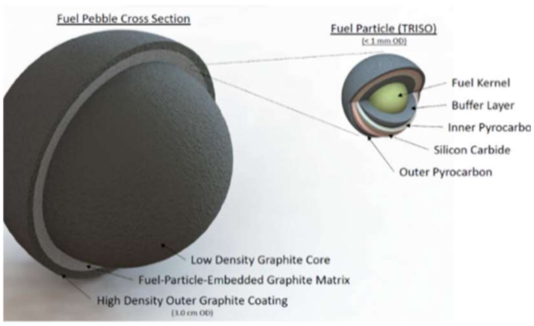
12. Development Milestones

2018	Pre-Conceptual Design Completed
2018	Commissioned R-Lab
2018	Initiation of Pre-Application Review with NRC
2020	Acquisition of KP-Southwest Facility in Albuquerque, NM
2020	Commissioned S-Lab
2020	Formed strategic partnership with Materion Corporation to produce Flibe
2020	Awarded $629M cost-shared Advanced Reactor Demonstration Program Risk Reduction Award from the U.S. DOE
2021	Cooperative Development Agreement with Tennessee Valley Authority to deploy Hermes low-power demonstration reactor
2021	Testing Facility construction completed in Albuquerque, NM
2021	Hermes Construction Permit Application accepted for review by U.S. Nuclear Regulatory Commission
2021	Construction begins on the Engineering Test Unit (ETU)
2022	Kairos Power establishes the KP-OMADA Advanced Nuclear Alliance with leading North American utilities and generation companies
2022	30,000th pebble manufactured for ETU
2022	Manufacturing infrastructure expanded at KP-Southwest
2022	[Planned Milestone] Molten Salt Purification Plant (MSPP)
2022	[Planned Milestone] ETU commissioning



Mk1 PB-FHR (UC Berkeley, United States of America)





MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	University of California, Berkeley, United States of America
Reactor type	Fluoride-salt-cooled high temperature reactor (FHR)
Coolant/moderator	Li_2BeF_4 / graphite
Thermal/electrical capacity, MW(t)/MW(e)	236 / 100
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.3 / 1.85 (compressor outlet)
Core Inlet/Outlet Coolant Temperature (ºC)	600 / 700
Fuel type/ assembly array	TRISO particles in graphite pebble matrix / pebble bed / 470 000
Fuel enrichment (%)	19.9
Fuel burnup (GWd/ton)	180
Fuel cycle (months)	On-line refuelling
Fuel core residence time (months)	2.1, average of 8 passes to achieve full burn up
Main reactivity control mechanism	Negative temperature coefficient; control rod insertion
Approach to safety systems	Passive
Design life (years)	60
Plant footprint (m^2)	45 000
RPV height/diameter (m)	12 / 3.5
Seismic design (SSE)	~ 0.3g
Distinguishing features	Large fuel and coolant thermal margin, high temperature operation
Design status	Pre-conceptual design

1. Introduction

The Mark 1 Pebble-Bed Fluoride-Salt-Cooled High-Temperature-Reactor (Mk1 PB-FHR) is a small, modular graphite-moderated reactor. FHRs are differentiated from other reactor technologies because they use high temperature, coated particle fuels, and are cooled by the fluoride salt flibe ($^7Li_2BeF_4$). The Mk1 PB-FHR design described here is the first FHR design to propose driving a nuclear air-Brayton combined cycle (NACC) for base-load electricity generation.

2. Target Application

The Mk1 PB-FHR is designed to produce 100 MW(e) of base-load electricity when operated with only nuclear heat, and to increase this power output to 242 MW(e) using gas co-firing for peak electricity generation. This provides a new value proposition for nuclear power to earn additional revenues by providing flexible grid support services to handle the ever-increasing demand for dispatchable peak power. This is in addition to traditional base-load electrical power generation.

3. Design Philosophy

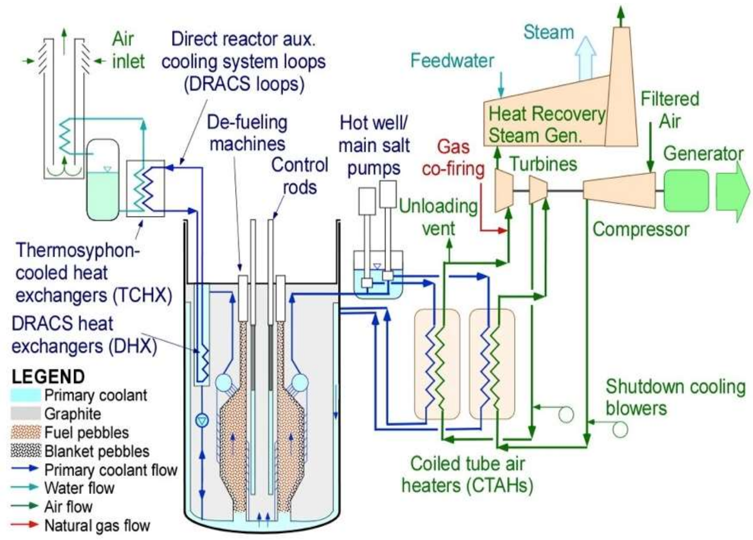
The Mk1 PB-FHR is designed with advanced passive safety features and intrinsic fuel and coolant properties which make the consequences of severe accidents studied for light water reactors much easier to manage. Passive safety mechanisms include natural circulation decay heat removal activated by a passive check valve in accident conditions and buoyant control rods for emergency shutdown without operator intervention.

Fluoride salt coolants have uniquely high volumetric heat capacity, low chemical reactivity with air and water, very low volatility at high temperature, effective natural circulation heat transfer, and high retention of most fission products. These characteristics are in addition to reasonably low neutron capture probability (when using enriched ^{7}Li), and good neutron moderation capability.

4. Main Design Features

(a) Power Conversion System

The 236-MW(t) Mk1 PB-FHR uses the NACC power conversion system. It uses a General Electric (GE) 7FB gas turbine (GT), modified to introduce external heating and one stage of reheat, in a combined-cycle configuration to produce 100 MW(e) under base-load operation, and with natural-gas co-firing to rapidly boost the net power output to 242 MW(e) to provide peaking power. The Power Conversion Unit consists of the reactor core, two coiled tube air heaters (CTAHs) to transfer heat from the main salt to pressurized air, a heat recovery steam generator system (HRSG), steam condenser, and the GT. During normal operation, the primary coolant relies on forced circulation



(b) Reactor and Core-unit

The core design incorporates an annular pebble-bed core geometry composed of a homogeneous mix of fuel pebbles adjacent to the center graphite reflector, with a layer of inert graphite reflector pebbles on the outside that reduces the fast-neutron fluence to the outer fixed radial graphite reflector sufficiently for it to last the life of the plant.

The center reflector provides 8 channels for insertion of buoyant control rods, and it also provides flow channels for radial injection of coolant into the pebble core, to provide a combined radial and axial flow distribution that increases the effectiveness of heat transfer from the fuel and results in lower average fuel temperature. The center graphite reflector internals need to be replaced periodically due to radiation damage.

(c) Reactivity Control

For reactivity control, the Mk1 is designed to have negative fuel, moderator, and coolant temperature reactivity feedbacks. The design uses a buoyant control rod system for normal reactivity control, and the system also provides a passive shutdown capability because the buoyant rods will insert if the reactor coolant temperature in the control-rod channel exceeds 615°C, the buoyant stability limit. The design also uses shutdown blades that can insert directly into the pebble bed for reserve shut down. In the event that electrical power is interrupted to the drive mechanisms for the motors of the control rod and shutdown blade cable drums, they will insert and shut the reactor down.

(d) Reactor Vessel and Internals

To enable near-term deployment, the Mk1 design uses a core barrel and other core internal structures fabricated from the same metallic material as the reactor vessel and main salt piping. The outer radial reflector blocks are aligned and held against the metallic core barrel using a system of axial alignment ribs and radial retaining rings quite similar to designs originally developed for the Molten Salt Breeder Reactor (MSBR) project. The use of metallic core internal structures, rather than advanced ceramic composites, simplifies fabrication and licensing for the Mk1 design.

(e) Fission Product Management

The coated uranium particles are packed in an annular fuel zone around a low-density graphite core. One Mk1 pebble contains 1.5 g of uranium enriched in ^{235}U to 19.9% and encapsulated inside 4730 coated particles. The very low circulating power for the coolant in salt-cooled reactors, compared to helium-cooled reactors, makes it practical to use smaller pebbles. This small-pebble design doubles the pebble surface area per unit volume and halves the thermal diffusion length, enabling a substantial increase in power density while maintaining relatively low peak fuel particle temperature. The coated particles have the main fission retention function, but the molten salt coolant, primary circuit and building also serve as barriers to release.

(f) Heat Recovery Steam Generator System

The Heat Recovery Steam Generator System (HRSG) and steam condenser need to be sized for full power operation at co-firing conditions. The large HRSG inlet temperature variation between baseload and co-fired operation modes introduces certain caveats to the steam cycle design. With the expected frequent power ramping of the GT and dissimilar ramping rates compared to the steam turbines/HRSG, special design considerations are needed such as opening steam turbine inlet valves and or allowing some bypass flows.

5. Safety Features

The safety objective with the IMSR® design is to achieve high inherent safety, and a walk-away safe nuclear power plant. No operator action, electricity, or externally-powered mechanical components are needed to assure the primary safety functions of controlling, cooling, and containing.

(a) Engineered Safety System Approach and Configuration

For reactivity control, the Mk1 has a combination of intrinsic features and passive systems. It has negative fuel, moderator, and coolant temperature reactivity feedbacks. The reduced fuel temperature in the PB-FHR provides improved response to hypothetical ATWS accidents. The negative fuel temperature reactivity feedback in FHRs is significantly larger than the coolant temperature reactivity feedback, because the coolant does not boil—the boiling temperature of flibe is 1430°C—as in light water reactors (LWRs), and larger than the graphite moderator temperature reactivity feedback. Under the beyond design basis ATWS accident where reactor scram does not occur upon loss of flow or loss of heat sink, the FHR coolant equilibrates to a temperature close to the original fuel temperature. Simplified analysis for the Mk1 design indicates that this equilibrium ATWS temperature will be below 800°C.

(b) Decay Heat Removal System

In the PB-FHR core, the emergency heat removal safety function is also controlled by passive mechanisms. The PB-FHR design concept employs a passive check valve to activate natural-circulation-driven heat transport to a set of three Direct Reactor Auxiliary Cooling System (DRACS) loops and ultimately to Thermosyphon-cooled Heat Exchangers (TCHXs) upon Loss of Flow Condition (LOFC). Heat removal from the TCHXs is regulated by fail-open valves that supply water to the thermosyphons integrated into these heat exchangers. The valves are held closed during normal operation, and can also be closed to control over-cooling during prolonged reactor shutdown. In addition to the passive emergency decay heat removal provided by the DRACS, the PB-FHR power conversion system and the normal shutdown cooling system provide heat removal capability and defence in depth in assuring adequate core heat removal.

(c) Emergency Core Cooling System

Coolant inventory control is provided by fully passive mechanisms that require no RPS or manual operator actions. The primary salt fulfils dual roles during design basis events, by providing natural-circulation heat removal and preventing chemical attack to fuel pebbles from exposure to air. The PB-FHR utilizes a pool-type reactor configuration, similar to the design adapted for many sodium fast reactors. For BDBEs where the vessel leaks or ruptures, the Mk1 refractory cavity liner insulation system controls the level change in the vessel and prevents uncovering of fuel.

(d) Containment System

The Mk1 design introduces another novel feature, a “gas gap” system, to make it physically impossible to transmit excessive pressures to the reactor vessel and reactor cavity/containment from potential tube or manifold pipe ruptures in a CTAH. The gas gap is created adjacent to the containment penetrations for the hot and cold legs. For the Mk1 PB-FHR, water pools are used inside the shield building to provide water to thermosyphon-cooled heat exchangers (TCHXs) in the DRACS modules, as well as to the reactor cavity liner cooling system. Because these water pools also provide a source of water for evaporative cooling under beyond-design-basis event (BDBE) conditions, they are provided with a secondary confinement following the “tank-within-tank” design principle.

6. Plant Safety and Operational Performances

Due to the high thermal efficiency of the NACC system, the steam-bottoming condenser requires only 40% of the cooling water supply that is required for a conventional LWR, for each MWh of base-load generation. As with conventional natural-gas combined cycle (NGCC) plants, this makes the efficiency penalty of using dry cooling with air-cooled condensers much smaller, enabling economic operation in regions where water is scarce. The advantage of the NACC system arises from additional revenues earned by providing flexible grid support services because under base-load operation NACC power conversion has lower fuel costs than NGCC, and under peaking operation has higher efficiency in converting natural gas to electricity than NGCC, NACC plants will always dispatch before conventional NGCC plants.

7. Instrumentation and Control Systems

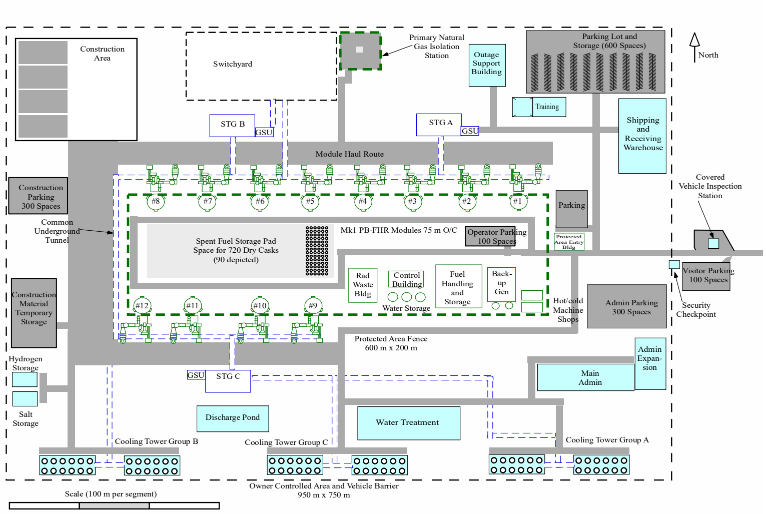
The digital control system is designed so that neither its actions nor its failure to act would have any deleterious impact on the ability of the PB-FHR to respond safely to design basis events. The quality requirements for the control system then arise from the economic incentives to maximize system performance and to preserve the invested capital, thus high-quality commercial-grade equipment is anticipated to be used.

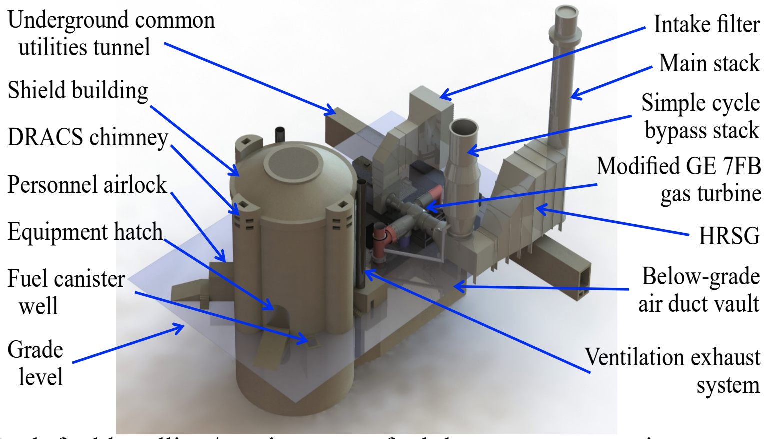
Except during startup and low-power conditions, the PB-FHR operates with constant core inlet and outlet temperatures. Load-following capability is made possible by air bypass flow to respond to rapid load-change transients and turbine inlet temperature control (by bypassing air around the CTAHs) for slower transients.

Pump speed control is then used to control the core temperature difference, and control rod position is used to control the average temperature. The control system adjusts the pebble loading and unloading schedule to maintain sufficient excess reactivity to accommodate a xenon transient equivalent to a rapid power reduction from 100% to 40%.

8. Plant Layout Arrangement

The figure on the right presents a notional 180-acre site arrangement for a 12-unit Mk1 PB-FHR power plant capable of producing 1200 MW(e) base load and 2900 MW(e) peak power output. Due to the much smaller cooling requirements, they do not need to be sited near bodies of water. Population centers tend to be located near bodies of water which means that FHRs can be sited in areas where fewer people want to live. So, rather than attempt to minimize the site footprint, the more important goal is likely to facilitate construction of modules adjacent to operating modules, and to optimize the degree to which some services are shared.





(a) Reactor Building

The Mk1 reactor building and NACC system arrangements supports a multi-module plant configuration by allowing multiple units to be lined up in a row with a clear boundary between the reactor and its vital areas, versus the balance of plant (BOP). The GT and associated equipment are configured to minimize the air pressure loss and circulating power in the air ducting while maintaining a clear boundary between the reactors and the BOP. This configuration makes it easier to co-locate combined nuclear services on one side of a multi-module plant (training, fresh fuel handling/receipt, spent fuel dry storage, security, access control, multi-module control room, hot-rad/Be shops, etc.), and have BOP combined services on the other side (off-site transmission, process steam loads and/or steam bottoming turbines, cooling towers, etc.).

The Mk1 reactor building is partially embedded below grade, with the reactor deck located slightly above grade, shortening the air duct lengths and the depth of the air-duct vault. The baseline Mk1 reactor building design uses a cylindrical shield building fabricated from steel-plate/concrete composite (SC) modules, quite similar to the Westinghouse AP1000 shield building. The overall height and diameter of the Mk1 shield building are 47.5 m and 24.5 m, respectively, compared to 83 m and 42 m for the 1150-MW(e) AP1000, so the Mk1 shield building volume is 2.2 times greater than the AP1000, per MW(e) baseload.

9. Design and Licensing Status

Further work is needed in the definition and design of: plant staff capabilities and size, instrumentation requirements, systems and equipment for operations and maintenance, future plant reliability and availability, and licensing strategies for licensing commercial prototypes in the U.S. as well as internationally.

10. Fuel Cycle Approach

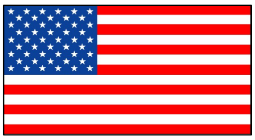
The initial design is based on once through LEU cycle but other thermal spectrum based fuel cycles (U-Th; Pu-Th, Pu burner) as illustrated by HTR coated particle fuel, are in principle possible.

11. Waste Management and Disposal Plan

The fuel pebbles are based on HTR coated particle fuel with its excellent radioactivity containment characteristics. Upon removal from the primary coolant, the spent fuel will require cleaning to remove residual salt and cooling to maintain acceptable fuel temperatures in the gas environment.

12. Development Milestones

2018	Pebble bed FHR technology, with significant similarities to the Mk1 PB-FHR, is being developed by Kairos Power



Molten Chloride Salt Fast Reactor MCSFR (Elysium Industries, USA)

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Elysium Reactor 500 MWth

Elysium Reactor 3000 MWth

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Elysium Industries, United States of America
Reactor type	MSR – Fast Chloride
Coolant/moderator	NaCl-XCl_v-YCl_z-$UCl_{3/4}$-$PuCl_3$-$FPCl_y$ fuel salt / None
Thermal/electrical capacity, MW(t)/MW(e)	(125 / 50), (500 / 200), (1 000 / 400) , (3 000 / 1 200)
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	0.1 + pump head+hydrostatic / slightly higher
Core Inlet/Outlet Coolant Temperature (°C)	650 / 750 (Goal: 950)
Fuel type/assembly array	Molten Chloride Salt
Number of fuel assemblies in core	n/a, Full liquid fuel core
Fuel enrichment (%)	10% Pu fissile/(Pu+U total) or ~15% enriched HALEU
Core Discharge Burnup (GWd/tonne)	300-500 (40-60 years), fertile added, fuel never damaged
Purification cycle (months)	480-720
Main reactivity control mechanism	Fuel expansion in/out of core; Fertile fuel addition; Passive fuel draining
Approach to safety systems	Passive to Air Cooling
Design life (years)	Unlimited core, 15-40 for components, 100+ for plant
Plant footprint (m^2)	1/3 size of LWR
RV height/diameter (m)	9/4
Seismic design (SSE)	Tension skirt Lateral snubbers
Fuel cycle requirements / Approach	U/Pu Closed Fuel Cycle SNF/DU/NU (1t/GWe-yr)
Distinguishing features	Fast spectrum, no in-core structure, 60 years fuel life
Design status	Conceptual design

1. Introduction

The MCSFR is a modular configuration and construction reactor. The MCSFR enables the closing of the fuel cycle, while providing reliable, passively safe, proliferation-resistant, and environmentally-friendly energy (heat/electricity) generation. The fuel is part of the liquid heat transport eutectic fluid with heat directly deposited in the liquid fuel/coolant.

2. Target Application

The MCSFR is designed for mass production for domestic use and export to address global markets for cost-competitive, low-emission electricity, and high temperature process heat (e.g.: H_2, syn-fuel, syn-fertilizer, desalination, district heating, cement, steel, etc). The MCSFR uses Spent Nuclear Fuel (SNF), plutonium (Pu), or depleted uranium (DU) ‘waste’ with fuel production denaturing and complete consumption (e.g.: US, Canada, UK, Japan, South Korea, etc.), as fuel sources for internal or export applications and developing countries with fuel take-back. Utilities can ‘start small’, then add heat exchangers (Hx’s) to increase capacity without purchasing and licensing a new reactor.

3. Design Philosophy

Elysium's team designed the MCSFR using attributes from:

1) water reactors, common, low cost coolant (table salt), fuels (nuclear waste), and qualified materials,

2) liquid metal reactors, the low pressure and materials without corrosion concerns,

3) gas reactors, the flow pattern for low containment temperatures, and very high peak temperatures,

4) heat-pipe reactors, passive temperature dependent on/off heat-pipe decay heat removal.

Elysium philosophy includes mitigating public concerns with passive safety, high thermal and fuel efficiency, SNF/Pu waste denaturing and consumption, dramatically increased fuel supply, elimination of coolant reactions with water/air/structure, significantly reduced construction, fuel, and operational costs, in addition reduced proliferation concerns.

4. Main Design Features

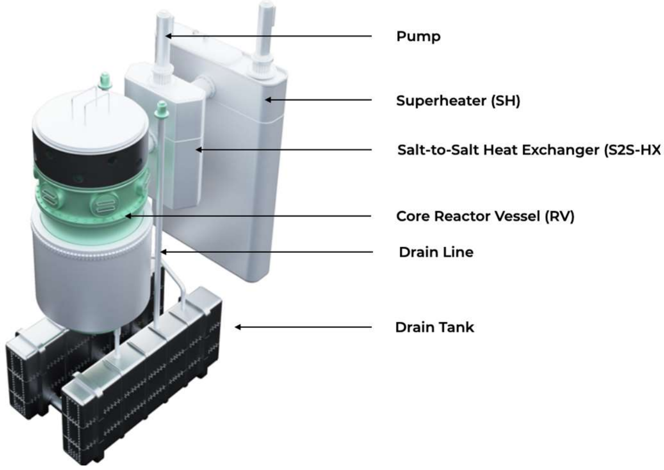
(a) Power Conversion System and Cogeneration

The intermediate clean salt heats power conversion unit (PCU) saturated steam to SH steam in the SH, with 35% of SH steam to the steam Rankine turbine-generator (40-50% efficiency) and 65% of SH steam to the Loeffler boiler.

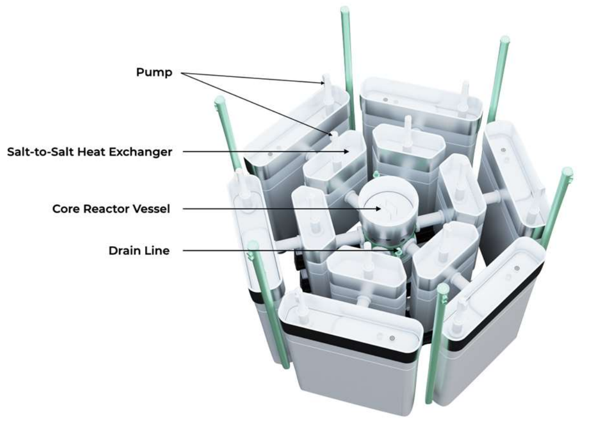
The use of a salt to SH, i.e. a gas Hx allows use of other process heat gasses, especially at 950ºC, on a per Hx basis to allow flexibility of products. With the high outlet temperatures, process heat applications include H2, synthetic fuel and fertilizer, oil/gas recovery and refining, industrial process heat, and cement manufacturing. Other applications include district heating/cooling, thermal storage in chloride salt, and desalination. Use of salt to SH Hx's, dramatically reduces water and potential for transient pressures in containment.

(b) Reactor and Core-unit

The core size is minimized to barely maintain criticality with no in-core structures and is near spherical. The RV and all fuel and intermediate component shells are cooled by cold coolant inside, so includes pipe-in-pipe nozzles. The core is pure salt, except for a downcomer shroud near the edge of the core, and the lower RV is the core edge with an ex-RV radial reflector and above core reflector/shield.



500 MWth Elysium Reactor Unit



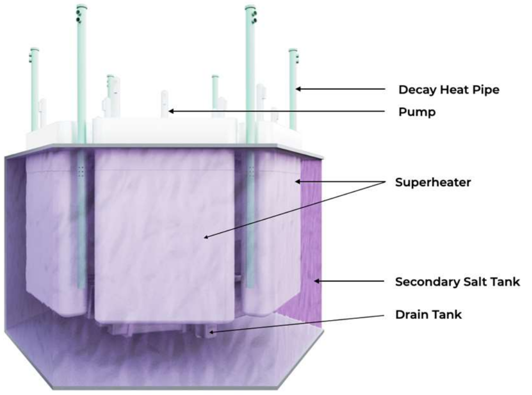
3000 MWth Elysium Reactor Unit – Axial View

(c) Reactivity Control

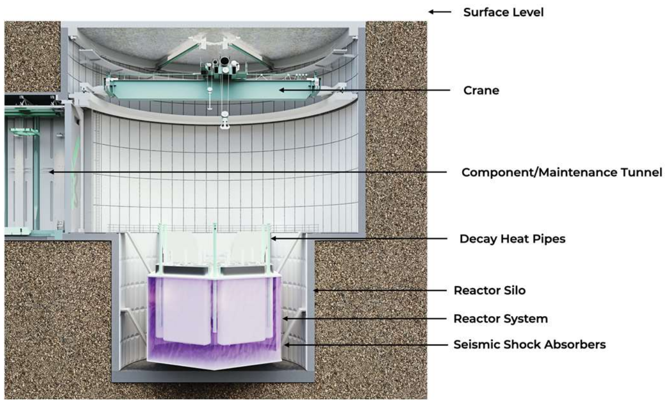
Reactivity control is via the negative temperature and void coefficients. As the fuel-salt temperature increases the fuel-salt expands and fissile/fuel salt is squeezed out of the core, reducing power and vice versa. The long-term reactivity adjustments are made by on-line fertile fuel additions. The reactor can also be shut down by tripping pumps to allow draining to criticality safe/passively cooled drain/expansion tanks.

(d) Reactor Vessel (RV)

The RV is low pressure, thin-walled stainless steel, with up to 6 nozzles, cooled by cold fuel salt everywhere inside, and submerged in a cooling/shield tank of clean salt for corrosion prevention/cooling outside. The RV near the core is the core outer diameter and is replaceable separately from the upper RV nozzle region. A reflector, if needed, is outside the RV.



3000 MWth Elysium Reactor Unit – Front View



View of the MCSFR Underground Reactor Building

(e) Fuel Characteristics

The fuel-salt is NaCl-XCl_v-YCl_z-$UCl_{3/4}$-$PuCl_3$-$FPCl_y$, allowing it to contain ~30% total actinide Chlorides with a 10-20% fissile fraction and 99.9% actinide consumption. Fissile options include: 1) Preferred-Reactor Grade or Weapons Grade Plutonium (RGPu and WGPu) due to revenue and is denatured with SNF, if needed, 2) high assay low enriched uranium (HALEU), 3) high enriched uranium (HEU) denatured at fuel production site. Fertile options include: a) Prefered-LWR/CANDU SNF, b) Depleted Uranium (DU), c) Natural Uranium (NU), d) residual U from other mining, coal ash, or seawater extraction, e) thorium (Th) combined with >88% U_{238} for denaturing. Pu fissile is added for startup only. HALEU requires continued feed-in declining in enrichment over 5-10 years. Fissile is iso-bred plus enough to counteract fission product poison buildup to prevent needing purification for many decades.

(f) Nuclear Heat (Steam) Supply System (NHSS)

The MCSFR has a fuel/coolant loop, clean intermediate salt loop same as fuel salt, without the fuel/fission products (FPs), etc., the steam super-heater (SH) loop all inside containment, with a Loeffler steam boiler, outside of containment. The reactor has 1 to 6 Hx/pump loops for each fuel and clean salt loop. Each of the loops can be used for 6 different heat applications/customers. Loops can operate at variable power up to maximum with fuel cost not a concern at part-load.

(g) Primary Loop

The primary loop is chloride fuel salt. The arrangement is modular, like an HTGR to allow for low temperature containment, yet high temperature for efficiency and process heat applications. Six pipe-in-a-pipes connect in a modular arrangement to up to 6 fuel salt to clean intermediate salt Hx's, with top mounted pumps after the Hx to prevent motor immersion and heating. Radial dimension is ≤ 4m for road shipping.

(h) Intermediate Loop

The intermediate loop is the same salt as the fuel salt without the actinides or FPs, and is at a higher pressure than the fuel salt to ensure any leakage is not of fuel salt outward, inward for dilution/shutdown of the core. The intermediate loop transports heat to the modularly mounted SH Hx, with the intermediate salt pump on top of one end of the SH. Radial dimension is ≤ 4m for road shipping.

(i) Fuel Cycle and Length Approach

On-line fertile fuelling is used. Iso-breeding ratio is ~1.014/yr to offset fission product buildup with a doubling time of 50-60 years with a fuel life of 40-60 years when it is sent to a central facility for partial purification and recycling of all actinides, carrier salt components, including all chlorine. The fuel consumption rate is ~1 ton U / GWe-yr. Fuel is added every ~8 hrs-7 days depending on the power level by dropping a fertile pebble into a perforated basket in the drain tank flow path.

(j) Cooling System

Four different cooling loops are included: the fuel salt, intermediate salt, the power/process heat loop, and the heat sink loop. With the high temperature and very low fuel cost, dry cooling is an economic alternative.

(k) Proliferation Considerations

The MCSFR never contains weapons grade materials. Fuel is not removed for 40-60 years. WGPu or HEU is denatured at the single fuel production facility. Purification at 60 years never separates Pu from U, Cs and Sr. This low separation of FPs reduces recycling cost. Only relatively short-lived, ~100 years to below U background, FPs are removed as waste.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

Passive features include the large mass and heat capacity of the fuel, intermediate, and tank salt to slow heat up. Pumps constantly fill the primary system from the bottom and can be tripped passively by high or low salt temperatures, or by the reactor operator. When pumps stop, fuel salt immediately starts draining from the RV and Hx's to a passively shut down high neutron and heat leakage drain tank with external tank salt cooling. The RV and Hx's are also tank salt cooled.

(b) Decay Heat Removal System

Passive on/off heat pipe decay heat HX's remove decay heat to air from the tank salt heat mass, when temperatures exceed the vaporization point, while also shutting down preventing freezing of the fuel salt in the drain tanks when decay heat decreases to allow faster plant recovery.

(c) Containment Function

The MCSFR has 3 intrinsic and 3 physical containment barriers to leakage: chemical binding, inward leakage, salt freezing, fuel salt RV and HX shells, shield/cooling tank shell, underground low pressure containment cylinder. Containment is below grade with the top being an airplane shield.

6. Plant Safety and Operational Performances

The MCSFR is designed for passive, fast load/source following. With a closed fuel cycle, no solid fuel manufacturing costs, and extremely low fertile fuel costs or revenue from SNF, fuel cost is very low. Turbine bypass is economically viable for fast load following. Consuming more fuel via bypass operation, i.e., more SNF consumption, is a revenue offset.

7. Instrumentation and Control Systems

The MCSFR operates on load following the by changing the temperature difference. Temperatures decrease as fertile is consumed and FPs build up, which is a signal to add more fertile to increase fissile breeding. Temperature, pressure, level, and flow rate are the primary control systems, with significant peak, or minimum temperature changes passively initiating draining, and average temperature controlling fuel expansion in/out of the core. Elemental, isotopic, and molecular composition are monitored.

8. Plant Layout Arrangement

The plant is a modular configuration, like an HTGR. The RV is in the center, surrounded by short pipe mounted, i.e., modular, salt to salt, and salt to process heat HX's radially around the core in a large salt tank. The reactor is sized to just achieve criticality. Low initial power can use one HX pair, and allow uprating by adding Hx pairs as needed. The reactor is underground for aircraft and security cost reduction.

9. Design and Licensing Status

The conceptual design of pilot fuel production and reactor are in progress.

10. Fuel Cycle Approach

High fissile startup fuel is made in a single fuel production facility and shipped to the new plant. The MCSFR is a closed fuel cycle requiring only 1 tonne SNF, DU, NU fuel/GW(e)/yr for each plant. Feed-in fuel is added daily/weekly, depending on power/burnup rate, for 40-60 years. Every 40-60 years the fuel is removed/sent to one or two recycling facilities. Short-lived FPs are removed. All actinides and long-lived Cs and Sr are left together, split in two, and sent out to plants as new/replacement fuel. Chlorine is recycled. Breeding ratio (BR) is ~1.014/yr, with a ~50 yr doubling time. Elimination of solid fuel manufacture dramatically reduces fuel and recycling cost, consumes Pu and SNF waste.

11. Waste Management and Disposal Plan

The waste forms removed are fission gasses (which are allowed to decay), noble metal solid precipitates, 100-year FPs and carrier salt anions. These short-lived FPs can be mined for valuable radio- and stable-isotopes, then are allowed to cool as high temperature salts in surface stored cans cooled by air for ~100 years until stable.

12. Development Milestones

2018-2021	Pre-Conceptual design is near completion. Small scale tests of key concepts are underway with large scale testing estimated to start in 2020, followed by an Integrated Systems Test
2020-2027	Design and building a pilot fuel facility and 10-30 MW(t) pilot plant
2027-2030	Licensing/building Commercial unit

Lithium Fluoride Thorium Reactor (Flibe Energy, United States of America)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Flibe Energy, Inc., United States of America
Reactor type	Molten salt reactor
Coolant/moderator	$LiF\text{-}BeF_2\text{-}UF_4$ fuel salt / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	600 / 250
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Ambient
Core Inlet/Outlet Coolant Temperature (ºC)	500 / 650
Fuel type/assembly array	$LiF\text{-}BeF_2\text{-}UF_4$
Fuel enrichment (%)	Not applicable, uses ^{233}U derived from Thorium
Refuelling Cycle (months)	Continuous refuelling from ^{233}U produced in blanket
Reactivity control mechanism	Negative temperature coefficient; control rod insertion
Approach to safety systems	Passive
Distinguishing features	Complete consumption of thorium resource for energy generation
Design status	Conceptual Design

1. Introduction

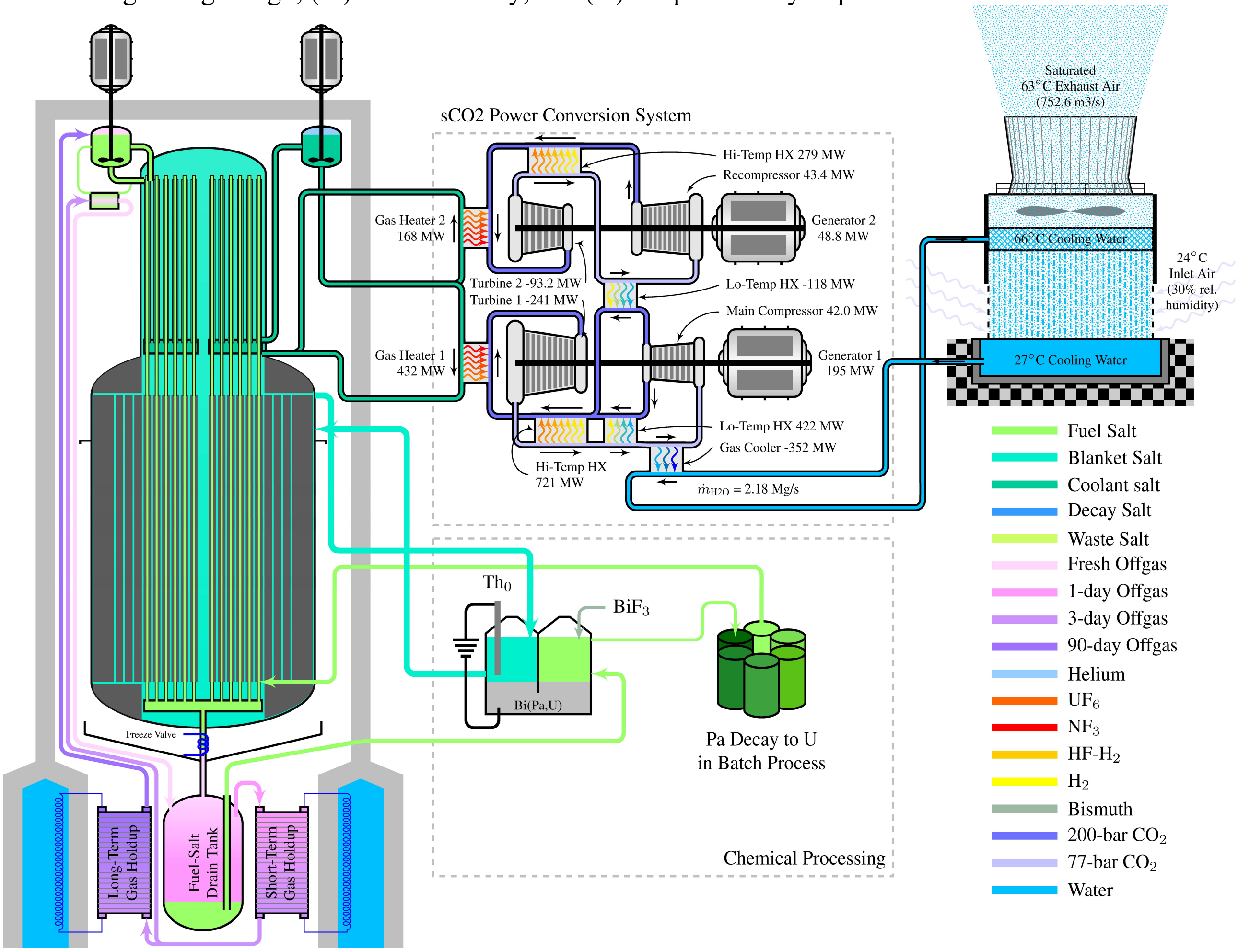
The lithium-fluoride thorium reactor (LFTR) design by Flibe Energy is a graphite-moderated, thermal-spectrum reactor with solutions of liquid fluoride salts containing both fissile and fertile materials. Thermal power generated from nuclear fission would drive electrical generation in a closed-cycle gas turbine power conversion system. The objective is to produce electricity at low cost by efficiently consuming thorium.
Mixtures of fluoride salts raised to a sufficient temperature to allow them to liquefy form an ideal medium in which nuclear fission reactions can take place. The ionically bonded nature of the salts prevents radiation damage to the mixture and allows for operation at high temperature yet at essentially ambient pressure.
The high operational temperatures of the fluoride salts (500-700°C) make them excellent candidates for coupling to a closed-cycle gas turbine power conversion system (PCS). The supercritical carbon dioxide gas turbine employing the recompression cycle is proposed and can generate electricity at high efficiencies (approximately 45%).
The LFTR design has a two-region core (feed/breed) and utilizes a closed fuel cycle based on thorium. The reactor vessel incorporates two plena with a central active core region and the outer blanket area, both filled with fluoride salt. The ^{232}Th in the blanket region is ultimately converted to ^{233}U through neutron capture and beta decay. The chemical processing system is used to separate and re-introduce the fertile and fissile material to the two fluorides fuel-salt streams respectively. Utilizing thorium fuel in a thermal neutron spectrum, the reactor can extract almost all the energy content thus assuring practically unlimited thorium resources and the associated insignificant basic fuel costs.

2. Target Application

Develop a power-generating nuclear reactor that will produce electrical energy at low cost by efficiently consuming thorium.

3. Design Philosophy

The objective of the liquid-fluoride thorium reactor (LFTR) design proposed by Flibe Energy is to develop a nuclear power plant that will produce electrical energy at low cost. By utilizing thorium fuel in a thermal neutron spectrum, the reactor can utilize the energy content of the thorium at a very high efficiency, and to a point where the Earth's thorium resources practically becomes unlimited. The main design principles are (i) inherent safety, with a no meltdown and non-pressurized core; (ii) simplicity, to have an intrinsically stable and self-regulating design; (iii) fuel efficiency, and (iv) the potentiality to produce far less waste.



4. Main Design Features

(a) Nuclear Steam Supply System

The nuclear heat supply and power conversion system is included in the simplified flow diagram above. It includes the reactor and primary loop, intermediate loop, power conversion system, and external cooling system. The individual systems are described in more details below followed by other system design descriptions.

(b) Power Conversion System

The function of the PCS is to convert the maximum amount of enthalpy contained in the heated working fluid into shaft work and to reject the remaining enthalpy to the environment in an acceptable manner. The supercritical carbon dioxide gas turbine employing the recompression cycle appears to be the best candidate for coupling to the reactor. The PCS includes four heat exchangers: the gas side of the gas heater, the gas cooler, and the high-temperature and low-temperature recuperators. It also includes the main turbine, the main compressor, the recompressor, and the electrical generator. The PCS interfaces with the intermediate loop through the gas heater, and interfaces with the external cooling system through the gas cooler.

(c) Reactor Core

The reactor vessel functions to hold fuel salt, blanket salt, and moderator material together in such a way to maintain a critical configuration at the temperatures and thermal power levels required. In addition, it incorporates reactivity control mechanisms both active and passive. The fuel and blanket salts are kept separated in two plena integrated into a single structure within the reactor vessel. Fuel salts are directed into the appropriate channels as it is circulated through the reactor.

The reactor vessel design incorporates several safety functions. In many accident events, a freeze valve, which

forms part of the vessel and primary loop system, melts and allows fuel salt to drain from the primary loop and the reactor vessel into the drain tank. The separation of the fuel salt from the solid graphite moderator retained in the reactor vessel, assures that a subcritical configuration can be established in the drain tank.
The internal graphite structures need to be replaceable since they are subject to a fast and thermal neutron flux that is greatly in excess of that which will be experienced by the metallic reactor vessel itself, and the replacement of these graphite structures will enable the reactor vessel to continue to operate and serve its function.

(d) Reactivity Control

The reactor vessel accommodates passive and active control rod systems which also have important safety functions. The blanket salt held within the reactor vessel is a strong neutron absorber, and a blanket salt leak from the reactor vessel could lead to the reduction in the blanket salt inventory contained in the reactor vessel, increasing reactivity by removing a neutron-absorbing medium. To compensate for this introduction of positive reactivity, a series of control rods that float in the blanket salt and are thus held outside of the core could be used. An accidental drain of the blanket salt would remove the buoyancy effect of these rods, allowing them to slide down into the core and add negative reactivity to replace and overcome the negative reactivity lost from by the drain of the blanket fluid. These rods would be designed to enter the core passively, without any operator action, in the event of blanket loss. But it is anticipated that there would also be an active drive system present that could drive these rods into the core intentionally in order to have a shutdown effect on the reactor. It would not be possible to start the reactor unless these rods were fully withdrawn from the core due to their strong negative reactivity.
An active set of control rods, of a more conventional design, would also be present in the reactor vessel and would serve a safety function, allowing the operator to control the reactivity level of the reactor. These rods, which would comprise a smaller and less potent source of negative reactivity, would be clustered near the centre of the core and provide finer control over reactivity levels. Other possible designs are also considered.

(e) Reactor Pressure Vessel

The reactor vessel shall be constructed from a material that is suitable for accomplishing its functions at the anticipated temperatures, stresses, and neutron fluxes that will exist during operation. Current evidence points to a modified form of Hastelloy-N as the suitable construction material.

(f) Primary Loop

The function of the primary loop is to direct fuel salt through the primary heat exchanger (PHX) in normal operation, where the fuel salt transfers its heat to the coolant salt. The primary pump provides the necessary forced circulation. The primary loop system includes the primary pump, the PHX (integrated in the reactor vessel), the bubble injection system, and the fuel salt drain tank and its associated external cooling system.

(g) Intermediate Loop

The intermediate loop transfer heat from the primary loop to the PCS. The intermediate loop system includes the PHX, the coolant salt pump, the salt side of the gas heater (or intermediate heat exchanger, IHX), the coolant salt drain tanks, and the pressure relief (blowout) valves. The intermediate loop also isolates the primary loop from the high pressures of the PCS using pressure relief valves. The isolation is an important safety function. In case of a failure in the high-pressure PCS it will prevent the transmittal of high pressure back through the coolant salt to the primary loop. The primary loop is not designed for high pressures and without isolation a break in the PCS could cause component rupture and potentially disperse radioactivity into the containment. In the event of a failure in the gas heater and the pressurization of the intermediate loop, the pressure relief valves allow coolant salt to leave the loop. This deprives the primary loop of cooling capability and will lead the melting of the freeze valve in the primary loop and the drain of the primary loop fluid contents into the fuel salt drain tank (also see passive shutdown and heat removal later).
The use of a coolant salt in this loop leads to a compact PHX, also reduces the fuel salt inventory, and thus the amount of fissile material needed for a given power rating.

(h) External Cooling System

The function of the external cooling system is to reject the heat that was not converted to shaft power in the PCS to the environment in an acceptable manner. The design shall also prevent the transmission of tritium to the outside environment.

(i) Fuel Characteristics

Thorium fuel is introduced as a tetrafluoride into the blanket salt mixture of the reactor. The blanket salt surrounds the active ‘core’ region of the reactor and intentionally absorbs neutrons in the thorium, which leads to the transmutation of the ^{232}Th via nuclear beta decay, first to protactinium-233 (^{233}Pa) and later to ^{233}U. Both the protactinium and the uranium are chemically removed from the blanket salt mixture and introduced into the fuel salt mixture in the reactor to fission. The fission products are later chemically removed from the fuel salt and in some cases separated and purified before final disposition.

(j) Fission Product Retention

The integrity of the reactor vessel plays an important role in minimizing radiation hazards by confining radioactive fluids to the flow channels and volumes defined by the vessel and its internal structures.
Most fission products, including all of those of greatest radiological concern, form stable fluoride salts that are retained in the overall mixture under all conditions. Fission products gases, whose removal is important from a performance and safety basis, are easily separated from the fluid mixture and allowed to decay to stability in a separate system.

5. Safety Features

(a) Passive Shutdown and Heat Removal

An important safety function is embedded in the primary loop and is activated when the reactor overheats or loses its coolant flow. A freeze valve is integrated into the primary loop that is maintained frozen by an active coolant system. When this coolant is lost or if the temperature of the system exceeds its cooling capability, the freeze valve fails open and the fuel salt drains out of the primary loop and out of the reactor vessel into the fuel salt drain tank. The fuel salt drain tank is integrated with a separate cooling system that is passively connected to the outside environment and provides the necessary cooling for the fuel salt within it.

(b) Fluoride Salt Characteristics

The fluoride salt mixtures in question have high volumetric heat capacity, comparable to water, and do not undergo vigorous chemical reactions with air or water in contrast to many liquid metals. The components of fluoride salt mixtures have both desirable and undesirable aspects, and the two most important are lithium-7 fluoride and beryllium fluoride. The two natural isotopes of lithium must be separated from one another since Li_6 (7.5% of natural lithium) is far too absorptive of neutrons to be a suitable component of a reactor fluid. Beryllium fluoride is chemically toxic but has very attractive nuclear and physical properties. The chemical processing and purification of fluoride salt mixtures typically involves using powerful reactants such as gaseous fluorine and hydrogen fluoride which are very toxic and reactive. But the fact that fluoride salt mixtures are processed in a salt form rather than being dissolved into an aqueous solution mitigates issues of accidental criticality considerably, since water is an excellent moderator whereas salts are poor.
Fluoride salts, due to their exceptional chemical stability, have the potential to corrode most structural metal alloys, but some alloys have been developed that hold up very well against any corrosive attack. Invariably these alloys are based on nickel with a variety of other metallic constituents. Fluoride salts moderate neutrons sufficiently on their own to prevent the formation of a truly fast neutron spectrum, but are still insufficiently effective to generate a thermal neutron spectrum. Thus, separate moderator materials are necessary for the reactor and graphite has been proven to be very attractive.
Graphite is not wet by the fluoride salts and does not require cladding. If the surface of the graphite is treated so that small pores are closed, most fission product gases can be excluded from the graphite and overall performance will be high. Graphite does experience issues from dimensional distortion over time, but this effect can be quantified and compensated for in reactor design.

6. Plant Layout Arrangement

The reactor cavity or silo is below grade and contains the primary circuit.

7. Design and Licensing Status

The design is in an early stage of development and licensing activities have not yet been undertaken.

8. Fuel Cycle Approach

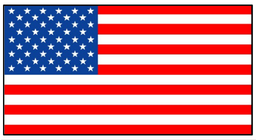
The LFTR two-region core facilitates feed and breed and it utilizes a closed fuel cycle based on thorium.

9. Waste Management and Disposal Plan

LFTRs have the potential to produce far less waste than LWRs along the entire fuel cycle and process chain, from ore extraction to nuclear waste storage. LFTR technology can also be used to consume the remaining fissile material available in spent nuclear fuel stockpiles around the world and to extract and resell many of the other valuable fission byproducts that are currently deemed hazardous waste.

10. Development Milestones

October 2015	EPRI-funded study of LFTR design published
July 2018	DOE award announcement for advanced fluorination development work



ThorCon (ThorCon International, United States of America and Indonesia)

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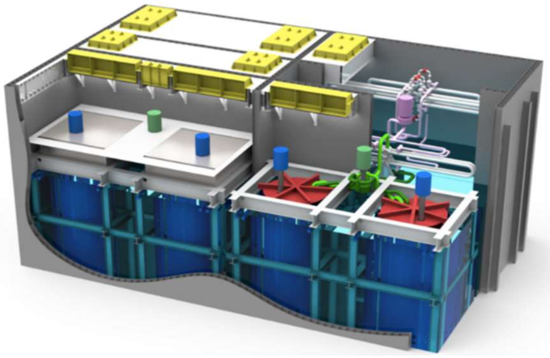
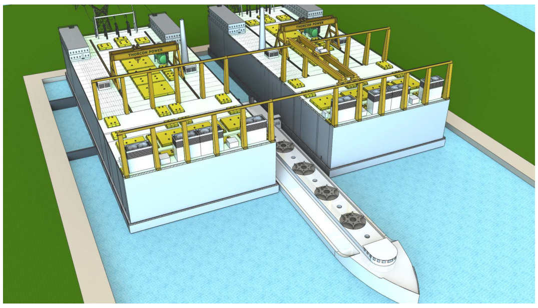
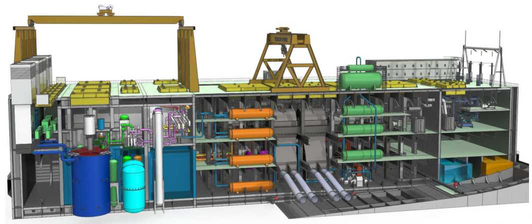
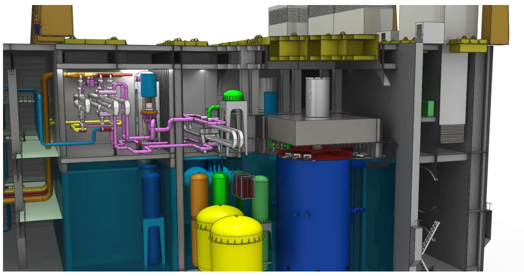








MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ThorCon International, first deployment in Indonesia
Reactor type	Thermal molten salt reactor
Coolant/moderator	Molten salts / graphite
Thermal/electrical capacity, MWt/MWe	557 / 250 (per module)
Primary circulation	Forced (4 pumps), 1 per loop
NSSS operating pressures (4 loops: fuel salt, clean salt, solar salt, steam), MPa absolute	1.06 / 1.30 / 0.56 / 25.7
Core Inlet/Outlet fuel-salt temperature (ºC)	560 / 704
Fuel type/assembly array	UF_4 / molten salt
Power conversion process	Supercritical steam turbine
Fuel enrichment (%)	Startup 2.3 / makeup 4.95
Discharge burnup (GWd/ton)	145.8
Refuelling cycle (months)	12 months to adding fuel 48 months to Can changeover
Reactivity control mechanism	Negative temperature coef.
Approach to safety systems	Intrinsic, passive
Design life (years)	80
Plant footprint (m^2)	67x162
Can height/diameter (m)	10.3 / 7.8
Can weight (metric ton)	343
Seismic design (SSE)	1.0
Fuel cycle requirements / approach	Fissile LEU05, or LEU19 with thorium conversion
Distinguishing features	Full passive safety, short construction time, low cost
Design status	Preliminary design

1. Introduction

ThorCon is a molten salt fission reactor. Unlike all current operating reactors, the fuel is in liquid form. The ThorCon reactor operates at near atmospheric pressures and is constructed using automated, ship-style steel plate construction methods.

The top picture shows two pairs of 557 MWe reactor Cans in (blue) silos within the hull wall. Each 250 MWe power module contains two replaceable reactors in sealed Cans, depicted in red, which are inside the silos. One Can of each module produces power while the other is in cooldown mode. After four years the cooled Can is replaced with a fresh Can, the fuel salt transferred to it, and the used Can starts its 4-year cool down.

The second picture above shows two 500 MWe ThorCon power plants and a CanShip exchanging the Can containing the reactor vessel and radioactive primary loop. The yellow rectangles are hatches for access by gantry cranes. The middle graphic shows the air cooling towers, fission island, heat exchangers, steam turbine-generator, and switchgear. The bottom graphic also shows the steam generation cell and secondary heat exchanger cells, flooded in water used as a backup decay heat sink.

2. Target Application

The first planned application of ThorCon reactors is to generate electric power in developing nations with fragile grids, so ThorCon is capable of demand discontinuities and black start without grid power. Capital cost and generated electricity costs are critical in these markets. ThorCon is cheaper than coal and deploys rapidly.

3. Design Philosophy

Walkaway Safe – If the reactor overheats for any reason, it will automatically shut itself down, drain the fuel from the primary loop, and passively remove the decay heat. Operators cannot prevent draining and cooling. ThorCon has three gas tight barriers between the fuel salt and the environment. In a primary loop rupture, there is no coolant phase change and no dispersal energy..

Ready to Go – The ThorCon design needs no new technology. ThorCon is a scale-up of the successful Molten Salt Reactor Experiment (MSRE). A full-scale dual 250 MWe ThorCon prototype can be operating under test in 2026 and then subjected to failure testing before commercial production can begin.

Rapidly Deployable – The entire ThorCon plant is designed to be manufactured in blocks in a shipyard. The 150 to 500 ton, fully outfitted, pre-tested blocks are assembled into a hull containing the complete power plant, towed to a customer site, and firmly settled in 5-10 m of water. A single large reactor yard can turn out twenty 500 MWe ThorCons per year. ThorCon is more than a power plant; it is a system for building power plants.

Fixable – No complex repairs will be attempted on site. Hatches and cranes permit components of the fission island to be replaced. The primary loop is totally contained within a Can. Every four years the Can is changed out, returned to a centralized recycling facility, decontaminated, disassembled, inspected, and refurbished.

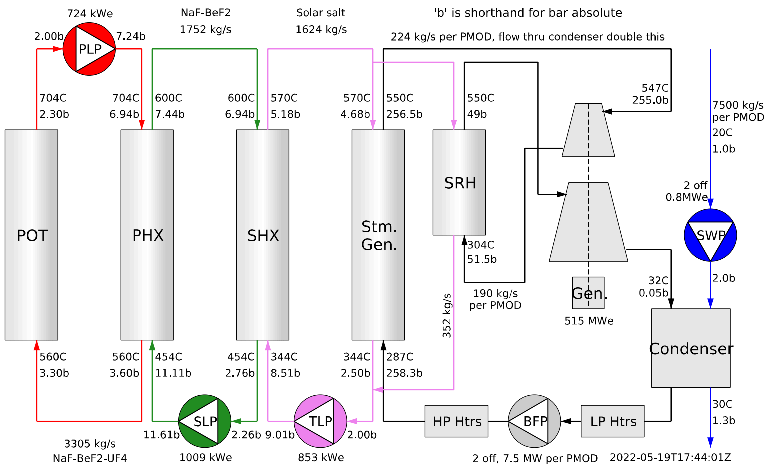
Cheaper than Coal – ThorCon requires far fewer resources than a coal plant. Assuming efficient, evidence-based regulation, ThorCon will produce clean, reliable, carbon-free electricity at less than the cost of coal.

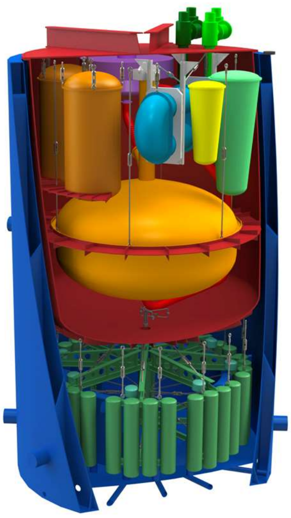
4. Main Design Features

(a) Power Conversion

Each power module employs four loops for converting fission heat to electric power: the primary fuel salt loop inside the Can, the secondary salt loop, a solar salt loop, and a supercritical steam loop.

The solar salt loop captures any tritium that has made it past the secondary loop, and more importantly ensures that a rupture in the steam generator creates no nasty chemicals and harmlessly vents to the Steam Generating Cell via an open standpipe.





ThorCon is a high temperature reactor, with a thermal efficiency of 46.5% using 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.

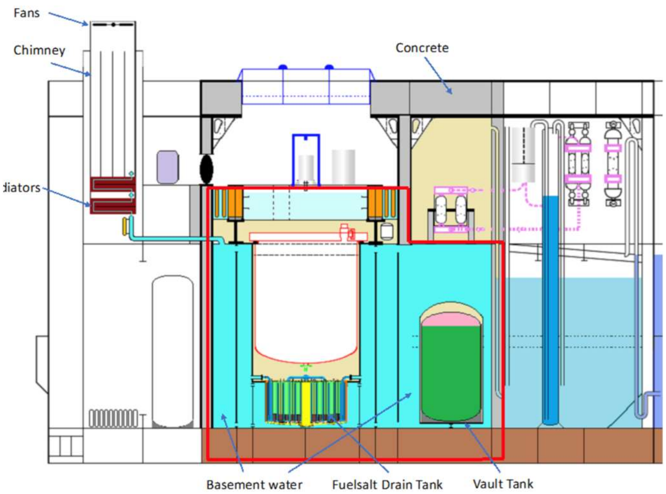
The fuel salt is a mixture of sodium, beryllium, and uranium fluorides at 704ºC. The (red) Can contains the (orange) reactor called the Pot. The primary loop pump pushes the fuel salt at 3300 kg/s through the (blue) primary loop heat exchanger (PHX).

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 ^{238}U to fissile fuel.

The steel silo cold wall shown in dark blue is cooled by surrounding water. The Can is cooled by thermal radiation to the silo cold wall. The heated water rises by natural circulation to above-deck, air-cooled radiators, where it is cooled and returns to the bottom of the cooling wall via the basement.

A loop of hot secondary salt of mixed NaF and BeF2 is pumped through the PHX in the Can to a Secondary Heat Exchanger (SHX). It transfers heat to a third loop of NaNO3/KNO3, called solar salt because of its use in thermal solar power plants. The solar salt, shown in purple, transfers heat to a supercritical steam generator and steam reheater.

All radioactive fission products and fissile material are contained within the Consolidated Boundary indicated by the red outline. The space is shielded by steel sandwich walls protecting the crew and prevents intrusion. This space is guarded by IAEA seals and monitoring.



(b) Reactor Core

The reactor core is inside the pot. The core is 90% filled with graphite logs that moderate neutron energies. The core is 6.5 m in diameter and 4 m high.

(c) Reactivity Control

Reactivity is controlled by fuel salt temperature, which rises as steam turbine power generation heat demand drops. Fission decreases as salt, graphite, and Pot temperatures rise. Temperature adjustment rods compensate for neutron absorption by xenon during power changes.

(d) Reactor Pressure Vessel and Internals

The Pot reactor vessel pressure is 0.33 MPa, insufficient to spread radioactive material into the environment in an accident, so it does not have the central safety importance that it does in an LWR. The Pot contains circulating fuel salt, graphite moderator, shutdown rod channels, and temperature adjustment rods.

(e) Fuel Characteristics

The fuel salt is $NaF\text{-}BeF_2\text{-}UF_4$ 72/16/12. As fissile U is consumed fissile Pu_{239} is generated, but not enough to replace the fuel burned. Makeup fuel must be added continuously.

(f) Fission Product Management

Fission product off-gases include xenon and krypton, mixing into helium cover gas flowing over the fuel salt surface in the Pot header tank. The off-gas mix flows through tanks inside the Can to allow most of the radioactive gases to decay to solid fission products which are captured within. The cooled off-gas flows to holdup tanks and charcoal beds in the SHX to allow the longer lived radioactive noble gas to decay except Kr-85 and tritium. The xenon and krypton are removed to tanks and the helium recycled.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The ThorCon negative temperature coefficient provides passive temperature stability. The large margin between the operating temperature of 704°C and the fuel salt boiling temperature of 1430°C exceeds possible temperature excursions, so radioactive material cannot be vaporized.

(b) Decay Heat Removal System

The FDTs radiate fuel salt decay heat to steel jackets cooled by the same naturally circulating water circuit used to cool the cold wall, passively removing the decay heat to the air cooling towers. No operator intervention is needed at any time. No electric power is necessary, though fans keep basement water cooler. Cooling using natural air flow continues indefinitely.

(c) Emergency Core Cooling System

If the air-cooled radiators in both cooling towers become disabled, basement water provides passive cooling for at least 40 days. Cooling time can be extended indefinitely by supplying more water from sources such as the desalination system, domestic water, ballast tank water, or external water via on-deck fittings.

(d) Containment System

The primary loop is totally contained within the removeable Can and FDT. This system constitutes primary containment for fuel salt during power operations and accidents from power operations. The primary containment for spent fuel during transfer and storage are double walled piping and tanks, also within the consolidated boundary secondary containment. These first two containments are gas tight. Tertiary containment is provided by the ship hull structure of three meters of concrete sandwiched between 25 mm steel plates.

(e) Chemical Control

Makeup UF_4 fuel enriched to 5% ^{235}U is added continuously to increase reactivity. Fertile $^{238}UF_4$ can be added to decrease reactivity. Beryllium metal is added to maintain proper redox potential as fluorine is freed via UF_4 fission. No boron additions occur.

(f) Spent Fuel Cooling Safety Approach/System

ThorCon has two gas tight barriers and one containment barrier between the fuel salt and the environment. ThorCon reactor operates at near-ambient pressure. In the event of a primary loop rupture, there is little dispersal energy and no phase change and no vigorous chemical reactions (like zirconium and steam).

(g) Spent Fuel Salt

Operating with uranium fuel in feed-bleed mode, continuous addition of makeup fuel salt causes removal of used fuel salt from the primary loop to an overflow tank in the Can. Periodically, fuel salt in the overflow tank is transferred to a vault tank. The removed, cooling fuel salt is as well protected as the fuel salt being fissioned. Spent fuel salt may later be transferred to a shipping cask and transferred to the visiting CanShip and shipped to a fuel salt handling facility for future recycling.

6. Plant Safety and Operational Performances

Load following relies on reactor physics. As electric power demand decreases, less heat is removed from the fuel salt by the PHX and reactivity and power decrease as temperature rises.

7. Instrumentation and Control Systems

Instrumentation and control systems are not safety critical for ThorCon. Argonne National Lab is adapting its isotopic concentration sensors to monitor ThorCon fuel salt components. Numerous sensors will record and report the condition of power generation..

8. Plant Layout Arrangement

Two 557 MWt power modules drive a single 500 MWe turbine/generator. This allows using competitively priced, efficient supercritical steam turbine-generators.

9. Testing Conducted for Design Verification and Licensing

Measurements of molten salt properties are being conducted at independent laboratories. Soil testing has been completed in the bay of the island site for the Indonesia demonstration plant. The non-fission test platform is being designed to test components such as molten salts, sensors, pumps, valves, pipes, vessels, graphite moderators, and heat exchangers at operating temperatures and pressures, using externally supplied electric power rather than fission power. This facility will be used to verify safety systems including performance for containment and cooling during severe accidents. Neutronics testing will wait for the demonstration plant.

10. Design and Licensing Status

Preliminary design is complete. Some detailed designs are being discussed with specialty component suppliers. License discussions are continuing with the Indonesian regulator, Bapeten, along with university professors and government officials.

11. Fuel Cycle Approach

Fuelled by 5% enriched uranium additions, ThorCon can operate in feed and bleed mode, continually feeding in fresh fuel salt and bleeding excess volume to the onboard storage vault tanks. Once HALEU becomes available in power plant level quantities, a 500 MWe ThorCon can operate as a thorium converter using 5.3 kg of 19.7% enriched uranium and 9.0 kg of thorium per day, on average. After each 8-year fuel cycle, the used fuel salt is transferred to the vault tanks, which have capacity for 80 years of this operation.

12. Waste Management and Disposal Plan

Spent fuel salt is transferred to passively cooled vault tanks in the SHX cell for storage up to 80 years. Once appropriately cooled, the fuel salt can be transferred to a shipping cask and removed by crane and loaded to a CanShip to transfer it for reconditioning or final storage.

13. Development Milestones

2022	Pre-licensing vendor design review in Indonesia; Preliminary design complete
2023/24	Construction of Non-fission Test Platform; Testing of the Non-fission Test Platform
2025/26	Construction of the demonstration power plant; Begin testing of the demonstration power
2027	Complete testing of the demonstration power plant and obtain type license
2028	Begin commercial construction of multiple power plants; Start of commercial operation

PART V. MICROREACTORS

Energy Well (Centrum výzkumu Řež s.r.o., Czech Republic)

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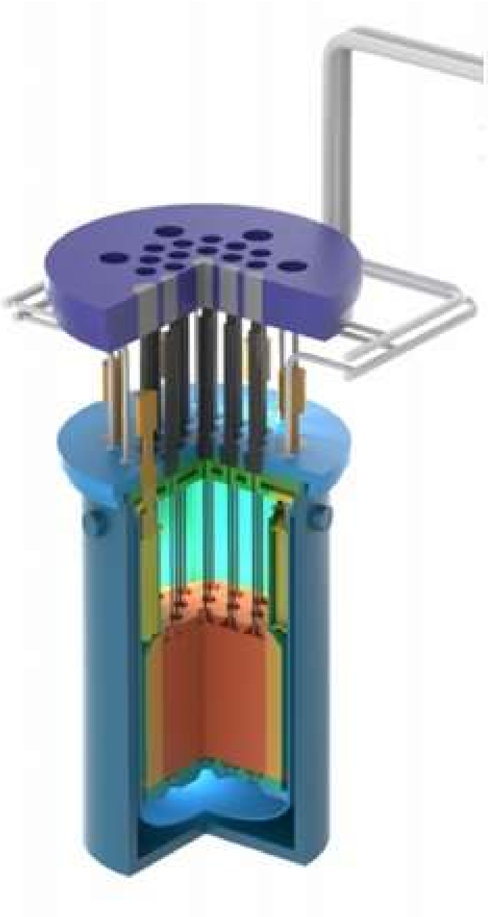


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Centrum výzkumu Řež Czech Republic
Reactor type	Fluoride high temperature reactor
Coolant/moderator	Molten salt FLiBe
Thermal/electrical capacity, MW(t)/MW(e)	20 / 8
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	Atmospheric
Core inlet/outlet coolant temperature (ºC)	650 / 700
Fuel type/assembly array	TRISO
Number of fuel assemblies in the core	25
Fuel enrichment (%)	15
Refuelling cycle (months)	84
Core discharge burnup (GWd/ton)	70
Reactivity control mechanism	Control rods
Approach to safety systems	Active/passive
Design life (years)	Not defined
Plant footprint (m^2)	< 4000
RPV height/diameter (m)	7 / 3.5
RPV weight (metric ton)	< 100
Seismic design (SSE)	Yes
Fuel cycle requirements/approach	Once through
Distinguishing features	High passive safety features
Design status	Pre-conceptual design

1. Introduction

Energy Well is a Fluoride High temperature micro Reactor of 20 MW(t) under development using unique knowledge in the Czech Republic on molten salt technologies. The project is under grants from the Ministry of Industry and Trade. The purpose of the project is to develop an advanced, inherently safe low-power high-temperature reactor. The design is mainly intended for remote areas as a long-term source of electrical energy and heat for island networks. Therefore, the reactor and associated power plant and/or a heat plant must meet the following requirements, among others: 25 MW(t) maximum power; transportable; long fuel cycle; fuel enrichment < 20%; and thermal efficiency > 40%.

2. Target Application

Energy Well is focusing on operation both in remote and in populated areas, focusing on production of electricity, heat and hydrogen as a means of energy storage. The purpose is to provide a clean stable energy source in synergy with the large-scale nuclear power reactors, heating plants and the renewable sources of energy.

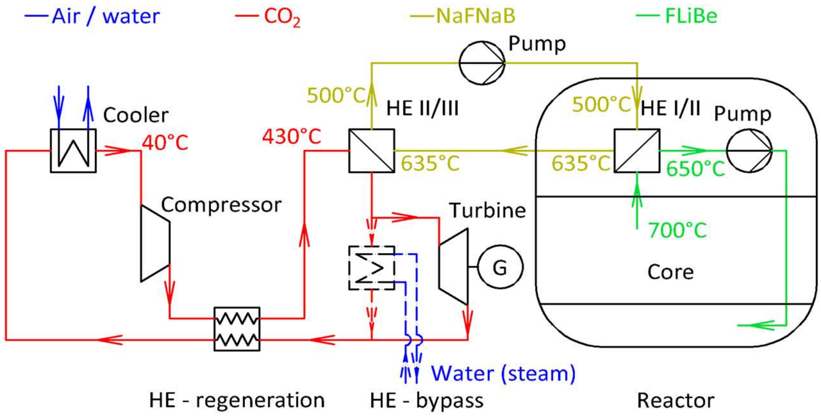
3. Design Philosophy

The 20 MW(t) Energy Well adopts a 7-year fuel cycle, low power density, high use of passive safety and simplicity. The design allows a modular approach and requires minimum on-site welding operations.

4. Main Design Features

(a) Power Conversion

The power plant includes three cooling circuits. Liquid fluoride salts are used as a heat transfer medium (FLiBe, NaBF4) in the primary and secondary circuits. Carbon dioxide in a supercritical state (sCO2) is used in the tertiary circuit. The tertiary circuit considers an Ericsson–Brayton-based cycle optimized configuration for transformation of the heat to electric power. The primary circuit removes the heat generated in the core of the reactor, while the secondary circuit separates the active primary and the high-pressure tertiary circuit while ensuring the heat transfer.



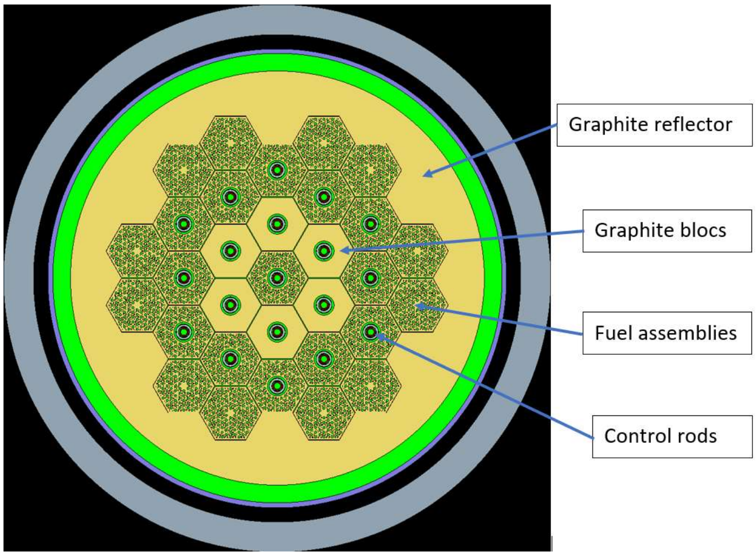
Process Flow Diagram of Energy Well reactor with simple heat recovery system

(b) Reactor Core

The reactor core includes, 25 hexagonal fuel assemblies, 6 central graphite blocs, an external graphite reflector and 19 control rods. The core is 2 m high and has a diameter of 3 m with the reflector. The current core arrangement is shown in the figure on the right.

(c) Fuel Characteristics

The Energy Well core fuel assemblies are made of TRISO spherical fuel particles that are blended together to form fuel compacts. TRISO coated fuel particles are composed of a uranium kernel coated in successive layers of pyrolytic carbon, silicon carbide, and an outer layer of pyrolytic carbon. The fuel compacts are stacked in a drilled hexagonal graphite block that also contains holes for molten salt coolant flow. A fuel enrichment of 15% has been selected.



Core Arrangement

(d) Reactivity Control

The primary safety system ensuring reactivity control are the control rods with a set scram signal. Scram is activated when the neutron flux or other technological parameters in the core are increased. During normal operation, control rods are kept in operational position by magnets that are power fed. In case of deviation from normal operation parameters, the Limiting System (LS) regulates the reactor power. A secondary independent shut-down system is currently being studied and shall be implemented in the design.

(e) Reactor Pressure Vessel and Internals

Energy Well is a pool type reactor with molten salt FLiBe as the primary coolant. The primary cooling circuit includes the following main parts: a reactor core with top mounted control rods, a reactor vessel with cover lid, a graphite reflector, a core supporting plate, a flow skirting, six primary heat exchangers molten salt/molten salt and main circulation pumps.

The molten FLiBe flows upward through the core and then, guided by the flow skirting enters the main circulation pumps that circulate the molten salt to the six primary heat exchangers located on the periphery of the reactor vessel above the core. At the outlet of the heat exchangers the molten salt flows in a annular space between the flow skirt and the reactor vessel downward through the core supporting plate.After passing through the core supporting plate, the FLiBe is directed upward to enter the bottom of the core.

(f) Reactor Coolant System

The primary coolant is a molten fluoride salt containing lithium and beryllium (Li2BeF4— referred to as "FLiBe"). It has a melting point of 459°C, a boiling point of 1430°C, and a density of 1.94 g/cm3 . The heat capacity of Flibe is 4540 kJ/m3 , which is similar to that of water, more than four times that of sodium, and more than 200 times that of helium (at typical reactor conditions). There is also considerable experience with FLiBe in nuclear systems since it was used in both the primary and secondary loops of the Molten Salt Reactor Experiment (MSRE) and related test loops. The relatively high melting point of FLiBe will require special design features. Experience feedback from sodium-cooled fast reactor, lead cooled submarine reactor and MSRE is being considered during the Energy Well development. The use a pool-type reactor vessel shall reduce some of the challenges related to molten salt freezing in the primary circuit. Molten salts are also transparent which is an advantage during refuelling, maintenance operations and inspection.

(g) Secondary System

The secondary circuit physically separates the primary circuit from the tertiary circuit and creates a pressure barrier in case of leaks in the salt/salt exchanger between the primary and the secondary circuit. The secondary circuit includes 3 main components: a salt/salt heat exchanger; a salt/sCO2 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 sCO2 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.

(h) 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.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The design of Energy Well reactor has a high focus on passive safety and simplicity. The main safety features of the reactor include: Atmospheric pressure in primary and secondary circuits; primary circuit is underground; low power density; and use of natural circulation and passive safety systems.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

Passive residual heat removal from the primary circuit through the reactor vessel is adopted in case of loss of flow. In the initial phase of event, the high heat capacity of molten salt will allow the heat output to accumulate in the primary coolant until the heat losses exceed the residual power. Maximum fuel temperature reaches 730°C.

(c) Spent Fuel Cooling Safety Approach / System

The spent fuel cooling system is still to be defined.

(d) Containment System

The TRISO fuel envelope is the first barrier to prevent the spread of fission products. Depending on the design, the fuel assembly could be considered as a barrier since the TRISO fuel is enclosed in a thick graphite matrix. The reactor vessel, the containment vessel, the pit (together with the maintenance room shielding ceiling) and the reactor building are additional barriers of the containment system.

(e) Chemical Control

The chemical control of the molten salts in the primary and secondary circuit is still to be defined. Feedback experience from MSRE operation shall be considered.

6. Plant Safety and Operational Performances

Air or water are foreseen as heat sink for the facility. The facility is foreseen to be operated at constant power with a refuelling performed on-site every 7 years The refuelling is performed by a fuel handling machine similar to the one used at the Fort Saint Vrain reactor. Fuel handling equipment is designed to be portable and will not be left on site when not needed to prevent access to fissile material.

The main circulation pumps and the primary heat exchangers connecting flanges are bolted to the reactor vessel lid. The limited size of the primary circuit equipment and the location of the primary circuit connecting flanges above the molten salt level shall facilitate the maintenance operations.

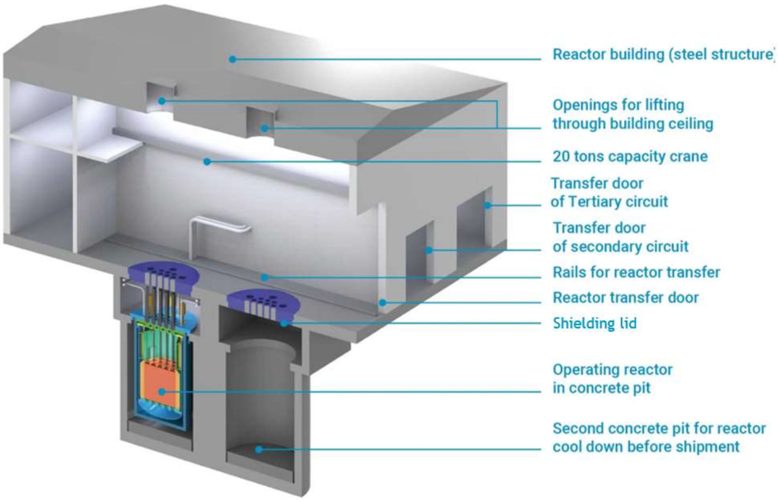
To target markets with a demanding power up to 100/200 MW(t) a multiple unit approach is foreseen.

7. Instrumentation and Control System

The I&C system is currently under development and being tested on the experimental loops currently operating in the Centrum výzkumu Řež s.r.o.

8. Plant Layout Arrangement

The Energy Well reactor building layout is shown in the figure on the right. The primary, secondary and tertiary circuit are located in a common steel building. The three circuits are located in separate rooms with separate transfer doors. The reactor vessel with the primary circuit is located in a concrete pit in the reactor room. Additional pits can be added in the reactor room to implement more reactor units and increase the delivered power. The reactor room is equipped with an overhead crane to handle shielding lids, transfer casks and equipment. Control rods actuators and 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.



3D view of Energy Well reactor building

9. Testing Conducted for Design Verification and Validation

Material corrosion test facilities are built and currently in operation in Centrum výzkumu Řež. A forced circulation loop with molten FLiBe is currently being built to verify thermohydraulic parameters.

10. Design and Licensing Status

As of 2022 the Centrum výzkumu Řež is preparing a basic design of the Energy Well reactor including experimental tests. Build an integral testing facility is planned. The Centrum výzkumu Řež is in close coordination with the State office for Nuclear Safety.

11. Fuel Cycle Approach

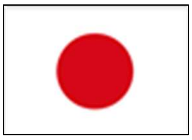
A once-through scheme approach is foreseen for the fuel. Once the 84 months fuel cycle accomplished, the spent fuel is removed from the core and transfer to the on-site spent fuel storage area. Fresh fuel is shipped to the site, inspected and then installed in the reactor vessel with the fuel handling machine.

12. Waste Management and Disposal Plan

The waste management and disposal plan are under development.

13. Development Milestones

2010 2017	Basic research in the neutronics, thermohydraulics and material compatibility in regard to the FLiBe salt
2017 - 2022	Pre-conceptual design phase and technology validation
2022 - 2027	Basic design
2027 - 2032	Experimental verification using an integral test facility
2032 - 2040	Finalization for manufacturing



MoveluX (Toshiba Energy Systems & Solutions Corporation, Japan)

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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)
Thermal/electrical capacity, MW(t)/MW(e)	10 / 3 – 4
Primary circulation	Natural
NSSS operating pressure (primary/secondary),	0.1 /0.3
Core inlet/outlet coolant temperature (ºC)	Heat pipe: 680 / 685 Heat Exchanger: 450 / 680
Fuel type/assembly array	Silicide (U_3Si_2) / Hexagonal
Number of fuel assemblies in the core	66 (fuel), 72 (CaH_2)
Fuel enrichment (%)	4.8 – 5.0
Refuelling cycle	Continuous
Core discharge burnup (GWd/ton)	1.0
Reactivity control mechanism	In-Ga Expansion Module (IGEM)
Approach to safety	Active / Passive
Design life (years)	10 – 15
Plant footprint (m^2)	100
RPV height/diameter	2.0 / 6.0
RPV weight (metric ton)	TBE
Seismic design (SSE)	0.3 g
Fuel cycle requirements/approach	Either once-through or closed fuel cycle scheme depending on country's fuel cycle policy
Distinguishing features	Use inherent passive natural principles for reactor shut down by moderator material property and decay heat removal from the surface of the reactor vessel by natural circulation of air
Design status	Conceptual design

1. Introduction

MoveluX, Mobile-Very-small reactor for Local Utility in X-mark, is a 10 MW(t) class multi-purpose micro reactor. A heat-pipe is used as a primary core cooling that provides passive safety as well as system simplification. MoveluX uses low enriched uranium fuel of less than 4.99 wt% that improves nuclear security and non-proliferation. Moderator material is required to reduce the core size. In addition, high temperature operation is essential for the multi-purpose micro reactor. Therefore, calcium-hydride (CaH_2) capable of operating at up to 800°C is adopted for the moderator material.

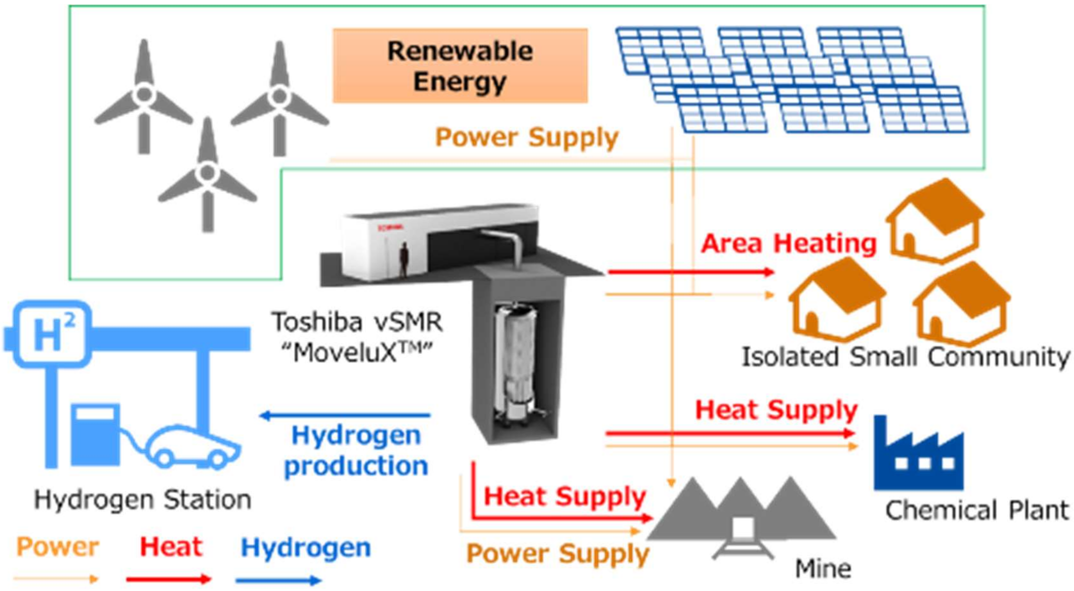
2. Target Application

The MoveluX reactor system is a multipurpose energy source that can be used to produce electricity, hydrogen and high temperature heat. Since the reactor system can be installed in remote sites, the heat can be provided for chemical plants and steel mills. When it is used as a power plant it can be used as a base load power source on small grids, possible in combination with renewable energy sources. Since MoveluX generates around 3 –

4 MW(e) and it can also be used for off-grid applications in remote places.

3. Design Philosophy

The MoveluX reactor system is designed as a multi-purpose energy source that includes off-grid and micro-grid electricity, high temperature heat source, hydrogen production and so forth. Designed to produce 10 MW(t), it can be used also for electricity production, possibly varying the output. Therefore, when the MoveluX reactor system is connected to a small/micro grid with load following as required and achieved by the passive reactivity control system. The figure below shows the MoveluX reactor system as a multi-purpose energy source in such a micro grid.
The major provisions of the MoveluX reactor for resource use optimization are as follows:
- Simplified plant design contributes to wastes reduction during operation and decommissioning.
- Low maintenance requirement using no-moving parts component contributes to low maintenance costs/labours and low waste amount.
- Reduced emergency planning zone contributes to accident management burden and/or cost such as for evacuation.



Utilization image of the MoveluX

4. Main Design Features

(a) Power Conversion

The MoveluX is considering using the gas turbine system for power conversion. The MoveluX can provide a high temperature of around 680°C, therefore, the gas turbine was selected from the viewpoint of the conversion efficiency.

(b) Reactor Core

The MoveluX core consists of the fuel, moderator, heat-pipe and control devices. In this core, uranium silicide and calcium-hydride were used as a fuel and moderator materials. The maximum fuel enrichment is set as 4.99 wt%, from the viewpoint of economics and non-proliferation. Fuel components are installed to the core in the fabrication phase and loaded as a lifetime core, i.e. this fuel will not be extracted from the core and therefore fuel handling is not required during operation. The fuel and moderator has 10 cm wide-hexagonal shape except for the sides where provision is made (cut off is made) for the heat-pipe installation.

(c) Fuel Characteristics

The Uranium silicide fuel has high melting point and large thermal conductivity.

(d) Reactivity Control

A safety-rod is placed at the centre of the core for criticality safety assurance and reactor start-up. The core reactivity is controlled by the passive reactivity control device, such as In-Gd Expansion Module (IGEM).
In the case of emergency such as loss of cooling ability and/or RIA (with a postulated failure of the safety rod insertion) the core will be shut down autonomously by the material property of calcium-hydride. Specifically, calcium-hydride ability to moderate neutrons (neutron moderation power) decreases with increasing temperature since hydrogen dissociates in high temperature environments above 800°C. The proposed core design therefore has inherent safety for criticality in emergencies. If the hydrogen pressure is increased by its dissociation, hydrogen will be processed at a hydrogen processor (details still to be examined).

(e) Reactor Pressure Vessel and Internals

Reactor vessel with atmospheric pressure is adopted.

(f) Reactor Coolant System

The primary circuit of the MoveluX reactor system makes use of heat-pipes which is one of the passive cooling devices, therefore, the primary circuit does not have pumps or other forced circulation devices. In the current design, sodium is selected as a working fluid of the heat-pipe from the viewpoint of usable temperature and heat transportability. The pressure in the primary system can be set close to atmospheric pressure since the proposed system does not utilize a pump for primary fluid circulation. Therefore, the risk of large-scale radioisotopes release can be reduced.

(g) Secondary System

The secondary side of the MoveluX reactor system is currently a helium gas system. This gas system can provide high temperature around 700°C and therefore usable not only for electric power generation, but also for heat supply, hydrogen production and so on. For electricity generation a Brayton cycle can be used as the power generation system.

(h) Steam Generator

The MoveluX not uses the steam generator in the current design, however, this option can be examined.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The reactor vessel is place below ground to enhance reactor protection and radiation shielding. The radiation shield on top of the reactor is protect the reactor from physical attacks. For the natural hazards, the MoveluX reactor protection is by the core inherent safety characteristics based on the natural principles.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

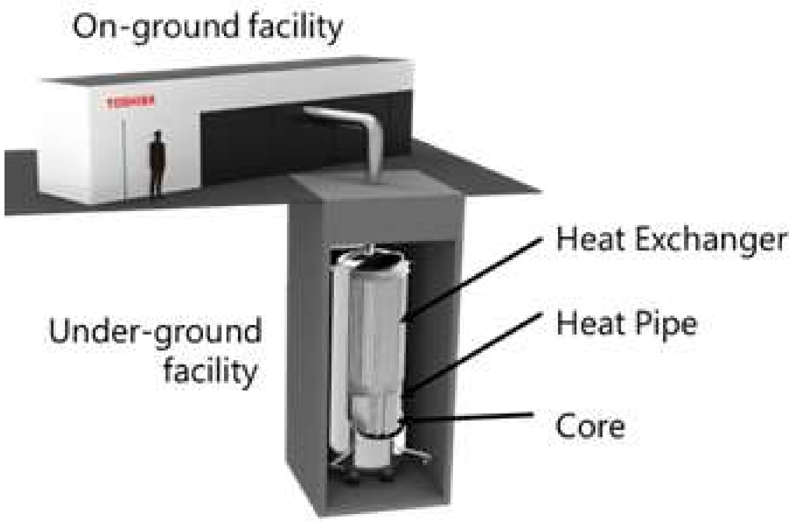
The decay heat after reactor shut down is removed passively. If heat-pipe and secondary circuit are keeping their function, decay heat removal are the same as for heat removal during nominal operation. In the case of loss of cooling ability of the heat-pipe, the decay heat is removed from the surface of reactor vessel by natural circulation of air. Furthermore, in the reactor vessel, the heat is transported from the centre to the periphery of the core by natural circulation and thermal conduction of liquid Pb-Sn, present in the gap between the fuel, heat-pipe and moderator. This decay heat removal system does not require a power source and can therefore realize a long (infinite) grace period. Consequently, the risk of core meltdown is expected to be very small.

(c) Containment System

The fuel material is contained in the reactor vessel. The fuel material is separated from primary cooling system, because, heat-pipe is closed heat transportation device. Additionally, the heat-exchanger between heat-pipe and secondary circuit is functioning as one of the boundaries. Therefore, radionuclide will be confined in the reactor vessel unless reactor vessel breaking.

(d) Chemical Control

The MoveluX reactor does not required chemical control.



Schematic view of the MoveluX reactor 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 System

Few I&C devices are installed to the MoveluX reactor system for the reactor start-up, monitoring and active control. Technically, the manned operation is not required during nominal operation in current design concept.

8. Testing Conducted for Design Verification and Validation

To be developed.

9. Design and Licensing Status

MoveluX is at the conceptual design stage.

10. Fuel Cycle Approach

The MoveluX reactor can be applied to either once-through fuel cycle scheme or closed fuel cycle scheme. It mainly depends on user country's fuel cycle policy.

11. Waste Management and Disposal Plan

The reactor vessel contains the spent fuel as it is, additionally, this spent fuel is carried out to temporary storage site with reactor vessel. After that, treatment of the spent fuel depends on the country's fuel cycle policy. On one hand, in the once-through scheme, spent fuel is extracted from the core at the facility, then, spent fuel is stored to the cask for disposal. On the other hand, in the closed fuel cycle scheme, spent fuel is re-processed and re-fabricated as a fresh fuel for recycling use in MoveluX, LWR or FR.

12. Development Milestones

2015	Start fundamental study based on the space reactor design
2017	Complete reactor type decision
2019	Start concept design
2028	Complete concept design and component demonstration
2030	Complete system demonstration without nuclear fuel
2035~	FOAK demonstration

ELENA (NRC “Kurchatov Institute”, Russian Federation)

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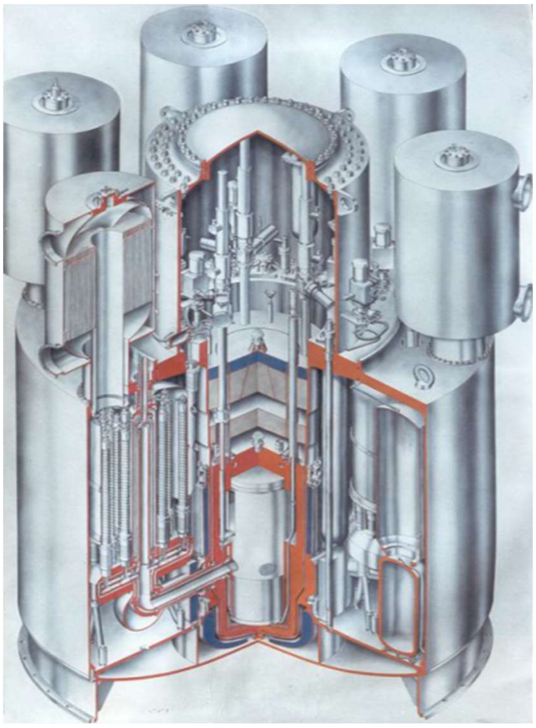


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	National Research Centre "Kurchatov Institute" (RRC KI), Russian Federation
Reactor type	PWR
Coolant/moderator	Light water / light water
Thermal/electrical capacity, MW(t)/MW(e)	3.3 / 0.068
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	19.6 / 0.36
Core Inlet/Outlet Coolant Temperature (°C)	311 / 328
Fuel type/assembly array	UO_2 pellet; MOX is an option
Number of fuel assemblies in the core	109
Fuel enrichment (%)	15.2
Core Discharge Burnup (GWd/ton)	57 600 / 27 390
Refuelling Cycle (months)	300
Reactivity control mechanism	Control rods and absorber rods
Approach to safety systems	Passive
RPV height/diameter (m)	3.7 / 1.25
Seismic Design (SSE)	VIII (MSK-64)
Fuel cycle requirements / Approach	Initial factory load for the entire lifetime
Distinguishing features	25 years without refuelling, passive reactivity control and unattended operation
Design status	Conceptual design

1. Introduction

The ELENA nuclear thermoelectric plant (NTEP) is a direct conversion water-cooled reactor without on-site refuelling capable of supplying 68 kW(e) of electricity and 3.3 MW(t) of heating capacity for 25 years without refuelling. The technology and techniques were developed incorporating experience from the construction and operation of the GAMMA reactor for marine and space application. The ELENA NTEP is designed as an "unattended" nuclear power plant (NPP), requiring nearly no operating or maintenance personnel over the lifetime of the unit. The conceptual design was developed by the National Research Centre “Kurchatov Institute” (NRC KI). The ELENA NTEP is a land-based plant; however, in principle it is also possible to develop underground or underwater versions. The reactor and its main systems are assembled from factory-fabricated finished components or modules, whose weight and dimensions enable any transport delivery method for the complete plant, including helicopter and ship. The specific features of the design include capability of power operation without personnel involvement, compensation of burn-up reactivity swing and other external reactivity disturbances without moving the control rods and the use of thermoelectric energy conversion to produce electricity.

2. Target Application

The unattended ELENA NTEP plant is designed to produce heat for towns with a population of 1500–2000 and located in remote areas where district heating is required. Since it is auxiliary in nature, the electricity generation of 68 kW could be used for the in-house power needs of the plant and to supply electricity to consumers requiring a highly reliable power supply, such as hospitals, etc. A desalination unit can be used in combination with the ELENA NTEP.

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

(b) Reactor Core

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

(c) Reactivity Control

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

(d) Reactor Pressure Vessel and Internals

The cylindrical core with a height of 850 mm and an equivalent diameter of 833 mm is installed in a steel shell with a diameter of 920 mm and encircled by an iron-water shield. The strengthened stainless steel reactor vessel has an internal diameter of 1250 mm and a height of 3700 mm with a wall thickness of 132 mm.

(e) Reactor Coolant System

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

(f) Pressurizer

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

- The primary circuit (circuit I) with natural circulation of the coolant (water with a pressure of 19.6 MPa) transports heat from the core to the thermoelectric generator (TEG) modules cooled by the circuit II coolant (water with a pressure of 0.36 MPa);
- Circuit II (intermediate circuit) removes heat from the cold joints of the thermal elements and transfers it through natural circulation to the intermediate heat exchanger of circuits II–III; the coolant is specially treated water, which also acts as part of the steel-water radiation shield;
- Circuit III is designed as a thermo-siphon with water or low-boiling coolant. Circuit III transfers heat through natural circulation to the heat exchanger of the heat supply circuit, the coolant being ethanol;
- Circuit IV transfers heat from the heat exchanger of circuits III–IV to the consumers using forced circulation; the circuit IV coolant is A-60 antifreeze.

5. Safety Features

The reactor is installed in a caisson forming a heat-insulating gas cavity in the strengthened area of the reactor vessel and a caisson space above the reactor cover to house control and protection system (CPS) drives and to prevent radioactive substances from escaping into the surrounding space in case of a circuit I break. The localizing safety systems provide defence in depth and secure the plant safety based on inherent safety features and predominantly passive phenomena; they require no human intervention or external power sources. The safety barriers of the ELENA-NTEP are: (1) Fuel elements; (2) Leak-tight primary circuit; (3) Caisson; (4) Reactor vessel and the guard vessel designed to withstand the pressure arising within each of them at their consecutive failure; and (5) An embedded silo sealed with a protective plate. Special measures for the protection of hot water consumers ensure that radioactivity is never released into the network circuit.

(a) Engineered Safety System Approach and Configuration

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

(b) Decay Heat Removal System

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

(c) Emergency Core Cooling System

The control safety system (CSS) consists of a control safety system for emergency shutdown and a system to input process and transmit safety-related plant information. During normal operation the emergency shutdown CSS is permanently awaiting a scram actuation request; it also periodically provides information on the state of the plant.

(d) Containment System

The reactor is installed in a caisson forming a heat-insulating gas cavity in the strengthened area of the reactor vessel and a caisson space above the reactor cover to house control and protection system (CPS) drives, and to prevent radioactive substances from escaping into the surrounding space in case of a circuit I break. In turn, the caisson is encircled by the external containment, which is the next barrier to the spread of radioactivity; water that fills the containment volume is circuit II coolant and acts as a biological shielding for the reactor. The external containment forms the cylindrical geometry of the plant with a height of 13 m and a diameter of 4.45 m.

6. Plant Safety and Operational Performances

The ELENA reactor does not require an operator during nominal power operation of the plant. Operators are required for assembly, startup and beginning of nominal operation. The reactor is designed to operate in a base load mode. The reactor installation is based on passive principles of heat removal (natural convection in all circuits, except for heat transport to the consumers) in normal operation and in shutdown conditions. A decrease in heat or consumer power is automatically compensated through the discharge of excess heat to the atmosphere via a dry cooling tower, with no changes in the electric power. There are no valves or mechanical parts which require maintenance over the lifetime of the plant.

Once operational the ELENA reactor depends upon natural processes to maintain the reactor power without the actuation of control rods. The control and safety systems, including the control rods, control rod drive mechanisms and sensors are used only for the reactor startup, or for the times that the reactor is scrammed. Startup is done by an on-site operator who can leave the site once steady-state power has been obtained. The reactor startup is done by measuring the neutron flux and calculating the reactor period. The reactor outlet temperature and pressure in the coolant loop is monitored, but do not provide feedback through the control loop during start-up. To begin the operation, the poison rods are pulled completely from the core, and are never inserted during nominal operation. To start up and reliably shut down the reactor in any situation, a grid is included that compensates the excessive reactivity. The compensation grid consists of six groups of the boron carbide absorber rods in stainless steel claddings of 1.45 cm external diameter. Each group (34 rods) has an individual drive.

7. Instrumentation and Control Systems

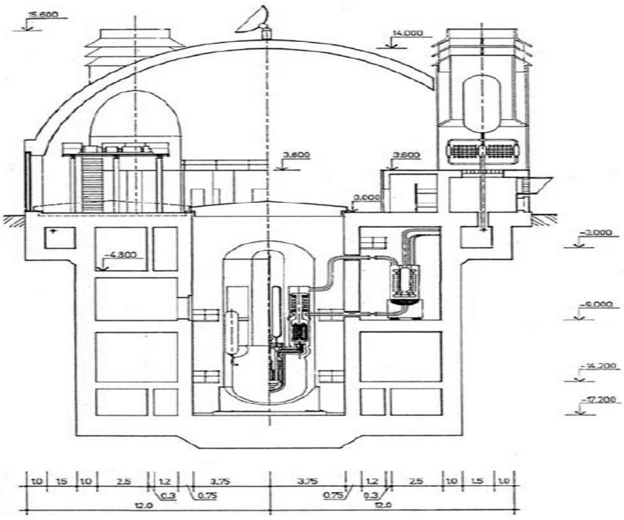
The instrumentation and control (I&C) system of the ELENA reactor is specially arranged to register parameter deviations at early stages of the accidental conditions to predict their further progression.

8. Plant Layout Arrangement

The reactor system can be broken into two parts for shipment. It is possible to fuel the system on-site, thereby eliminating problems associated with shipping a fuelled reactor. The plant includes instrumentation and control systems; a system for heat removal to consumers; an auxiliary power supply system; and a radiation monitoring system, including process radiation monitoring, dosimetric monitoring, and environmental monitoring.

(a) Reactor Building

The plant building has a cylindrical shape and is embedded in the ground for the entire reactor installation height with a foundation plate elevation of –19.2 m. The elevation of +0.0 has a domed ceiling. The underground portion of the structure, the walls and the overlaps are monolithic reinforced concrete. The vessel head of the system is removable. The plant incorporates a physical protection system, has a fence and is equipped with external lighting.



(b) Control Building

The plant has a main control and monitoring room accommodating the start-up and instrumentation and control equipment, as well as the equipment necessary to prepare information to be transmitted to a monitoring centre.

(c) Balance of Plant

i. Turbine Generator Building

A TEG is used as a heat exchanger between circuits I and II; it is based on semiconductor thermo-elements enabling the generation of 68 kW of power in the reactor nominal operating mode simultaneously with heat transfer to circuit II. This power is used for plant auxiliary needs; it could also be supplied to a small town without district power supply, partially replacing a diesel power plant. The TEG consists of eight identical thermoelectric units (TEU). Each of them includes 36 thermoelectric modules equipped with thermoelectric packs of bismuth tellurides with electronic and hole conduction.

ii. Electric Power Systems

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

9. Design and Licensing Status

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

10. Fuel Cycle Approach

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

11. Waste Management and Disposal Plan

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

12. Development Milestones

Not determined.

UNITHERM (NIKIET, Russian Federation)

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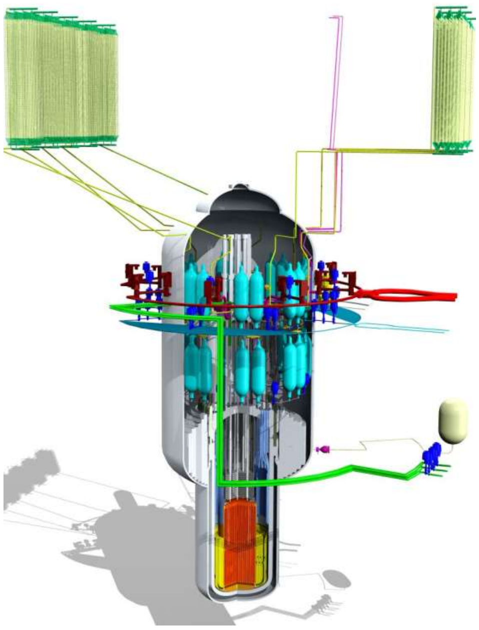
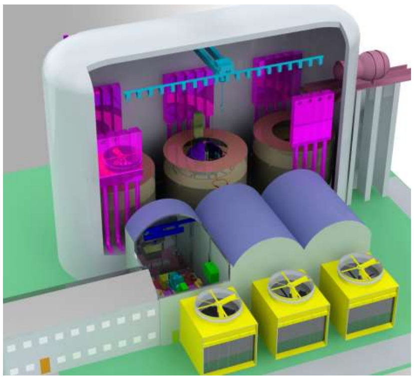




MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	NIKIET, Russian Federation
Reactor type	PWR
Coolant/moderator	High purity water
Thermal/electrical capacity, MW(t)/MW(e)	30 / 6.6
Primary circulation	Natural circulation
NSSS Operating Pressure (primary/secondary), MPa	16.5
Core Inlet/Outlet Coolant Temperature (°C)	249 / 330
Fuel type/assembly array	UO_2 particles in a metallic silumin or zirconium matrix, metal-ceramic/ 54-55
Number of fuel assemblies in the core	265
Fuel enrichment (%)	19.75
Core Discharge Burnup (GWd/ton)	1.15
Refuelling Cycle (months)	200
Reactivity control mechanism	Soluble boron and control rod insertion
Approach to safety systems	Hybrid (passive + active) system
Design life (years)	30
Plant footprint (m^2)	~10 000
RPV height/diameter (m)	9.8 / 2.9
RPV weight (metric ton)	32
Seismic Design (SSE)	VIII-IX-MSK 64
Fuel cycle requirements / Approach	Traditional
Distinguishing features	Autonomous passive reactor decay removal system; guard vessel; iron-water biological shielding; and the biological shielding tanks
Design status	Conceptual design

1. Introduction

The UNITHERM is a small transportable nuclear power plant (NPP) with a capacity of 30 MW(t) and a rated electrical output of 6.6 MW developed based upon NIKIET's experience in designing marine nuclear installations. The UNITHERM reactor is intended for electricity supply to urban areas and industrial enterprises in remote regions. UNITHERM adopts a natural circulated primary cooling system and is intended for minimal operational staffing with an option for unattended operation and a centralized regional support facilities monitoring. The UNITHERM design adopts proven technology and operational experience of the WWER type reactors. The design aims for fabrication, assembly and commissioning of the NPP modules to be carried out at factory. The UNITHERM reactor is designed to operate for 20-25 years without refuelling as both a land-based and barge mounted NPP. NPP with UNITHERM may consist of a number of units depending on the purpose and demand of costumers need.

2. Target Application

The UNITHERM NPP can be used as a source of energy for the generation of electricity, district heating, seawater desalination and process steam production. In general, the configuration and design of the UNITHERM is sufficiently flexible to be adjusted or modified for different target functions and user requirements, without compromising the underlying principles of the design.

3. Design Philosophy

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

4. Main Design Features

(a) Nuclear Steam Supply System

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

The intermediate circuit system is intended for heat transfer from the intermediate circuit coolant to the secondary coolant (consumer's circuit) inside a steam generator (SG). This system provides an additional localizing safety barrier to protect the heat consumers against the ionizing radiation from radionuclides generated by primary coolant activation, from structural material corrosion products dissolved in the primary coolant as well as fission products entering the primary circuit in case of fuel cladding failure. Primary coolant circulates by means of natural convection.

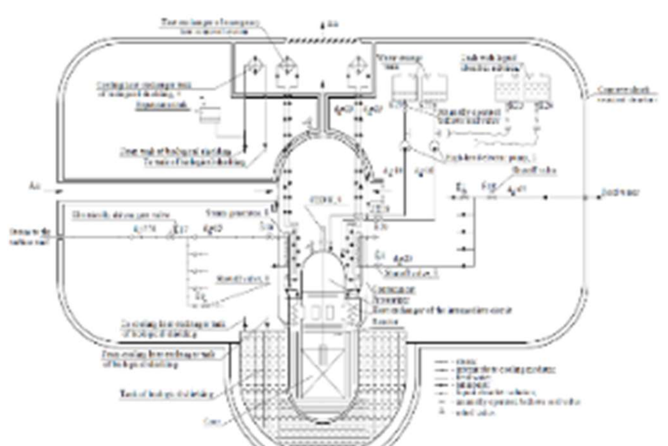
The secondary system (consumer's circuit) is intended to generate a superheated steam from the feedwater (supplied by NPP turbo generator pumps) by means of heat transfer from the intermediate circuit coolant inside the steam generator. Secondary coolant circulates by means of natural convection.

(b) Reactor Core

The reactor core consists of 265 fuel assemblies installed in the plates of the removable reactor screen at the points of a regular hexagonal lattice. The UNITHERM fuel element is designed as a cylindrical rod with four spacing ribs on its outer surface. The fuel is in the form of tiny blocks of UO_2 grains coated with zirconium and dispersed in a zirconium matrix. The gap between the fuel-containing matrix and the cladding is filled with silumin. A fuel element of such design has a high uranium content and radiation resistance. These features, taken together, make it possible to operate such fuel elements during the whole specified core lifetime. A specific feature of the UNITHERM fuel cycle is the long and uninterrupted irradiation of fuel inside the reactor core throughout the reactor lifetime, without core refuelling. The metal ceramic (CERMET) fuel chosen for the UNITHERM is composed of UO_2 particles in a metallic (silumin or zirconium) matrix. This design is characterized by a high volume ratio of nuclear fuel; the use of the metallic matrix ensures minimum swelling and high thermal conductivity. Optimally shaped cladding is formed when the cladding is filled with the matrix composition.

(c) Reactivity Control

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



(d) Reactor Pressure Vessel and Internals

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

(e) Reactor Coolant System

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

(f) Steam Generator

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

(g) Pressurizer

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

5. Safety Features

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

(a) Engineered Safety System Approach and Configuration

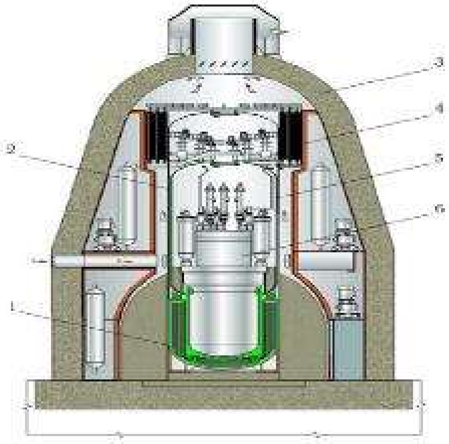
The UNITHERM safety systems are based upon redundancy, diversity and the maximum use of the fail-safe systems. The UNITHERM employs passive safety systems and devices which do not require actuation (such as containment, independent heat removal system, etc.) or can be passively actuated (such as primary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

(b) Emergency Core Cooling System

An independent passive heat removal system is adopted which acts as a cooldown system in emergency shutdown of the reactor. During a postulated loss of coolant accident (LOCA) scenario, some primary coolant and steam-gas mixture from the pressurizer are discharged to the containment. The emergency core protection system is activated in response to the signals from pressure transducers. Coolant leakage continues until the pressure values in the reactor and containment are equalized. The remaining coolant inventory in the reactor is sufficient to maintain circulation in the primary coolant circuit. The reactor is passively cooled via the intermediate circuit and the independent heat removal circuit, whereas the containment heat is removed by the component cooling system. Additionally, an active user circuit with feedwater supplied to the SG and steam-water mixture maybe utilized to increase the cooling rate. The iron-water biological shielding acts as a system of bubble tanks for cooling water storage. The shielding removes heat from the RPV, preventing a core melt in a postulated beyond design basis accident with reactor core voiding.

(c) Containment System

The integral reactor for land-based deployment is placed inside the leak-tight containment, which is located within the concrete shock-resistant structure together with the biological shielding and reactor unit components. This structure enhances physical protection of the reactor unit from external impacts such as airplane crash, hurricane, tsunami, unauthorized access, etc.



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

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

6. Plant Safety and Operational Performances

Electrical output of the NPP with UNITERM-30 reactor equals to N(e) – 6.6 MW(e). Electrical voltage provided to the user grid – alternate 3-phase 10.5 кV ± 10 %, frequency 50 ± 1 Hz. Basic regime of NPP operation lies within the power range from 20 to 100 % N(e) providing a daily and annual load following. The speed of power augment and drop – 0.1 % от N (e)/sec. Upon a customer request there may be foreseen additional provision of thermal power. Maximum rate of it could be up to 28 Gcal/hour.

7. Instrumentation and Control Systems

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

- Safe operation of the NPP and electrical generation; protection from the violations of safe operation limits and conditions; prevention of accidents; mitigation of accident consequences; bringing the NPP back to the controlled and safe condition during accidents and after them.
- The Automatic Control System consists of functionally completed systems developed on the basis of programmatic-technical systems and instruments that were trialed in the NPP conditions or other analogous objects.

Technical appliances for the ACS are manufactured at the enterprises according to approbated technology and methods of testing and control while strictly observing the requirements of quality control.

8. Plant Layout Arrangement

(a) Reactor Building

The NPP site is limited by perimeter of the protected zone that does not exceed a square of 2 hectares.
The site hosts reactor building for housing reactor(s) which possesses special transport locks for delivery of the reactor plant for mounting and other equipment necessary during outages and removal of the reactor facility; building to house turbine-generator(s) and some other auxiliary buildings. Turbine-generator assembly for UNITHERM NPP depends on the plant capacity and operation mode requested by its users. The turbine operates using dry saturated steam in the mode of steam outlet backpressure. With consideration of the continuous transfer of 5 % heat to the independent heat removal system, the total efficiency in this case is expected to be ~74 %. High efficiency is achieved from the utilization of low-parameter heat at the turbine exhaust. An electric generator with an output of 6.6 MW(e) in combination with a single-phase intermediate circuit allows to obtain a superheated steam temperature of 285°C under 1.35 MPa.

9. Design and Licensing Status

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

10. Fuel Cycle Approach

The duration of the campaign reactor core is 15 years.

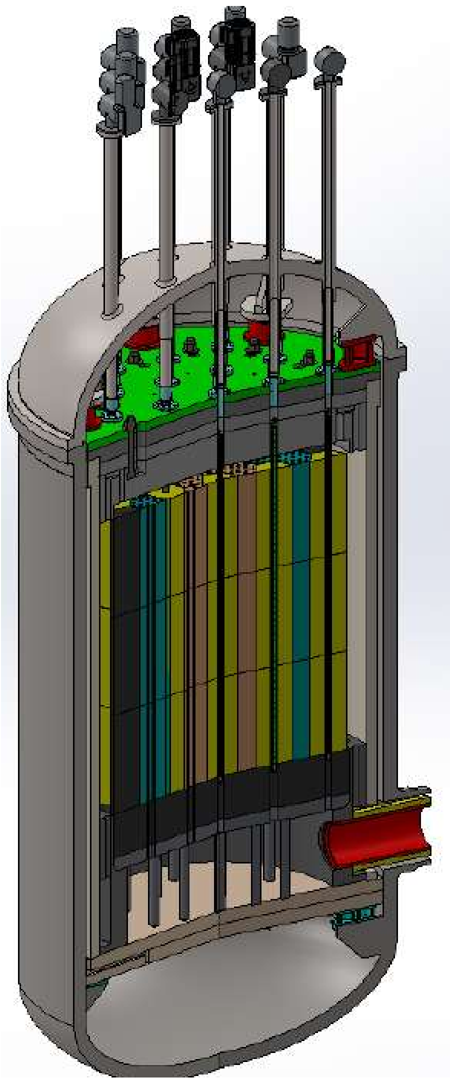
11. Waste Management and Disposal Plan

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

12. Development Milestones

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

Advanced Micro Reactor – AMR (STL Nuclear (Pty) Ltd, South Africa)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	STL Nuclear (Pty) Ltd., South Africa
Reactor type	HTR, Advanced Prismatic Type
Coolant/moderator	Helium, graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 / 3 for single module plant
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	4 / 1
Core inlet/outlet coolant temperature (°C)	450 / 750
Fuel type/assembly array	TRISO particles/ LBE eutectic/ SiC tubes
Number of fuel assemblies in the core	420
Fuel enrichment (%)	10 to 20
Refuelling cycle (months)	96
Core Discharge Burnup (GWd/ton)	80-90
Reactivity control mechanism	Control and shutdown rods in core, and the reflector
Approach to safety systems	Passive / inherent decay heat removal
Design life (years)	40
Plant footprint (m^2)	2 500
RPV height/diameter (m)	5.96 / 2.78
RPV weight (metric ton)	115
Seismic design (SSE)	0.3g (generic site) 0.5g under consideration
Fuel cycle requirements/approach	LEU UO_2
Distinguishing features	Additional SiC barrier; Lead Bismuth Eutectic (LBE) heat transfer medium
Design status	Pre-conceptual design

1. Introduction

STL Nuclear (Pty) Ltd., the University of Pretoria, the North-West University in conjunction with the South African Nuclear Energy Corporation (NECSA) are developing a 10 MW(t) Small Modular Reactor (SMR) called the Advanced Micro Reactor (AMR). This reactor falls in the category of the High-Temperature, Gas Cooled Reactors (HTGRs). The AMR uses helium as the coolant and is graphite moderated. It uses graphite hexagonal blocks as the moderator and these blocks are arranged to form a cylindrical configuration. The individual SiC fuel assemblies contain TRISO coated particles with either uranium dioxide (UO_2) or uranium oxycarbide (UCO) ceramic fuel kernels of between 10 wt% to 19.9 wt% enriched uranium. The voids between the coated particles are filled with a Lead Bismuth Eutectic (LBE) alloy to provide good heat transfer from the fuel particles to the fuel assembly wall.

2. Target Application

The AMR can supply electric power to any distribution grid and to standalone or isolated electricity users. It can be deployed as single modules or multi-module plants as well as for process heat applications.

3. Design Philosophy

AMR design utilizes proven HTR technologies albeit in a different configuration The reactor is designed to have excess reactivity to operate for several years before refuelling is required. The design is factory assembled to enable road transportation.

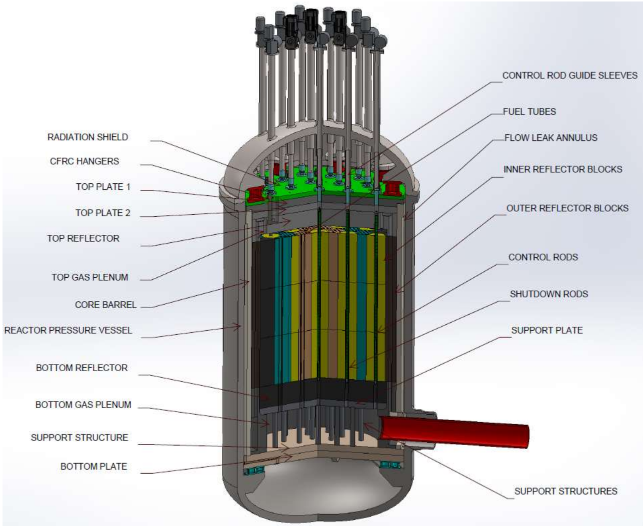
4. Main Design Features

(a) Power Conversion

A Heat Pipe Heat Exchanger (HPHE) is used to transfer heat from the reactor core to the secondary power conversion loop containing air. The LBE alloy is used to absorb heat from the primary helium coolant bundle (located in the centre) and surrounded by a flow-directing shroud. This heat is then exchanged by means of natural convection with the secondary power loop (in the form of a multiplicity of U-tubes) located in the annulus between the flow-directing shroud and the inner wall of the outer shell of the heat exchanger. The LBE returns to the central hot helium bundle at the bottom through several portals in the flow-directing shroud. This flow of LBE in this design can be described as "toroidal natural convection flow".

(b) Reactor Core

The AMR is a high temperature helium-gas cooled reactor with a power of 10 MW(t). The helium coolant is circulated through the reactor core by an electric blower located within the pressure boundary of the heat exchanger. As is the custom for thermal neutron-spectrum high temperature reactors, graphite is used as moderator, which in this design, is in the form of hexagonal graphite blocks packed to form an approximate cylindrical configuration with a diameter and height (including radial reflectors as well as top, bottom and) respectively of ~2.2 m (~1.72 m active diameter), and 2.4 m (active length 2.2 m) with a volume of 5.103 m^3 which results in a core power density of ~2 MW/m^3. The fuel assemblies are silicon carbide tubes that contain the fuel in the form of low-enriched uranium (LEU) dioxide (UO2) or uranium oxycarbide (UCO) TRIstructural-ISOtropic (TRISO) coated particles, immersed in an LBE (45% Pb and 55% Bi) alloy.



Reactor Core

(c) Fuel Characteristics

In typical coated particle. The SiC is the main layer for the retention of fission products. The AMR is also not only limited to kernel and coated particle diameters of 425 microns (µm) and 855 µm respectively but can also utilize kernel diameters of 500 µm and coated particle diameters of 920 µm. The AMR is not restricted to using only high assay low enriched uranium (HALEU) it can utilize enrichments of ~10 wt% to 15 wt%. Approximately ~463 775 coated particles will be contained in a typical fuel assembly.

(d) Fuel Handling System

The AMR is designed to operate between 8 to 10 years without the need for refuelling. Burnable absorbers are used to supress the flux at the beginning of life and extend the life of the reactor as they deplete. The reactor at time of refueling, will be transported back under IAEA seals to the NECSA site where the fuel will be unloaded under continuous IAEA containment and surveillance measures. The spent fuel will be removed and stored during the initial post-operational cooling period on-site in shielded vessels under IAEA seals. After this initial cooldown period and depending on the spent fuel storage strategy in force at the time, it will be transported to its final storage location. A new core will be loaded in the reactor vessel using the same process as described, and the newly loaded vessel will be transported its operational site.

(e) Reactivity Control

As a defence-in-depth, two diverse shutdown systems, each capable of shutting the reactor down with fresh fuel when the reactor is cold and with one member (of highest reactivity worth) stuck in the full extracted position. The shutdown margin is 8%. Thirteen neutron absorber rods are provided in graphite sleeves; 7 within the core and 6 within the side graphite side reflector blocks. There are also 6 shutdown rods located within the core. The absorber rods can be operated independently as a group or as sub-groups, as required by the reactor operating control system. A control rod consists of several rod absorber material segments, pinned together to form articulating joints. The segments consist of sintered B_4C absorber material, sandwiched between an inner and an outer tube segment. The inner tube segment allows cooling helium gas to flow form the top down in the circular channels. If all control and shutdown systems are accidentally withdrawn, it will not lead to fuel damage. This is normally defined as a "start-up accident".

(f) Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel (RPV) is constructed in compliance with the ASME III subsection NB code. It comprises two main components reactor of vessel body and vessel head which is bolted to the vessel body. The reactor vessel body consists of several forged ring-components circumferentially welded together.

The core structures consist of the metallic parts and the graphite structures. The function of these internal structures is to provide stable core geometry, neutron reflection, cold and hot gas channelling, fuel element flow, shielding, thermal insulation and support of the control and shutdown systems guide tubes and the neutron source. The functional design of the structural core internals is such that they can withstand the steady state and transient loadings during normal operation, anticipated operational occurrences and design basis accidents. The loads borne by the ceramic internals are transferred to the stainless-steel core barrel and then to the reactor pressure vessel through metallic components such as the lower support structure and the core barrel axial and radial supports. All areas of the core internals are designed for the service life of the reactor. Access for ceramic structure inspections can be done through the fuel loading channel and the reflector rod holes.

(g) Reactor Coolant System

The helium coolant enters the reactor vessel at 450°C through the annulus of the co-axial duct. The helium then flows upwards in the helium risers located in the outer graphite reflector. Helium leak flow also enters the annular space between the Core Barrel (CB) and the inside of the Reactor Pressure Vessel (RPV). The helium flow enters the top of the reactor core where it is evenly directed to the 420 borings containing the fuel assemblies as well as between the annuli of the control rod guide-tubes. The helium leak flow also enters the top and bottom RPV domes. The helium from the bottom dome re-joins the leak annulus between the CB and RPV, while the helium entering the upper dome flows past the control rod guides and metallic components and connects to the upper gas plenum and is routed downwards to join the major coolant flow past the fuel assemblies. Some helium from the upper dome is also forced into coolant holes within the control rod guide sleeves which flows in the inner anulus of the control rods, this then exits the sleeve and re-joins the major downward flow in the core. The helium then flows downward through the annulus in the borings past the fuel assemblies to remove heat and exits the core at 750°C. It is then collected in a lower core hot gas plenum that is part of the lower core support structures and flows back through a hot duct (connected to the hot gas plenum) to the HPHE.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The plant is designed to perform its safety functions without reliance on the automated plant control system, or the operator. The plant has no engineered safety systems in terms of active human or machine intervention to assure nuclear safety.

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 / Reactor Cooling Philosophy

The Reactor Cavity Cooling System (RCCS) removes heat radiated from the reactor towards the reactor cavity walls. It consists of welded membrane tubes arranged side-by-side on the inside of in the reactor cavity wall. Water is circulated through the tubes to form a cold wall. The RCCS is a passive system and consists of independent cooling trains and is designed for all postulated design basis conditions.

(c) Spent Fuel Cooling Safety Approach / System

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

(d) Containment System

The reactor has two barriers of SiC, that being the SiC layer surrounding each fuel kernel as well as the SiC fuel tube of the fuel assembly. This introduces an additional barrier which will further reduce the probability of any radioactivity being released into the environment. In addition to the two abovementioned barriers, the pressure boundary serves as a third barrier against release of radioactivity, and then as a final barrier the reactor building can also be accredited as a final enclosure.

(e) Chemical Control

Chemical stability: The design of the core and its coolant routing is such that in an event that could allow air to leak into the pressure boundary, there is no possibility that a sustained corrosion of core components by air can take place. The reactor also does away with the possibility of a water or steam ingress scenario as the helium coolant will transfer heat to a HPHE which is a single-phase natural convection heat pipe heat exchanger using LBE as working fluid. This heat exchanger is then coupled to a Brayton power conversion cycle. The use of a HPHE also introduces another important safety feature by eliminating the possibility of tritium, produced in the primary helium cooling circuit to contaminate the air in the secondary circuit by diffusing through a single tube wall.

6. Plant Safety and Operational Performances

The thermal-hydraulic calculations show the maximum fuel temperatures for the calculations was 1101.5°C and the maximum central fuel temperatures determined using a validated computer code being 1084°C (a difference of only ~17.5°C) which is 1.6%. LBE is used as filler material in the fuel assembly due to the fact that it reduces the central fuel temperature by ~130°C compared to a fuel assembly only containing helium as the filler material. The fuel will remain below the normal operational temperature guideline chosen of 1130°C and in the case of a DLOFC event will remain below the Germany set temperature limit of 1600°C.

7. Instrumentation and Control System

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

8. Plant Layout Arrangement

The reactor building contains the safety equipment that provides the necessary functions for the safe shutdown of the reactor under all design basis conditions. The reactor building (RB) is partially submerged below ground level such that the reactor and heat exchanger cavities are completely protected against postulated external threats. The reactor building, electrical building and auxiliary buildings are connected by means of underground tunnels, providing protection for interlinked services. The RB is seismically designed to withstand a design basis earthquake (DBE) and is the only safety related building structure of the AMR.

9. Testing Conducted for Design Verification and Validation

Experimental activities are being conducted to determine heat transfer enhancement of the Lead Bismuth Eutectic (LBE) in SiC fuel tubes. It is anticipated that the core neutronics will be verified in a future critical facility at the NECSA site. The Helium Test Facility on the NECSA site will be used to test the operational performance of key components under non-nuclear operational conditions of temperature, flow and pressure.

10. Design and Licensing Status

The AMR is at pre-conceptual phase. The neutronics, thermo-hydraulics and heat transfer analyses are being conducted to verify the safety analysis. Mechanical design and material selection are also underway.

11. Fuel Cycle Approach

The AMR is designed for UO_2 and UCO. However, it is not limited to the use these types of fuel. The AMR while maintaining safety characteristics, can use alternate fuels without modifications. Advanced fuel cycles for later investigation may range from a (Th, U)O_2 using both LEU and HEU, to a UC_2 fuel cycle.

12. Waste Management and Disposal Plan

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

13. Development Milestones

2020	Project Started
2021	Pre-conceptual design
2022	To begin with conceptual design

LFR-TL-30
(*new*cleo Ltd, United Kingdom)

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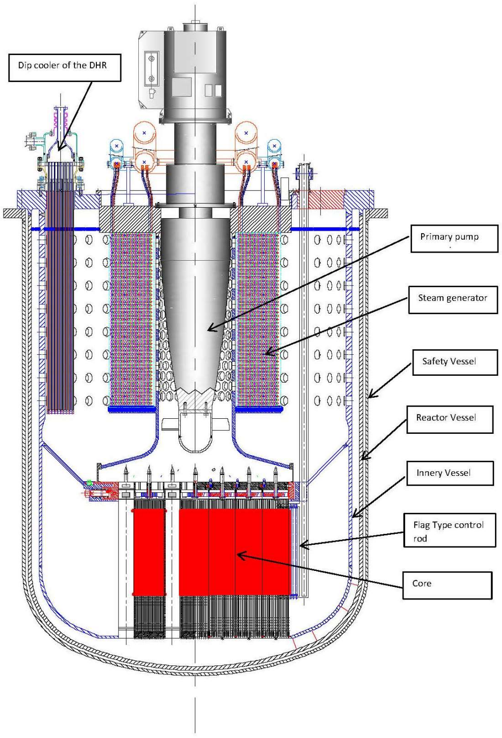


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	*new*cleo Ltd, United Kingdom Formerly developed by Hydromine S.àr.l, Luxembourg
Reactor type	Liquid metal cooled fast reactor
Coolant/moderator	Lead / none
Thermal/electrical capacity, MW(t)/MW(e)	90 / 30
Primary circulation	Forced circulation
NSSS operating pressure (primary/secondary), MPa	0.1 / 14 (absolute)
Core inlet/outlet coolant temperature (°C)	420 / 530
Fuel type/assembly array	UO_2 / hexagonal fuel assembly
Number of fuel assemblies in the core	37
Fuel enrichment (%)	13.0 % inner fuel, 19.75% outer fuel
Refuelling cycle (years)	One batch, each 15 years
Core discharge average burnup (GWd/ton)	49
Reactivity control mechanism	6 x 30° circular sectors made of absorber material
Approach to safety systems	Active + Passive
Design life (years)	30
Plant footprint (m^2)	N/A
RPV height/diameter (m)	5 / 3.7
RPV weight (metric ton)	~ 35
Seismic design (SSE)	0.5g
Fuel cycle requirements/approach	Wastes burned in another *new*cleo model
Distinguishing features	- Active + passive safety; - No intermediate loops; Simple, - compact primary system and compact reactor building.
Design status	Conceptual design

1. Introduction

The LFR-TL-30 is an innovative micro SMR (Small Modular Reactor), cooled by molten lead, conceived to comply with the GEN IV reactor criteria of safety, sustainability, proliferation resistance, and economics despite the unfavourable small-scale effect. LFR stands for lead-cooled fast reactor, TL stands for Transportable Long-lived and 30 is the net electricity in MW. The LFR-TL-30 is an integral pool-type fast reactor with all the primary components installed inside the reactor vessel (RV). Main primary components are the innovative core, the innovative Spiral-Tube Steam Generator (STSG), the mechanical pump (MP), reversed-flag type control rods, decay heat removal by dip coolers and a safety vessel air cooling system. The reactor design is derived from the LFR-AS-200 with the objective to verify up to which extent it is possible to apply the innovations embodied in the larger LFR-AS-200 to design a micro reactor with similar level of compactness. Several options regarding the fuel type, the power level and the thermal cycle, which are different in relation to LFR-AS-200, have been investigated. An important cost parameter to be considered when designing a micro reactor is the plant cost per unit power ($/W). It is possible, namely, to reduce the reactor size while reducing power, but the plant $/W ratio is likely to become prohibitively high, owing to the cost of the fuel handling machines and the buildings and facilities for storage of the fresh and spent fuel assemblies, which is relatively independent from the reactor power and hence increases the $/W ratio. It should be considered, too, that it is not wise, for risks of proliferation, to provide the predictably numerous micro reactor plants with these fuel storage facilities. A measure to overcome this proliferation and cost issue is to design

micro reactors capable to be transported, complete of the core, to centralised facilities for fuel handling and maintenance of main components. For this design approach to become viable, the micro reactor shall be provided with a long-life core and be capable of transport in upright position, in order not to affect its mechanical and thermal-hydraulic configuration while traveling. The LFR-TL-30 can be designed to comply with both features of long- life core and transportability in upright position. Long-life cores are possible owing to the high breeding capability of the fast reactor; transportability, that is bound to the compact reactor assembly, in particular to the short outline height, is a merit of the very compact pump-spiral-tube steam-generator (pump-STSG) assembly previously conceived for the LFR-AS-200. No on-site refuelling being requested, it is possible to install only one pump-STSG assembly centreline above the core. This is the main characteristic of the LFR-TL-30 reactor, which allows to balance the unfavourable scale effect with respect to the LFR-AS-200 and to approach the unprecedented title of merit of a specific volume of the primary system of about 1 m^3 /MW(e) which is an outstanding feature of the LFR-AS-200.
LFR-TL-30 (and LFR-AS-200) will profit of the development of the LFR-AS-30 research reactor, designed specifically by *new*cleo to support the design and qualification processes of the innovative solutions proposed.

2. Target Application

Target applications include: Sites without interconnected grids; Offshore oil platforms; Mines; Islands; Naval propulsion (after appropriate modifications).
These sites require an "inherent safety" with the associated no need of the "emergency zone" and a particular simplicity of operation for staffing reduction. The deployment of a micro reactor is facilitated by the unique safety features of the LFR, particularly for applications in oil platforms and/or naval propulsion: in exceptional accidents, like ship collision and sinking, frozen lead will always maintain the confinement of the core without water pollution and preserving potential for reactor recovery

3. Design Philosophy

The LFR-TL-30 is an integral pool-type fast reactor with all the primary components installed inside the reactor vessel (RV). Main primary components are, the innovative STSG, the mechanical pump (MP), flag type control rods, dip coolers of the decay heat removal (DHR) system. Thanks to the properties of lead, intermediate loops can be eliminated with several special precautions to deterministically eliminate any risk of important primary system pressurization: among them water and steam collectors located outside of the RV, and a short STSG. Reactor shutdown and DHR is performed by means of diversified and redundant systems which are passively operated, and which can also be passively actuated when the primary system exceeds certain pre-set threshold temperatures.

4. Main Design Features

(a) Power Conversion

The heat is directly transferred from the primary lead to the water/steam system by means of a single STSG. Superheated steam is produced and used in a turbine (or other steam consumers if needed).

(b) Reactor Core

The core consists of 37 wrapped, hexagonal FAs, each containing 469 fuel pins laid out on a triangular pitch.
The core is removable for replacement by a fresh core in an off-site facility. The heads of the FAs are interconnected by means of cams, which are integral part of each head. The result is a self-sustaining core anchored to an internal structure (the inner vessel). The FA foot is free from mechanical supports (no core grid of classical design such as the Diagrid of SPX1) except for the radial contact with adjacent FAs, which create a packed bundle, contoured by the bottom end of the inner vessel.

(c) Fuel Characteristics

Fuel considered is HALEU in oxide form, although alternative fuels, like metal, nitride or carbide fuel, can also be considered. To comply with non-proliferation norms, the highest enrichment is kept below 20% (19.75%). The consequence is that for very small reactors like LFR-TL-30, the volume of the core is mainly dictated by the need of a sufficient mass of fuel to reach criticality and ensure a reactivity margin to compensate for the reactivity swing during burn up. The mass of required uranium is in the range of 8.6-9.6 t.

(d) Reactivity Control

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

(e) Reactor Pressure Vessel and Internals

The RV is shaped as a cylindrical vessel with toro-spherical bottom head and flat roof. The free surface of lead is kept sufficiently below the roof to allow for a gentle thermal gradient between the vessel in contact with lead and the colder roof. The plenum above the free level is filled by argon as cover gas. The roof is made of a circular thick plate with small-diameter penetrations for the dip coolers of the DHR and of the control and shutdown devices, and a large-diameter central penetration for the Spiral-Tube Steam Generator with the

integrated Primary Pump. The inner vessel is shaped as a double bottom shell and comprises a substantially cylindrical lateral wall supported by the roof and upper bottom lower bottoms extending radially inwards from the lateral wall. The lower bottom has a central opening delimited by a peripheral edge for radial constraint of the Fuel Assemblies (FAs) of the core. The upper bottom internally comprises a plurality of jaws for vertical support and elastic radial constraint of the FAs. The FAs are fastened together on the heads by latches (bars pivoting into catch-slots) provided in their heads and are provided with a plurality of thermal expanders. At normal operating condition, the front plates of all thermal expander are in touch with each other, but idle. On trespassing a certain temperature limit chosen at the design phase, FAs are pushed radially outwards to reduce core reactivity and bringing core power into balance with the heat dissipation capacity of DHR systems.
All internals are hung to the reactor roof and have no connection with the reactor vessel. The MP, STSG, inner vessel and core are co-axial with the reactor vessel.

(f) Reactor Coolant System

The vertical axial-flow MP is installed on the RV centreline in the available space inside the tube bundle of the STSG. The pump rests on and is connected to the upper support plate of the SG by means of a flange which closes the pump's shaft penetration through the reactor roof and supports the variable-speed electric motor of the pump. The pump is characterized by a short, large-diameter, tapered hollow shaft containing lead brought in rotation by the shaft itself, in order to increase the mechanical inertia of the pump. There are no in-lead pump bearings. Primary lead circulates inside the core from the bottom to the top, then is conveyed by a funnel-shaped structure to feed upward the MP and then the STSG, which is thereby fed from the bottom. Hot lead flows radially through the perforated inner shell of the Steam Generator and, once past the tube spirals, continues flowing radially through the perforated upper portion of the inner vessel, and then downwards between the inner wall of the RV and the outer wall of the lower part of the inner vessel to feed again the core. The RV is always in contact with cold lead below the creep temperature of the steels.

(g) Secondary System

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

(h) Steam Generator

The STSG is an innovative SG conceived for compactness and because it offers several advantages in terms of reactor cost, safety, reactor operability and simplicity of the lead flow path. The SG tube bundle is composed of a stack of spiral-wound tubes, arranged one above the other and equally spaced. The inlet and outlet ends of each tube are connected to the feedwater header and steam header, respectively, both arranged above the reactor roof to eliminate, in case of their failure, the risk of large water/steam release inside the reactor vessel. The SG is thermally almost equivalent to a pure counter-current SG, because the feedwater in the tubes circulates from the outer spiral to the inner spiral, while the primary coolant flows radially in opposite direction from the inner shell to the outer shell. Because the flow path of the primary coolant inside the bundle is short, its speed can be increased while keeping the pressure loss limited.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

In a reactor cooled by lead there is a large margin between the operating temperature and the safety limit and the LFR-TL-30 exploits this margin for actuation of passive shutdown and passive decay heat removal systems, which do not need power sources, operator intervention and logics. The shutdown system is realised by two diversified systems. The first is an ex-core system envisaged to promptly shutdown the reactor in case of emergency. It is formed by an absorber pin bundle kept together by two main grids located at both its ends; the top grid is connected to the holding shaft, with the main function to keep the bundle in withdrawn position and to insert it in case of SCRAM signal. The system indeed, operates in a traditional way, being driven from gravity by means of a ballast. The absorber portion is also made of highly enriched B_4C. The second redundant safety shutdown system, unpublished, is passively actuated and passively operated. In case of unprotected transient with local clad failures, the steam-cladding accidental reaction and the resulting generation of hydrogen and associated explosions, encountered with other liquid metal concepts, are excluded with lead coolant. Finally, lead has also good retention capability of volatile fission products and in extreme conditions the reactor can be directly cooled by jets of water, with the further advantage that frozen lead builds up its own sarcophagus (barrier made of frozen lead) and definitively stop radionuclide dispersion.
The ultimate goal is the elimination of the need of an emergency preparedness zone.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

DHR is performed by means of two diverse (DHR1 and DHR2), redundant systems, each consisting of two identical loops. One loop is adequate to remove the decay heat. The DHR1 system (not yet disclosed) removes heat through the cold collector of the primary system. Each loop of the DHR2 system is equipped with spirals of square-cross-sectional tubes wrapped around the reactor safety vessel for transfer to a water-steam system the heat transmitted by radiation from the reactor vessel to the safety vessel. The steam is passively condensed in an air cooler which can be actively actuated or even passively actuated above a certain temperature threshold.

(c) Spent Fuel Cooling Safety Approach / System

The plant is not provided with refuelling capabilities (flasks, pools). The reactor must be transported in upright position in a refuelling factory where these facilities are foreseen. Spent fuel pool are supposed to work in "dry" conditions.

(d) Containment System

The reactor is provided with a concrete containment external-missile-proof. The dimension of the containment is kept small by the very low potential energy stored in the coolant (which operates at atmospheric pressure) and the small inventory of water/steam of the secondary circuit. A safety vessel eliminates any loss of coolant accident (LOCA) even in the event of a failure of the reactor vessel.

(e) Chemical Control

An oxygen control system is implemented to avoid the formation of lead oxides and at the same time the preserve the passivation of steel structures and components. Several systems, already successfully tested on experimental loops, are under investigation for implementation in the research reactor LFR-AS-30 and then in the industrial reactor LFR-TL-30.

6. Plant Safety and Operational Performances

Taking into account the envisaged uses of this type of reactors, a major aim of the design is to avoid the need of safety-grade diesel-generators and, for example, to use multiple smaller units on the same site, in order to improve the electrical network availability, preferring the use of UPS (batteries) with an important capacity. The expected refuelling cycle is about 10 to 15 years, without the need of important maintenance outages in between. The system is conceived to operate mainly in reactor priority mode, but the potential for operation in load follow mode is under investigation.

7. Instrumentation and Control System

Core instrumentation (mainly thermocouples) will be used by the automated control and surveillance system and by the protection one, with due attention to diversification at sensor and platform levels. Remote surveillance and shutdown capabilities will be implemented.

8. Plant Layout Arrangement

The key for economics of the LFR-TL-30 is based on reactor compactness and associated compactness of the reactor building. The compactness of the reactor, the absence of intermediate loop, of the fuel handling facilities and spent fuel storage allows to drastically reduce the size of the reactor island civil structures (be it on land or on a ship).

9. Testing Conducted for Design Verification and Validation

The LFR-TL-30 is in the conceptual design stage. Most of design verification and validation will be made on LFR-AS-30.

10. Design and Licensing Status

Licensing of the LFR-TL-30 has not started, yet. Licensing will take advantage of the return of experience of the design and operation of the LFR-AS-30 under fast development.

11. Fuel Cycle Approach

In a scenario of revived interest in nuclear energy, the LFR-TL-30 fuel cycle would naturally fit in an assumed large nuclear energy use programme of reactors of various sizes and for various uses. In this scenario the spent fuel could be reprocessed for the recovery of the important residual fraction of ^{235}U to be reused in transportable reactors, and of plutonium and possibly also of minor actinides as a fuel of central-station fast reactors.

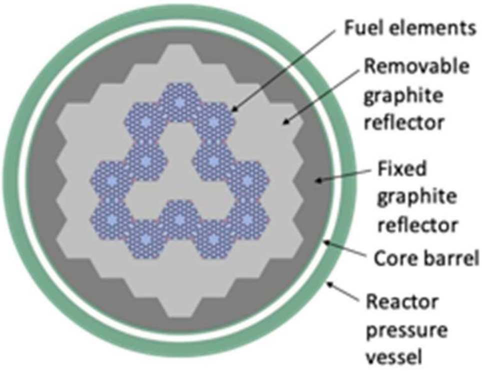
12. Waste Management and Disposal Plan

After fuel reprocessing and Pu recovery (to be burnt for instance in the LFR-AS-200 always proposed by newcleo for this purpose), wastes are meant to be disposed. At the time of refuelling, the components of the primary system can be reused after inspection and requalification. Lead will be recycled in other reactors.

13. Development Milestones

2017	Completion of pre-conceptual desing (by Hydromine Nuclear Energy S.àR.L, Luxembourg)
2019	Conceptual design started
2021	Hydromine Nuclear Energy S.à r.l. is incorporated in *new*cleo Ltd for a fast development program. Design restarted after an idle phase.

U-Battery (Urenco, United Kingdom)



MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Urenco, United Kingdom
Reactor type	High-temperature gas-cooled micro nuclear reactor
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	10 / 4
Primary circulation	Forced circulation helium
Secondary cycle	Nitrogen secondary circuit (no water)
Fuel type/assembly array	TRISO / Hexagonal
Number of fuel assemblies in the core	12 x 5
Fuel enrichment (%)	< 20%
Core Discharge Burnup (GWd/tonne)	~80 (average)
Reactivity control mechanism	Control rods, fixed burnable poisons, secondary shut-down absorber spheres
Design life (years)	5 EFPY core life, 30 year design life
RPV height/diameter (m)	Approx. 5.5 / 2.2
Distinguishing features	Simplicity, established technology basis, demonstrated fuel. Multiple modules can be installed at a single site.
Design status	Conceptual design

1. Introduction

U-Battery is an advanced/small modular High Temperature Reactor (HTR), capable of providing a low-carbon, cost-effective, locally embedded and reliable source of power and heat for energy-intensive industries and remote locations. The U-Battery concept is the result of a challenge initially set by Urenco to the University of Manchester and the Technical University of Delft to use existing, developed technology to bring nuclear to a market that, to date, has been serviced by diesel or other smaller sized fossil fuel or renewable technologies. The universities were asked to consider all nuclear technologies that were available but to then discount those that would require long research programmes. The study confirmed that there were opportunities to design a reactor that would be competitive when deployed at industrial sites and remote locations, and later developed for further applications.

U-Battery is being developed by Urenco in collaboration with a range of delivery partners, including: Jacobs, Cavendish Nuclear, and Kinectrics on reactor systems and safety, BWXT and NNL on fuel and fuel cycle, Rolls-Royce and Howden on key components, Costain on construction and civil works, and Mammoet and Daher on transport. The project has achieved some significant milestones, notably winning a funding under the UK Government's Advanced Modular Reactor competition, announced in June 2020, and engagement with the Canadian Regulator (CNSC) as part of their Vendor Design Review process. Development work to be undertaken in the frame of the AMR and VDR activities include design and development of the main nuclear island components (reactor pressure vessel and cross-duct, intermediate heat exchanger, helium circulator, control and instrumentation, and auxiliary systems for reactor cavity cooling, secondary shut-down, helium purification) and conventional equipment including an aero-derivative turbine and generator set, process heat exchanger, and building/civil works.

2. Target Application

U-Battery is a multipurpose reactor consisting of a standardised reactor block (comprising the primary helium and secondary nitrogen circuits, as well as a spent fuel store, and representing all of the nuclear-specific systems and components), coupled to a user-specific interface that utilises the energy contained within the nitrogen coolant. Three broad user-specific applications are envisaged: a unit focussed on providing process heat to a range of industrial applications; a unit focussed on electricity production via a gas-turbine; and a co-generation unit capable of providing a variable mix of heat and power. Potential applications include: heat and power supply to remote regions, localised heat and/or power supply for a range of existing industrial applications, and potential new applications such as hydrogen production. In common with many nuclear systems, rejected heat from the U-Battery primary applications could also be utilised for applications such as district heating and desalination.

3. Design Philosophy

The U-Battery design aims to (i) employ only established and nuclear-validated technology, (ii) maximise the use of modular manufacturing and off-site factory fabrication, (iii) be a cost-competitive and flexible energy supplier, (iv) serve different regions and diverse markets utilising a common reactor island, and (v) support alternative applications that may generate additional revenue or value. U-Battery exchanges the economies of large-scale for economies associated with localised power delivery (no transmission costs), simple safety systems (made possible by the highly robust fuel, very small thermal output, high thermal capacity of the moderator, and efficient natural heat transport), and factory-based manufacture and modular construction (again made possible by its simplicity and small size).

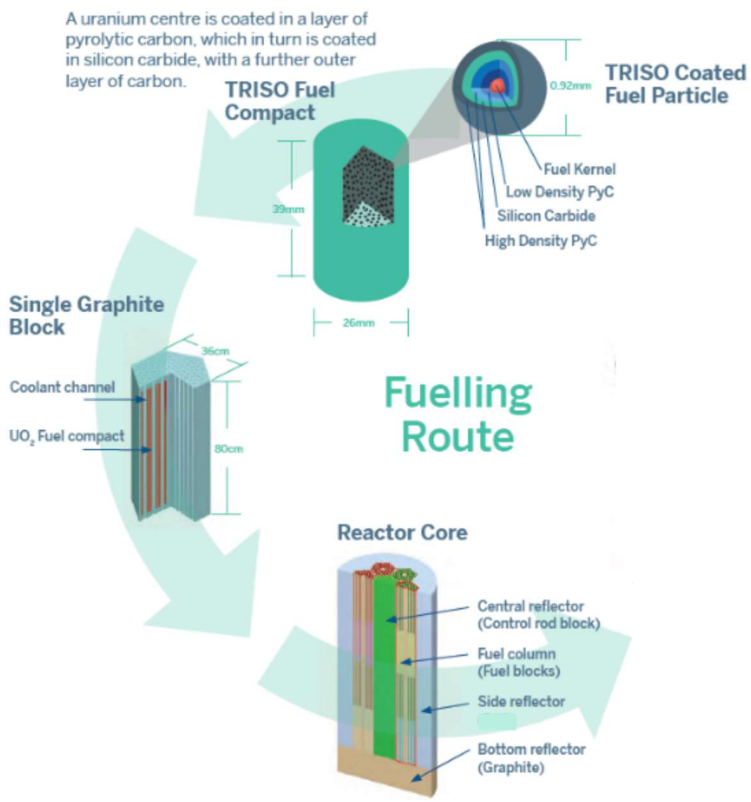
4. Main Design Features

(a) Reactor Core

The annular prismatic core is composed of 12 fuel columns, each composed of a stack of 5 hexagonal graphite fuel elements, and arranged in a 3-lobed pattern. The fuel elements contain cooling channels for the helium primary coolant, and closed channels containing cylindrical fuel compacts, each approximately 25 mm long and 12.5 mm in diameter. The central and radial moderating reflectors are formed from graphite blocks, identical in geometry to the fuel blocks such that, if necessary, they can be replaced by the fuel handling machine during refuelling. Fixed graphite blocks provide a cylindrical core-former, and location is provided by a core barrel and a restraint system similar to that used in UK gas-cooled reactors.

(b) Reactivity Control

Fixed burnable absorbers made from B_4C are used for through-cycle reactivity control and power shaping. Control rods are provided to achieve shut-down under all plant states and to allow part-power operation. Although not strictly necessary for nuclear safety, an independent shut-down system is also provided, based on small absorber spheres that can be gravity-fed into secondary shut-down channels. These provide additional regulatory confidence, and offer flexibility during shut-down (for example, if needed, control rods could be replaced early in the cycle without unloading the core).



(c) Fuel Characteristics

The U-Battery fuel compacts contain TRISO coated fuel particles, composed of a uranium kernel coated in successive layers of pyrolytic carbon, silicon carbide, and an outer layer of pyrolytic carbon. The TRISO particles are of a design recently validated as part of the US-DOE's AGR fuel demonstration program in the Advanced Test Reactor at Idaho National Laboratory, with uranium enriched to just under 20%. Although the maximum fuel temperatures in the U-Battery core never approach the well-established safety limit for TRISO fuel with UO_2 kernels (1600ºC), UCO kernels validated in the Advanced Gas-cooled Reactor (AGR) program will be employed because they offer even higher temperature capability, and were validated in greater numbers.

(d) Fuel Handling System

The fuel handling equipment for U-Battery is based on a well-proven pantograph system used in previous HTRs (and fast reactors), together with a fuel carrousel arrangement based on experience with the UK's gas-cooled reactors and on spent fuel handling experience at the UK's Sellafield site. An on-site spent fuel store is provided adjacent to the reactor core to facilitate core unloading prior to refuelling. The natural circulation cooled spent fuel store is also based on experience with similar facilities at Sellafield. Much of the fuel handling equipment is designed to be portable, and will not be left on site during periods of operation. This significantly enhances the resilience of the facility against unauthorised access to fissile material, as well as allowing optimal use of the refuelling equipment to service multiple U-Battery units.

(e) Reactor Coolant System

In common with all previous HTRs, the U-Battery employs a helium primary coolant. The possibility of employing CO_2 was briefly considered early in the project, but irradiation experiments conducted as part of the Manchester-Delft collaboration demonstrated that significant coolant chemistry intervention would be necessary to supress graphite oxidation, and this was not considered to be compatible with minimal (or even no) operating personnel. The use of a secondary steam (Rankine) power cycle was rejected for similar reasons, and also because of the positive reactivity injection that would result from a heat exchanger failure and steam ingress into the primary circuit. A direct-cycle configuration was rejected because of the significant development requirements associated with a helium turbine. The preferred solution has been to adopt an indirect cycle configuration with a nitrogen secondary coolant, which can either pass heat to a tertiary fluid for heat applications, or drive a gas-turbine based on an aero-engine but without the combustion stage and employing a closed-cycle configuration.

(f) Reactor Pressure Vessel

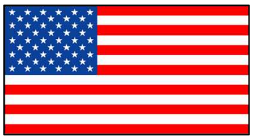
The U-Battery core is housed in a Reactor Pressure Vessel (RPV) manufactured from SA-508/533 steel in order to take advantage of the extensive experience from LWR RPVs. The RPV design is based on the ASME-III design code for a working pressure of 40 bar at 300°C, together with limited transient temperature excursions as permitted by the code. The helium coolant is passed to and returns from the primary heat exchanger via a coaxial duct housed within a cross-vessel, situated towards the bottom of the RPV reactor. Limiting the design to a single major RPV penetration reduces the impact of a major failure of the cross-duct by eliminating the "chimney effect" that could result from multiple penetrations. The RPV wall is maintained within acceptable temperature limits by the cool helium returned from the primary heat exchanger, which passes up the annulus between the RPV and the core barrel before passing into an upper plenum and being directed down through the core, into a lower plenum, and out to the heat exchanger through the inner part of the coaxial duct. The RPV is provided with a bolted head that can be removed for refuelling and inspection.

5. Plant Layout Arrangement

A plant layout has been developed based on a belowground reactor cavity and adjacent spent core storage facility, both served by an overhead crane and collocated with the power conversion module in a vented confinement building. The reactor and heat exchanger are positioned side-by-side, but each in its own compartment. Both these are served by an above maintenance floor, that also allow refuelling and has access to the used fuel cartridge store located next to the reactor cavity. Natural convection is used to cool the spent fuel assisted with the special fuel store ventilation. The turbine generator is located in a separate building (only above ground). The layout make provision for fuel to be loaded / unloaded into fuel casks with access via he fuel handling facility.

6. Development Milestones

2008	The project was initiated by Urenco and the concept design was developed by the Universities of Manchester and Dalton Institute in the UK and Technology University of Delft in the Netherlands.
2011	Feasibility study completed
2017	Memorandum of Understanding signed with Bruce Powers in Canada
2018	Green light to progress to Phase 1 of UK Government's Advanced Modular Reactor Programme
2019	Completed the first stage of the evaluation process in Canadian Nuclear Laboratories' (CNL) invitation to site a first-of-a-kind small SMR in Chalk River, Ontario. Established a service agreement with the Canadian Nuclear Safety Commission for pre-licensing Phase 1 vendor design review.
2020	U-Battery awarded funding by UK Government to conduct design and development to bring to market a new, innovative nuclear technology
2023	Development of detailed design
2025	Construction first-of-a-kind (FOAK) plant
2028	First-of-a-kind U-Battery operating



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

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MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Oklo Inc., United States of America
Reactor type	Liquid Metal Fast Reactor
Coolant/moderator	Liquid metal / no moderator
Thermal/electrical capacity, MW(t)/MW(e)	4 – 150 / 1.5 – 50
NSSS Operating Pressure (primary/secondary), MPa	Not pressurized
Fuel type/assembly array	Metal fuel
Refuelling Cycle (months)	120 – 240 months
Design life (years)	40+ years
Plant footprint (m^2)	Less than 10,000 m2
Design status	At detailed design status. Awarded a Site Use Permit by the U.S. DOE as well as fuel material by the Idaho National Lab and the U.S. DOE. Actively engaging the U.S. NRC from 2016

1. Introduction

Oklo is developing advanced fission powerhouses ranging up to 50 megawatts of electric power to provide emission-free, reliable, and affordable energy. Oklo's powerhouses can produce clean energy for decades without the need to refuel and have the capability to turn nuclear waste into clean energy.

2. Target Applications

The Oklo powerhouse is designed to provide affordable, reliable, emission-free electricity and heat. Oklo's business model is to provide power on a power purchase agreement basis.

3. Main Design Features

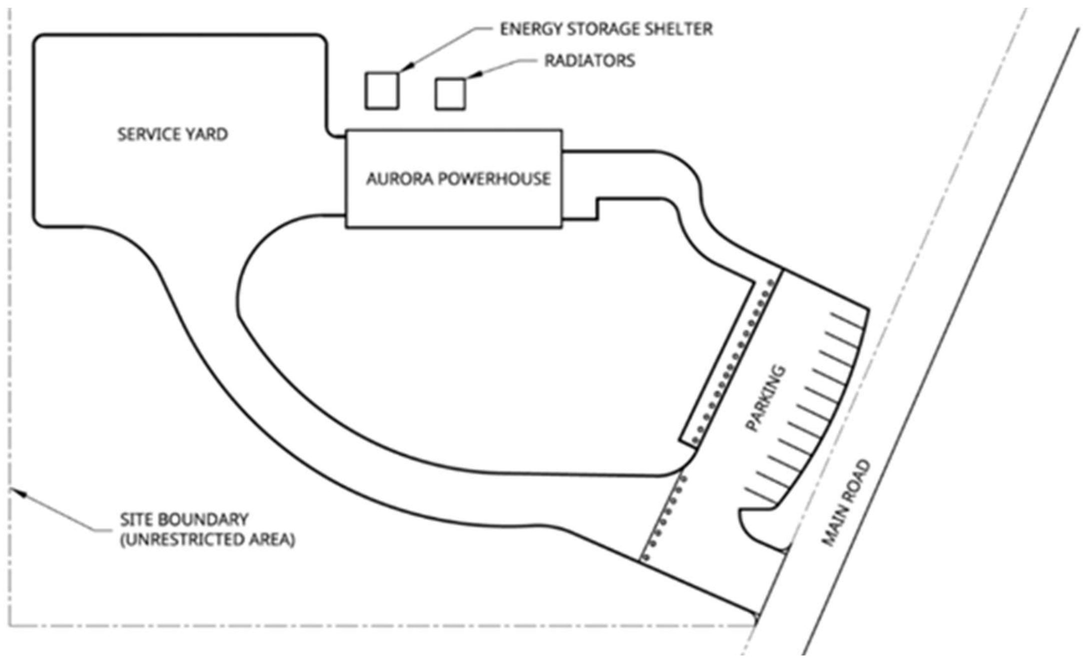
Safety is fundamentally accomplished in Oklo's design by its inherent characteristics, including:
- Low decay heat term, removed by inherent and passive means
- Inherent reactivity feedbacks ensure reactor power is controlled during overpower or overtemperature events
- Multiple barriers to fission product release
- Ambient pressure system removes sources of pressure and minimizes driving forces for release
- Water not required for safety-related cooling

4. Initial Proposed Site

Oklo is working with a range of customers related to the initial sites. It is anticipated that the first powerhouse will be sited at the Idaho National Laboratory (INL) Site in southeast Idaho, referred to as the "Aurora INL site." The Oklo Inc. site use permit request was evaluated by the Department of Energy Office of Nuclear Energy (DOE-NE), a field office of the DOE, through the site use permit process and received a permit on September 26, 2019. Additionally, the DOE provisionally approved a specific site within the INL Site for the location of the powerhouse in late 2021.

5. Plant Layout Arrangement

The Aurora site has a primary building with some supporting structures. The Aurora powerhouse has two floors and has a footprint of an acre or less. The radiators are located external to the Aurora powerhouse. The radiators exchange heat with ambient air as the ultimate heat sink for the power cycle. There is a parking lot for the site as well as landscaping surrounding the site. Due to the inherent safety characteristics, an evacuation zone outside of the site boundary is not generally required.



Plant Layout

6. Design and Licensing Status

The design is at the detailed design stage. Oklo has completed fuel prototyping activities, heat transport testing programs, and is now developing experimental advanced fuel recycling and fabrication capabilities. Oklo was awarded a Site Use Permit by the U.S. DOE and was awarded fuel material by the Idaho National Lab and the U.S. DOE. Oklo is the only company to have had a custom combined license accepted for review by the U.S. NRC. Oklo has actively engaged the U.S. NRC from 2016 to the present in pre-application activities, as well as application review activities.

7. Development Milestones

2016	Began pre-application interactions with the U.S. NRC
2017	Demonstrated prototype fuel fabrication
2018	Conducted thermal testing
2019	Granted a site use permit from the U.S. Department of Energy
2019	Awarded recovered fuel material from the Idaho National Laboratory
2020	Combined license application submittal to the U.S. Nuclear Regulatory Commission and acceptance for review (currently being revised)
2021	Commercial customer partnership with Compass announced
2021	Partnership with Centrus announced
2021	Awarded U.S. Department of Energy Technology Commercialization (TCF) Fund award related to fuel recycling
2016-2022	Awarded five awards from the Gateway for Accelerated Innovation in Nuclear for work on fuel data, fuel modeling, fuel fabrication, thermal hydraulic testing, licensing support, and more
2022	Awarded Advanced Research Projects Agency-Energy (ARPA-E) OPEN program award related to fuel recycling
2022	Awarded ARPA-E ONWARDS program award related to fuel recycling

HOLOS-QUAD (HolosGen LLC, United States of America)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	HolosGen, USA
Reactor type	High-Temperature Gas Reactor
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	22 / 10
Primary circulation	Helium
NSSS operating pressure (primary/secondary), MPa	No steam supply. Brayton Power Cycle Max 7MPa, Min 3.5MPa
Core inlet/outlet coolant temperature (°C)	590 / 855
Fuel type/assembly array	TRISO-UCO hexagonal graphite elements
Fuel enrichment (%)	19.95
Refuelling cycle (months)	96
Core discharge burnup (GWd/ton)	61
Reactivity control mechanism	Redundant independent banks of Control drums & Shutdown rods
Approach to safety	Passive cooling
Design life (years)	40
Plant footprint (m^2)	30 (reactor comprised in ISO container)
RPV height/diameter (m)	5 (reactor fits 12.19 m long ISO container)
RPV weight (metric ton)	60 (Full Mobile Power Plant)
Seismic design (SSE)	Designed for 3g max acceleration (all axis, with active magnetic bearings)
Fuel cycle requirements/approach	LEU, open cycle, spent fuel cartridges fit transport and storage dry casks
Distinguishing features	Automatic, fitting shipping containers, load-following, high efficiency.
Design status	Detailed design

1. Introduction

HOLOS-QUAD is a transportable high-performance gas-cooled reactor, with integral load following power conversion systems housed within sealed power modules (SPMs), fully comprised within standard transport containers. The design eliminates the need for the traditional BOP, greatly simplifies the design layout, reduces capital, operational, maintenance and long-term spent fuel storage cost, while overall increasing safety and efficiency. A closed-loop full Brayton gas thermodynamic cycle converts the fuel cartridges thermal energy into electricity at an efficiency of 45% with a load-following rate of 1MWe per minute. The whole microreactor system is comprised within a single 40-foot (12.19-meter) reinforced container in compliance with ISO dimensional requirements. The fuel is qualified, melt-resistant, TRISO fuel, graphite moderated, optimized to provide 22 MWt (10MWe at 45% efficiency) in excess of 8 years before refueling, with an anticipated power plant life span of 40 years.

2. Target Application

HOLOS-QUAD features high-temperature process heat that enables hydrogen production. The design features load-following capability to match power demand, as a result, it can be employed as a distributable energy source supplying power to microgrids or bulk power grids, and as a stand-alone electric island to, for example: i) charge electric vehicles; ii) supplement variable power compensating for the renewable energy sources; iii) supply power to water-vessels' propulsions. The design scalability and utilization of integral power conversion systems enables compactness for utilization as an emergency power generator rapidly deployable to disaster areas. Scaled-down versions of the design provide kW-class to MW-class generation for space applications.

Other applications include electric island capabilities to supply power to remote communities. Hardened configurations have also been developed to satisfy military submerged, terrestrial and space applications.

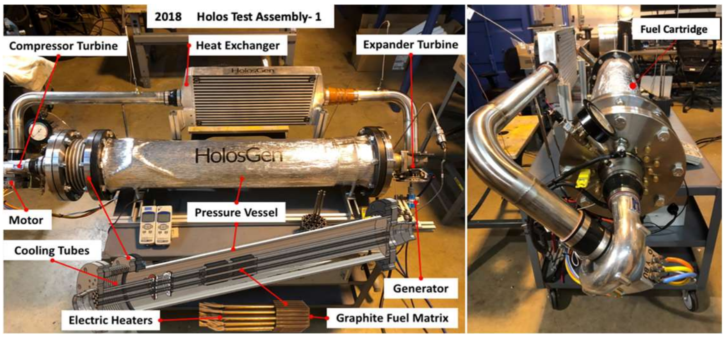
3. Design Philosophy

Safety is paramount under all operating and storage modes: the design does not reach challenging temperatures under design basis and beyond design basis accident scenarios, passively cooling the fuel under loss of coolant scenarios and total loss of electricity. The core features a very low power density. Under worst case scenarios the fuel maximum temperature briefly reaches approximately 1350°C, well below TRISO fuel melting temperature, and does not release volatiles. The Brayton high-efficiency power conversion system ensures minimum production of spent fuel per MW basis.

4. Main Design Features

(a) Power Conversion

Closed loop Brayton cycle with intercooler and recuperator heat exchangers integrated altogether with the fuel cartridge, thus eliminating the need for the traditional balance of plant.



Power Conversion System Test Assembly (HTA-1) for Brayton Cycle Proof of Concept

(b) Reactor Core

Graphite moderated TRISO-fueled with burnable poison ensuring optimized burnout with 8 Effective Full Power Years validated for safe transport and operations.

(c) Fuel Characteristics

Qualified TRISO fuel with UCO kernel.

(d) Reactivity Control

The reactivity controls are represented by independent, diversified, and redundant reactivity control systems based on sets of control drums and sets of shutdown rods.

(e) Reactor Pressure Vessel and Internals

The design features 4 identical 5.5MW(t) SPMs, each equipped with Brayton power conversion components in a configuration wherein the cycle pressure boundary is maintained by high pressure tubes. A simpler configuration enables accelerated licensing by utilizing a nuclear-stamp RPV.

(f) Reactor Coolant System

Core and auxiliaries are cooled by helium thermally coupled with the ultimate heat sink represented by air, or water for terrestrial applications, air or water for water-vessel propulsion, and radiators for space applications.

(g) Secondary System

The secondary system is only for process heat application, wherein helium circulates on one side of an intermediary heat exchanger and a user/application-selected working fluid circulates on the secondary side.

(h) Steam Generator

The design does not need steam generators, however, steam can be generated as part of process heat for selected applications.

5. Safety Features

(a) Engineered Safety System Approach and Configuration

The design is equipped with traditional HTGR safety features. The design configuration represented by 4 identical SPMs features additional redundant pressure boundaries preventing migrations of radionuclides from the fuel cartridges (e.g., from potentially defective or damaged TRISO elements). In this design configuration, the working fluid (helium) is only thermally coupled to the core: there is no physical contact between the working fluid and core internals. The design relies on passive cooling under environmental extremes, loss of coolant and loss of electricity so that the fuel temperature remains well below safety margins. The results of analyses on reactivity control aspects of the HOLOS-QUAD design under load-following operations published.

(b) Decay Heat Removal System / Reactor Cooling Philosophy

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 temperature spikes under total loss of coolant circulation and total loss of coolant scenarios. The validation on thermal-hydraulic and neutronics performance, as well as the effectiveness of passive decay heat removal of the HOLOS-QUAD have been published by the Argonne National Laboratory.

(c) Spent Fuel Cooling Safety Approach / System

Spent fuel cartridges are inserted inside relatively small dry casks. Cooling is executed passively by natural convection with the air surrounding the shielded dry cask.

(d) Containment System

The design utilizes TRISO fuel which features "functional containment": radionuclides resulting from operations are trapped within TRISO microspheres. In addition to the functional containment, the design features two additional pressure boundaries that prevent radionuclides migrating from potentially defective or damaged TRISO microspheres to the working fluid, into the environment.

(e) Chemical Control

The design eliminates the balance of plant normally dedicated to filter and makeup the working fluid, and the redundant active magnetic bearings eliminate "metal-to-metal" contact between rotary and stationary components, even during transport.

6. Plant Safety and Operational Performances

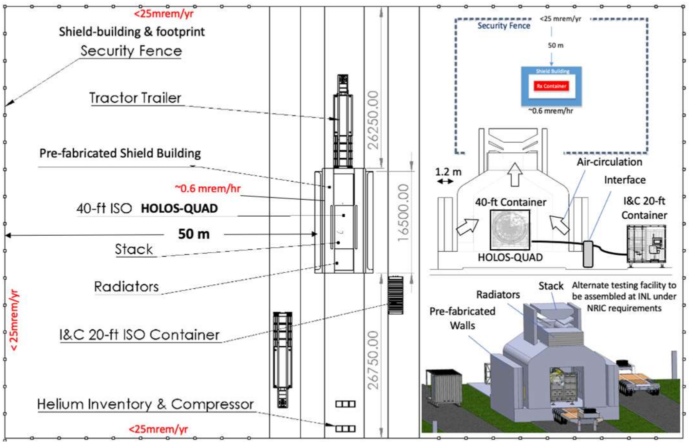
The design follows the characteristics of HTGRs and features a very low power density core and a very high heat capacity. These characteristics result in slow fuel temperature variations, which under worst case scenarios do not challenge the high temperature fuel loaded in the reactor under all postulated off-normal and accident scenarios.

7. Instrumentation and Control System

The design follows the electrical load automatically and features a control system that regulates the power flows between the thermal-to-electric conversion components, the ballast/black-start batteries, and the electrical load. Process instrumentation and control systems regulating the active magnetic bearings, control drum and shutdown rod drives, and the compressor speed, enable fast-acting power maneuvering during normal operations. The digital instrumentation and control equipment is located within a 20-foot "digital I&C" ISO container electrically coupled to the 40-ft ISO container via shielded, EMI- and EMP-shunted data and power cables.

8. Plant Layout Arrangement

The design features a highly integrated, transportable nuclear power plant fully comprised within the dimensional constraints of a standard shipping container with feasibility validated by the Argonne National Laboratory. For above surface applications, the ISO container is positioned via tractor trailer inside a prefabricated non-permanent concrete "shield building". For underground operations, the ISO container can be "verticalized" (the design's active magnetic bearings enable positioning of the container horizontally, angled, or vertically). The vertical ISO container can be positioned within a shielding bored hole with a prefabricated concrete liner and passive air-cooling pathways.



Pre-fabricated non-permanent shield building and dose rate calculations at the 50-m fence

9. Testing Conducted for Design Verification and Validation

The design has been scrutinized for approximately 4 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 under this program included technical and economic feasibility executed with high-fidelity codes developed by the national laboratories.

10. Design and Licensing Status

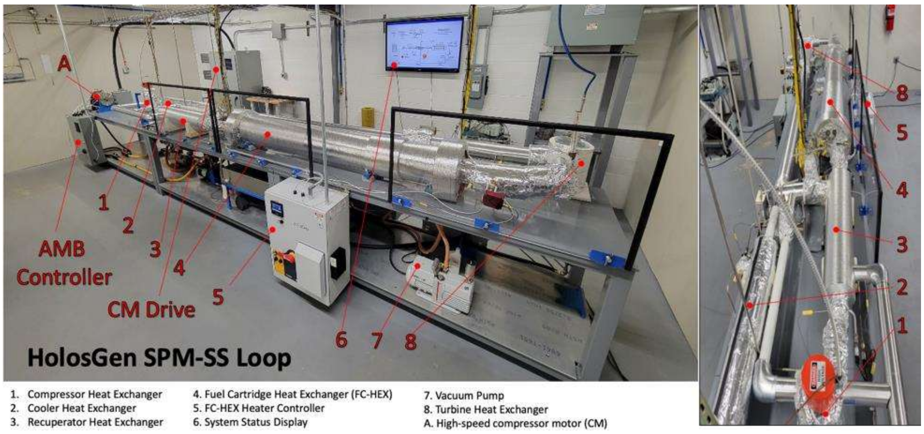
Preliminary safety performance analyses have been completed and verified through operations of the SPM subscale simulator (SPM-SS) configured to mimic design basis accident scenarios. Preliminary safety assessments through analyses executed with the cooperation of teams at the national laboratories and results validated via the electrically-heated SPM-SS were completed in 2022. A formal safety assessment request is planned for submission to the U.S. Nuclear Regulatory Commission in 2023.

11. Fuel Cycle Approach

The 10MWe HOLOS-QUAD design is equipped with a core optimized for 8 EFPY. Spent fuel cartridges passively dissipate decay heat from reactor shutdown, and can be placed within dual purpose transport and storage shielding dry casks within 72 hours from shutdown. The spent fuel cartridge dimensions altogether with those of the dry cask are designed to comply with ISO shipping container dimensional constraints. The spent fuel cartridges will be stored at long-term storage facilities and the spent fuel in them will not be processed as the type of fuel utilized is already in a stable form for long-term storage.

12. Waste Management and Disposal Plan

As the coolant in this design is not physically in contact with the materials forming the fuel cartridge internals, there is no working fluid contamination. As a result, the great majority of the power conversion components and their ancillary components are not exposed to irradiation effects (e.g., particulate circulating through the power conversion components) and will not require specialized decommissioning procedures. These components will be classified as non-radioactive waste, cleaned and refurbished, or disposed of. Cost of spent fuel handling, transport and storage for 50 years is included in the HOLOS-QUAD costing analyses.

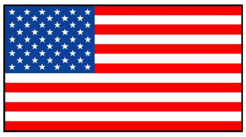


Sealed Power Module Subscale Simulator (SPM-SS) Validation Test Assembly

SPM-SS with helium at 7MPa

13. Development Milestones

2015	Conceptual design for F&A Holos Reactor.
2018	Demonstrated feasibility of the power conversion system using the HTA-1 test assembly
	Awarded the DOE ARPA-E MEITNER funding to validate HOLOS-QUAD design through modelling, simulation and testing.
2021	Awarded the DOE-NE GAIN Voucher with Argonne National Laboratory to develop innovative material solutions to enhance core performance.
	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.
2022	Completed design validation demonstrating: 45% net plant efficiency; fully fitted within a single 40-foot ISO shipping container; 8 EFPY, passive decay heat removal; and load-following capability at a rate of 1MWe/min. Key design aspects validated via testing.



MARVEL Research Microreactor (Idaho National Laboratory, USA)

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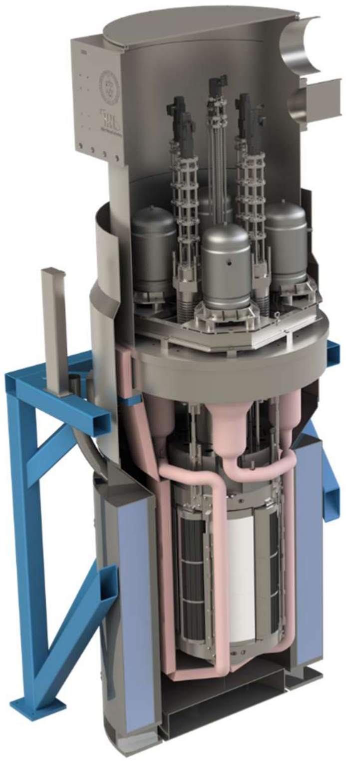


MAJOR TECHNICAL PARAMETERS	
Parameter	Value
Technology developer, country of origin	Idaho National Laboratory, USA
Reactor Type	Liquid Metal Cooled Thermal reactor
Coolant/Moderator	Sodium-Potassium Eutectic, Hydrogen in fuel
Thermal/electrical capacity, MW(t)/MW(e)	0.075-0.1 / 0.015-0.027
Primary Circulation	Natural Convection
NSSS Operating Pressure (primary/secondary)	0.39 / 0.22
Core Inlet/Outlet Coolant temperature (C)	480 / 548
Fuel Type/Assembly array	Uranium Zirconium Hydride
Number of Fuel Assemblies in the core	6
Fuel Enrichment	19.75%
Refueling cycle (months)	> 60
Core Discharge Burnup (GWd/ton)	2.098 (current; can be increased if needed)
Reactivity Control Mechanism	Four ex-core, safety-related control drums, One defense-in-depth central shutdown rod
Approach to safety systems	Doppler-based inherent safety in fuel; Primary coolant, intermediate coolant, and decay heat removal is conducted via passive, natural convection. Loss of on-site power actuates shutdown mechanism passively
Design Life (years)	2-40 (depending on transients)
Plant footprint (m2)	8.9
RPV height/diameter (m)	2.55 / 1.22
RPV weight (metric ton)	5.94
Seismic Design (SSE)	IBC 2015 Risk Category IV Site Class C
Fuel Cycle Requirements/approach	Has refueling capability, but its current system life is limited to 2 years
Distinguishing Features	Compact accessible core; naturally driven, integrated power conversion systems; adjustable high-grade heat extraction; low-grade heat extraction capability; small footprint
Design Status	Equipment manufacturing in progress

1. Introduction

The Microreactor Applications Research Validation and Evaluation (MARVEL) Project entails the design, development, construction, and startup of an INL test microreactor, funded by the United States Department of Energy (DOE) via the Microreactor Program (MRP). The goal of the project is to establish an operational nuclear applications test bed that can generate combined heat and power to enable integration and R&D with end-user technologies, as well as allow microreactor technologists to test next-generation control systems. The microreactor is a thermal reactor utilizing Uranium Zirconium Hydride (UZrH) fuel with review and authorization 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, approved facility safety basis, 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, Qnergy, Munro & Associates, and Creative Engineers Inc.

The primary purpose of investigating a near-term concept is to determine its feasibility and utility for future microreactor research and development, such as a test bed for integrating microreactors with electric and non-electric applications and demonstrating the technology to key end-users. It can also drive infrastructure development and capabilities to support future microreactor demonstrations.

The U.S. DOE Microreactor Program supports research and development (R&D) of technologies related to the development, demonstration, and deployment of very small, factory-fabricated, transportable reactors to provide power and heat for decentralized generation in civilian, industrial, and defense energy sectors. Such applications currently face economic and energy security challenges that can be addressed by this new class of innovative nuclear reactors. Led by INL, the program conducts fundamental and applied R&D to reduce the risks associated with new technology performance, manufacturing readiness, and deployment of microreactors. The program aims to verify that microreactor concepts can be licensed and deployed by commercial entities to meet specific use case requirements. Microreactors, often referred to as special-purpose reactors or very small modular reactors, are factory manufacturable, easily transportable, and designed to produce tens of megawatts of thermal (MWth) energy. This power limit allows microreactors to be classified as Hazard Category 2 per the Code of Federal Regulations (CFR) at 10 CFR 830 and DOE-STD-1027. These reactors are decentralized energy sources that can provide sustainable and affordable heat and power to remote communities and industrial users while having self-contained geometry that requires very low maintenance. Microreactors are intended to be self-regulating and do not rely on engineered systems to ensure safe shut down and removal of decay heat.

2. Target Application

MARVEL is targeted at researchers, technology developers, and end-users for enabling the successful deployment and adoption of microreactors through research and education. Some examples of R&D activities are:

<u>Test, demonstrate, and address issues related to installation, startup, and operation</u>

- Simplify siting and environmental review process
- Startup methodology for microreactors
- Normal operating transients such as startup and load management
- Cyber and physical security hardening
- Seamless integration to a net-zero, electrical microgrid
- Demonstration of high and low-grade heat extraction.

<u>Enable Autonomous Operation Technologies</u>

- Automate operator functions, while maintaining reactor safety
- Demonstrate radiation and temperature-hardened sensors and instrumentation to enable remote monitoring, advanced sensor reliability tests, and online calibration
- Live data can feed a digital twin of the reactor to “train” an artificial intelligence-based control system
- Demonstrate wireless transmission of live data of both electrical and thermal power output during startup, operation, and shut down. This allows real-time feedback on system output, performance, and prediction of any unplanned maintenance needed in an operating microreactor.

<u>Enable Seamless Application Integration</u>

- The control systems manage the energy grid demand and reactor power and heat supply. This management requires a carefully designed control system that can predict the interplay of controls, thermal inertia, and reactivity feedback.
- Demonstrate integration approaches for a range of applications investigating both reactor power management and load management approaches

3. Design Philosophy

The design philosophy is to select technologies that can lead to criticality within the shortest time possible and not necessarily on the technology readiness level. Time is the essential metric of project performance without sacrificing safety and quality. This drives the team to select technologies that are either (i) off-the-shelf, (ii) can be implemented, and/or (iii) can be developed to accelerate technology and manufacturing readiness levels. Innovation and creativity are embedded in the team culture and celebrated. This includes adopting multiple engineering tools and processes from non-nuclear technology industries, e.g. Scrum Agile Product development, and Design for Manufacturability (DFM) modeling from the automotive industry.

4. Main Design Features

(a) Power Conversion

MARVEL utilizes commercial, off-the-shelf Stirling Engines from Qnergy. Each Stirling engine can produce 5-7 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.

(b) Reactor Core

The MARVEL reactor will contain fuel elements sourced from TRIGA International (TI): LEU, 30% uranium, by weight, 19.75 % enriched (TRIGA LEU 30/20). Beryllium and graphite reflected, ZrH moderated, solid fuel, tank type reactor. The MARVEL reactor core comprises 36 LEU fuel elements arranged in a triangular pitch lattice around a central voided location for the central insurance absorber (CIA) rod assembly. Surrounding the core is a thick axial neutron reflector composed of beryllium oxide.

(c) Fuel Characteristics

The major components of the MARVEL fuel system are (1) U-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.

(d) Reactivity Control

The MARVEL reactor also has a very high net negative temperature feedback for prompt reactivity control inherently due to the Doppler broadening of resonances. Manual and Passive Reactivity Control and shutdown are achieved with four custom-designed, safety-related Control Drums (CDs) and a defense-in-depth Central Insurance Absorber (CIA) that control reactivity via motor drivers in the control cabinet to position the drums and rods within the reactor structure. Passive actuation functions are built into the design for loss of power and inadvertent energizations of the motors.

(e) Reactor Pressure Vessel and Internals

The Primary Coolant System (PCS) is a high-temperature, low-pressure boundary that houses the core internals, reactor primary coolant, and argon gas headspace. In addition, the PCS passively maintains decay heat removal capability. The boundary is a metal weldment made from 316H stainless steel for high-temperature reactors designed per ASME Section III Division 5.

(f) Reactor Coolant System

The primary coolant system (PCS) is a four-loop hydraulic circuit assembled to transport nuclear fission heat from the nuclear fuel to the intermediate heat exchanger (IHX) using the natural convection flow of the primary coolant. The PCS also transfers decay heat to the ultimate heat sink. Approximately 120 kg of NaK, liquid metal at room temperature, serves as the primary coolant. The Stirling engine heat exchangers connect to the reactor vessel and interface with the NaK coolant via the IHX containing lead. The Stirling engine coils or high-grade heat exchanger, depending on configuration, extract heat from the primary coolant and reduce the NaK temperature. The cooled and denser NaK then flows outward to the periphery of the core, then downward through four downcomer pipes outside the beryllium oxide side reflector and through in the lower plenum. The NaK then rises back up through the active core under natural circulation forces driven by the heated section of the active core.

(g) Secondary System

Lead has 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.

5. Safety Features

MARVEL employs a high safety pedigree for (i) reactivity control, (ii) safety shutdown, and (iii) decay heat removal by relying primarily on physics rather than engineered systems. The fuel has a high prompt net negative temperature reactivity feedback due to the strong Doppler broadening of resonances by having hydrogen close to the fuel. 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, and top & bottom guard vessel to confine hazardous materials throughout system life in all operational, postulated design-basis, and beyond-design-basis events.

MARVEL Integral Effects Test systems, structures, and components

6. Testing Conducted for Design Verification and Validation

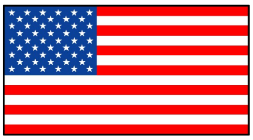
The MARVEL team conducts rapid prototyping tests to mature its technologies. So far, more than ten separate effects tests have been conducted, including, but not limited to, the intermediate heat exchangers, control drums, instrumentation control, neutron detection, shutdown rod actuators, and Stirling engines.

Due to MARVEL's novel thermal hydraulic circuit, where a liquid metal natural circulation primary loop is in series with four parallel liquid metal natural circulation loops, an integral effects test (IET) of the system was considered necessary for verifying the transient dynamics of the system before reactor construction. Hence the team has successfully designed and fabricated a full, scale electrically heated prototype of the MARVEL reactor. This is conducted per ASME boiler and pressure vessel code (Section III and VIII) within nine months by employing an agile development process. The test hardware includes (i) a Full-scale mechanical IET test article; (ii) eight electrical control cabinets; (iii) a structural frame; (iv) four IET flow meters; (v) more than 200 thermocouples and pressure transducers; and (vi) four Stirling engines, engine control, and heat rejection units.

The goals of this IET are to: (i) validate flow and heat transfer characteristics of MARVEL technology; (ii) benchmark modeling & simulation parameters; (iii) streamline manufacturing methods; (iv) de-risk supply chain; and (v) train operators.

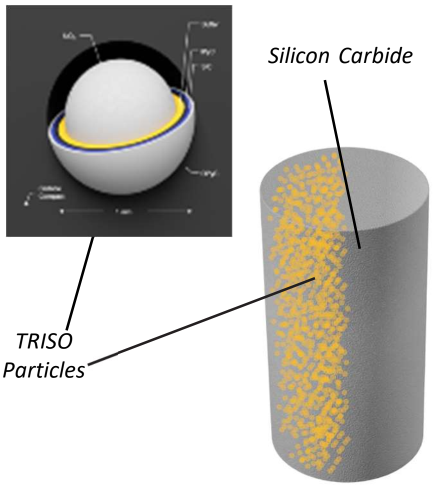
7. Design and Licensing Status

DOE conducted an environmental assessment (EA) as part of the NEPA process, which analyzed the potential environmental impacts of constructing the MARVEL microreactor inside Idaho National Laboratory's (INL's) Transient Reactor Test Facility. At the conclusion of the EA process, DOE issued a final EA with a "finding of no significant impact" (FONSI) for the MARVEL Project (DOE/EA-2146). A safety design strategy (SDS-119, Rev 1) has also been approved by DOE. The MARVEL reactor is currently in Final Design Phase, with 90% construction planned to conclude in Q2, CY23, and fuel load and initial criticality in Q2, CY24.



MMR® (Ultra Safe Nuclear Corporation, United States of America)

MMR Unit



Fully Ceramic Micro-encapsulated (FCM™) fuel

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Ultra Safe Nuclear Corporation, USA
Reactor type	High Temperature Gas-cooled Reactor / micro-reactor / nuclear battery
Coolant/moderator	Helium / Graphite
Thermal/electrical capacity, MW(t)/MW(e)	15 / >5 and 30 / >10
Primary circulation	Forced circulation
NHSS Operating Pressure (primary/secondary), MPa	3 / 0.1
Core Inlet/Outlet Coolant Temperature (ºC)	Helium 300 / 630, solar salt stored at 560
Fuel type/assembly array	FCM® (TRISO based) graphite / Hexagonal
Fuel enrichment (%)	HALEU 19.75%
Core Discharge Burnup (GWd/ton)	> 60
Refuelling Cycle (months)	240 (20 years)
Reactivity control mechanism	Control rod insertion, negative temperature coefficient.
Approach to safety systems	Passive (Category A), no moving parts, does not require fluid or natural convection
Design life (years)	20 (fuel core), 40 (vessel)
Plant footprint (m^2)	80 x 80 (citadel area)130 x 96
RPV height/diameter (m)	13.25 m (including lid and stand pipes) / 3.5 m
RPV weight (ton)	30.1 metric tonnes
Seismic Design (SSE)	0.3g
Fuel cycle requirements / Approach	20 year core life, can be refuelled for an additional 20 year cycle
Distinguishing features	No core meltdown; modular reactor and modular powerplant, adjacent non-nuclear power conversion plant; no EPZ required; load following / fully dispatchable; nuclear reactor isolated from load via molten salt loop; < 6 months assembly at site
Design status	Basic Design, project in formal licensing with the regulator in Canada

1. Introduction

The MMR® system is a small modular nuclear energy system that delivers safe, clean and cost-effective electricity and heat to remote mines, industry and communities. The energy system consists of two plants, the nuclear plant and the adjacent non-nuclear power plant. The Nuclear Plant is independent of the Adjacent Plant, requiring no supporting services for any event for its safe operation. The Adjacent Plant consists of the equipment and systems that convert the process heat to electrical power or other forms of energy as per client requirements. The Nuclear Plant generates approximately 15 MW(t) of process heat that can supply electrical power and/or heat the end users. The electrical power can also be supplied to an area grid. The fuel life is 20 years with the possibility of refuelling for an additional 20-year cycle.

2. Target Application

The MMR power plant is designed for remote communities as standalone micro grid or for heavy industry applications such as process heat and hydrogen production. MMR energy system can scale to match demand with multiple units at the same site.

3. Design Philosophy

The design philosophy of the MMR uses the following main principles: (i) Use proven and existing technology; (ii) Design for simplicity; (iii) Design for reliability; (iv) Separate the standardized nuclear plant from the application adjacent plant; (v) Design for low maintenance; (vi) Design for modular construction and replacement, and (vii) Design all components to be replaced in principle. The design builds on the High Temperature Gas Cooled Reactor knowledge base and is constrained by known material limits as determined by design codes and standards. The reactor is fuelled to operate for 20 years with the possibility of refuelling for an additional 20-year cycle.

4. Main Design Features

(a) Power Conversion

Electricity is generated in the power generation system located in the adjacent plant from the heat supplied from the nuclear plant via the molten salt. The power generation system consists of the turbine generator and supporting infrastructure. The adjacent plant has a main electrical grid connection for supply of the electrical power generated via transmission infrastructure.

(b) Reactor Core

The reactor core consists of hexagonal graphite blocks that contain stacks of the Fully Micro Encapsulated (FCM) fuel pellets. The MMR reactor core has a low power density (1.24 W/cm^3) and a high heat capacity. This results in very slow and predictable temperature changes. The MMR reactor is fuelled once for its initial 20-year operating life and sealed. The core can then be replaced to extend the operating life of the reactor for another 20-year cycle. The core provides adequate coolant flow paths for heat removal, and the graphite material of the blocks assists with further heat removal.

(c) Fuel Characteristics

The MMR reactor uses Ultra Safe Nuclear Company proprietary Fully Ceramic Micro Encapsulated (FCM) fuel. FCM fuel provides a new approach to inherent reactor safety by providing an ultimately safe fuel. FCM is based on very small particles containing low-enriched uranium. FCM fuel ensures containment of radioactivity during operations and accident conditions, which means that no fission products are released from the fuel.

FCM is manufactured with industry-standard Triple Coated Isotropic (TRISO) fuel particles, whose primary purpose is to retain fission products. The TRISO fuel, which contains the radioactive by-products of fission within layered ceramic coatings, is further encased within a fully dense silicon carbide matrix. 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.

(d) Fuel Handling System

The MMR is initially fuelled for a 20-year cycle, and may be refuelled for an additional 20-year cycle. There is no long-term fuel storage on site.

(e) Reactivity Control

The core provides for areas for insertion of control rods. The MMR reactor core has a low power density and a high heat capacity resulting in very slow and predictable temperature transients. The core has strong negative temperature feedback.

(f) Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel (RPV) houses the core. The MMR has a drain at the base of the Reactor Pressure Vessel and avoids major water ingress events because there is no steam generator in the primary circuit. The MMR Reactor Pressure Vessel is 'helium tight'. The MMR design includes metallic seals and a seal weld to ensure leak tightness. The Reactor Pressure Vessel is designed in accordance with the ASME Boiler & Pressure Vessel Code (BPVC).

The colder helium flows up against the reactor vessel wall to avoid overheating from the core outlet gas. The MMR employs a design principle in the primary circuit that ensures that the hot helium is always surrounded by cold helium at a higher pressure. This means that any leaks are from cold to hot and will eliminate the possibility of hot gas impinging on the pressure boundary.

The MMR design studies the effect of core bypass flow by modelling extreme and bounding cases. The MMR design avoids problems with flow across tube bundles in the intermediate heat exchanger by using printed circuit technology that does not have any tubes.

5. Safety Features

The MMR technology has been developed by Ultra Safe Nuclear Company and is based largely on proven designs with inherent safety features. The safety design is then further augmented with specific novel safety features. The use of proven inherent safety design features confers confidence in the operability and safety of the facility, while the novel safety features enhance the safety of the technology. Operations are simple with minimal operations and maintenance requirements, and no on-site fuel storage, handling, or processing.

The MMR reactor is a truly walk away, safe power reactor. In the case of a fuel or cooling system failure, the MMR reactor cannot melt down as the temperature will remain at a safe level while heat dissipates passively into the environment.

(a) Engineered Safety System Approach and Configuration

The MMR safety functions are achieved through a combination of inherent safety features and engineered safety features. The inherent safety features result from the selection of materials used in and design features of the reactor fuel and core, moderator and coolant. Engineered safety features are those introduced specifically to perform a safety function, and they may include both passive and active structures, systems and components (SSC).

(b) *Decay Heat Removal*

The reactor will maintain a safe temperature indefinitely even in the event of a cooling failure because of the fuel characteristics and the reactor design. In a cooling failure, the reactor is cooled through passive heat transfer. External water, electricity, and additional active systems to dissipate heat during a system failure are not required.

(c) *Chemical Control*

The inherent features for controlling chemical attack of the fuel of the MMR design include:

- Design features that limit water ingress
- The circulator magnetic bearings eliminate the source of water that was a major contributor to the low-capacity factor of the Fort St. Vrain HTGR. The primary heat transfer cycle and steam generation cycle has been separated with the addition of a low-pressure intermediate loop.
- The primary cycle which is pressurized to limit air ingress
- The reactor core graphite will not ignite if were to come into contact with air even at operating temperatures• The primary cycle coolant (helium) is chemically inert
- The use of ceramically-coated TRISO as a basis of the fuel
- The full ceramic microencapsulation of the TRISO fuel particles to further control chemical attacks• The use of a nuclear-grade vessel

(d) Containment

The MMR's fuel performs the function of containing fission products during such conditions. This sets the MMR apart from most current operating reactor technologies, which rely on highly specialized and complex safety systems to prevent and mitigate further releases of fission products that escape their fuel in the case of postulated accidents. Retention of fission product is achieved by the FCM fuel particles. The fuel safety margin is so large that fission product retention is accomplished entirely by the fuel. No other containment is needed. The reactor core is sealed for life – the fuel cannot be accessed.

6. Plant Safety and Operational Performances

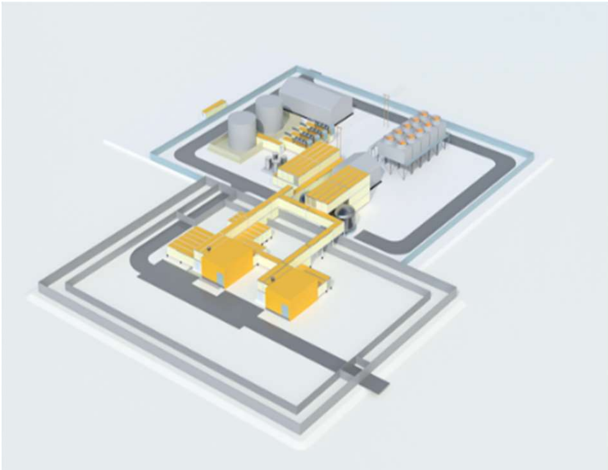
The MMR technology has been developed by Ultra Safe Nuclear Company and is based largely on proven designs with inherent safety features, further augmented with specific novel safety features. The degree of such proven inherent safety design features confers confidence in the operability and safety of the facility, while the novel safety features further enhance the confidence in the technology. Operations are simple with minimal operations and maintenance requirements.

7. Plant Layout Arrangement

The MMR facility includes a Nuclear Plant containing an MMR reactor, and an Adjacent Plant, which are the main physical works related to the Project.

(a) Nuclear Plant

The Nuclear Plant provides process heat to the Adjacent Plant where it is converted to electrical power and/or heat as per client requirements. The Nuclear Plant contains the Nuclear Building and the Citadel Building, where the MMR reactor and its associated Nuclear Heat Supply System are housed. This building protects the reactor and the Intermediate Heat Exchanger from hazards (both external and internal to the Citadel Building), and the wall provides biological shielding.



MMR two-unit Power Plant

(b) Adjacent Plant

The Adjacent Plant buildings contain the equipment required for the generation of electricity from the heat supplied by the Nuclear Plant and to interface with any customer end-use facilities. The Adjacent Plant Molten Salt Heat Storage System acts as an intermediary to transport and store the heat generated in the Nuclear Plant and transfer it to a steam cycle for the purpose of generating power and heat for customer applications.

8. Design and Licensing Status

Global First Power (GFP) has submitted a 'License to prepare site initial application' for the MMR Demo power plant at the Chalk River site to the Canadian Nuclear Safety Commission (CNSC) which regulates all nuclear activities in Canada. Formal licensing process is underway for the MMR Project at Chalk River, in Canada. The MMR commercial demonstration project has a Project Hosting Agreement with an identified site at Chalk River Laboratories in Canada. The formal licensing process is in the first phase (License to Prepare Site), which will be followed by the License to Construct phase.

9. Fuel Cycle Approach

The MMR will be fuelled to operate for 20 years. Refuelling is possible to enable a second 20-year cycle, with the balance of plant designed for a 40-year life. The Decommissioning phase activities are anticipated to take approximately 6 months. Spent fuel will be stored according to the national storage plan. No processing or conditioning of the spent fuel is required because FCM fuel is already in a geologically stable form.

10. Waste Management and Disposal Plan

All waste will be handled and processed in a responsible and safe manner that ensures minimum exposure to all personnel handling, transporting and processing the waste. Waste will be segregated at source as non-radioactive waste and radioactive or potentially radioactive waste. Waste will be temporarily stored on the Project site in defined areas and transported to authorized processing facilities at appropriate times, dependent on the category and type of waste.

11. Development Milestones

Ultra Safe Nuclear Company MMR Demo unit at Chalk River (Canadian Nuclear Laboratories)

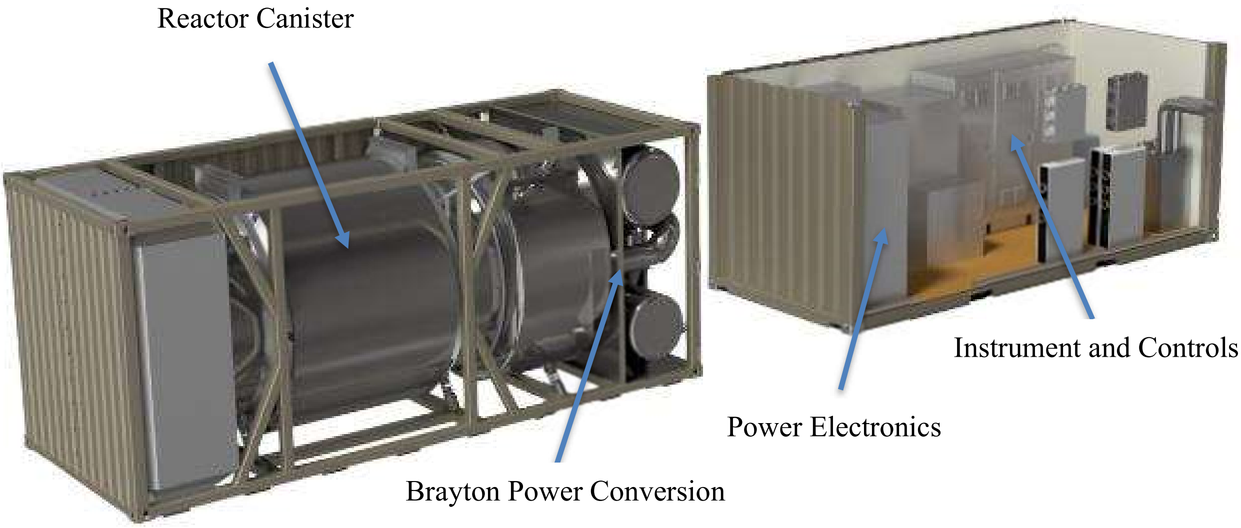
2011	Secured FCM™ fuel and MMR™ reactor patents
2016	Established R&D and fabrication laboratories
2017	Initiated FCM™ fuel development and qualification
June 2018	GFP submits proposal to CNL, supported by OPG & Ultra Safe Nuclear Company
February 2019	CNL announces GFP proposal enters Stage 3 proponent review process
March 2019	GFP submits license to prepare site initial application to CNSC
2020-2023	License to prepare site, complete environmental assessment, and begin site preparation
2021-2027	Site Preparation and Construction
2026-2052	Plant Operation
2052-2060	Decommissioning and abandonment

Westinghouse eVinci™ Micro Reactor (Westinghouse Electric Company LLC, USA)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	Westinghouse Electric Company LLC, USA
Reactor type	Heat pipe cooled
Coolant/moderator	Heat pipes / metal hydride moderator
Thermal/electrical capacity, MW(t)/MW(e)	7 – 12 / 2 – 3.5
Primary circulation	Heat pipes
NSSS Operating Pressure (primary/secondary), MPa	N/A
Core Temperature (ºC)	NA800
Fuel type/assembly array	TRISO or another encapsulation
Number of fuel assemblies	Monolith core
Fuel enrichment (%)	5 – 19.75
Core Discharge Burnup (GWd/ton)	Not Disclosed
Fuel cycle (months)	> 36
Reactivity control mechanism	Ex-core control drums
Approach to safety systems	Inherent and passive safety for shutdown and heat removal
Design Life (years)	40
Plant footprint (m^2)	< 4 000
RPV height/diameter (m)	N/A
Seismic design (SSE)	IBC Zone 4 Category F
Fuel cycle requirements / Approach	No onsite refuelling Replace reactor approach
Distinguishing features	Transportable reactor that can operates autonomously
Design status	Conceptual design

1. Introduction

The Westinghouse eVinci Micro Reactor is designed for energy generation in remote or isolated locations. The design can produce both process heat and electricity for remote communities, mining operations, or critical infrastructure installations. The key attribute of the design is its transportability within standard shipping containers. The design is based on heat-pipe reactor technology that has been developed and tested by Los Alamos National Laboratory for space applications. Because of its compact and simplified design, the eVinci Micro Reactor will be manufactured and fuelled in a factory, and then transported to an end user site. The figure below shows how the eVinci reactor and power conversion system can be packaged into two standard transport containers. One of the containers houses the reactor and the power conversion system. The other container includes power electronics and the instrumentation & controls (I&C) system.



Westinghouse's eVinci Micro Reactor schematic.

2. Target Application

The eVinci Micro Reactor is designed specifically to serve remote communities, mining operations, or military installations. It combines both heat and power generation capabilities, addressing diverse energy needs of these decentralized and off-grid markets.

3. 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 providing overall plant simplicity. The design utilizes the inherent safety features in the fuel, moderator and heat pipes to enhance safety and self-regulation capability.

4. Main Design Features

The fundamental reactor design is based on a solid core block that includes a matrix of nuclear fuel, moderator and heat pipes that extract the heat from the core region. The reactor core by itself is subcritical. It requires radial and axial reflectors to improve neutron utilization. Reactivity control and shutdown are performed with radial control drums placed around the core. Shutdown can also be performed with shutdown rods that can be inserted into the core block. The control drums are the only moving parts within the reactor canister leveraging the passive heat pipe thermal exchange. The reactor heat transport design utilizes the unique properties of sodium heat pipes. Sodium heat pipes are embedded in the solid core and under normal operation are used to move the core heat to the heat exchanger and associated power conversion system. Heat pipe operation removes the need for coolant pumps in the primary system. The complete reactor and power conversion systems are controlled with an autonomous control system that is based on proven nuclear I&C systems.

The eVinci Micro Reactor has inherent safety features. For example, in case of decay heat removal, the core design within the reactor canister enables heat removal by conduction to the outer sections of the reactor canister that passively allow heat transfer to the surrounding atmosphere (air). The core is designed with negative reactivity feedback, allowing for increased safety in the event of an accident scenario. If external power is lost or the reactor is tripped through its autonomous control system, the control drums automatically rotate to a high neutron absorption position around the core and subsequently shutdown the reactor. This solid core design eliminates many traditional accident scenarios such as high-pressure pipe ruptures or loss of coolant accidents.

An open-air Brayton system was incorporated due to its compactness, technical maturity and reliability. A single shaft gas turbomachinery composed of a compressor, turbine, and alternator operating as a single rotary spool. The high-speed single-shaft architecture results in a very compact power module. An annular recuperate and simplified, low pressure ducting will enable close integration of the engine with the primary heat exchanger. To avoid the use of lubricants, magnetic bearings will allow the engine to operate in any orientation, facilitating optimal integration and packaging. The high-speed alternator acts as a motor to start the engine during regular and black start up. The mechanical configuration is critical to starting the engine with the relatively slow reactor ramp-up and avoidance of thermal stress. Power electronics are combined with the variable speed engine to enable efficient part-load and load following power management.

5. Resilient Features

Resiliency of eVinci Micro Reactor is enabled through packaging within a secure canister, installed in a strong secure vault on site and connected to a micro-grid coordinating both heat and power. A dedicated device called SMART is incorporated within its power conversion and heat delivery systems, along with its autonomous load following capability make it ideal to operate together with other energy resources. The integration capability of the eVinci Micro Reactor to connect to a micro-grid can make users resilient by utilizing the diversity, high-reliability and safety of nuclear energy with other power generation resources.

6. Operational Performances

The eVinci Micro Reactor leverages its autonomous design to reduce the personnel required on site. It is planned that a few personnel would be required per shift for monitoring and security as the autonomous control system will limit the need for operators.

The eVinci design has the capability for load following and grid frequency control. The eVinci autonomous control system in combination with a micro grid, can perform load following for the most demanding situations in remote applications. Most of the load following demands can be executed utilizing the self-control capabilities, reactor control and power conversion control. Other high demand fluctuation will be supported by power resources coupled to the micro grid. The eVinci system has black start-up capability supported by batteries, for instances where the local micro grid cannot support the start-up process.

7. Instrumentation and Control Systems

The Autonomous Control System (ACS) is the primary I&C system of the eVinci Micro Reactor. The ACS facilitates autonomous control by utilizing fiber optic sensors for high fidelity temperature measurement of the reactor, neutron flux sensors for measurement of the reactor core operation, and load following logic programmed into the ACS. The ACS is programmed with functional logic for autonomous operation of the eVinci reactor. Temperature measurements from fiber optic sensors and neutron flux measurements from self-powered neutron flux detectors (SPNDs) determine trip decision and load following logic. The trip decision logic ensures the reactor does not transverse predefined operational limits. Load following logic autonomously adjusts control drum position based on load variances, reactor temperature and neutron flux measurements. The control drum interface logic provides a logical priority between the trip, load following logic and manual system controls. The ACS also collects environmental and operating data during normal operation, the autonomous logic functions control the reactor. The autonomous logic functions relinquish control of the reactor to a manual operator if required. Due to the autonomous control, no specific control room is required.

8. Plant Layout Arrangement

The eVinci Micro Reactor is designed for ease of logistics, with nearly no on-site construction and minimum on-site installation. The eVinci system is nearly 100% fabricated in the factory and packaged in shipping containers to be transported to site via air, water or land. After installation of the container at site, the only connection needed between the boxes are the electrical power conduits or pipe connections if combine heat and power is required. To ensure allowable dose levels are met, integrated shielding is designed for transportation. However, a shielding structure is planned for the reactor container to limit dose rate to operators and the public.

Designed for remote applications, the eVinci Micro Reactor can provide heat and power via a micro grid system. The system can be configured for all the voltage and frequencies that might be required at remote applications. If power demands require several eVinci systems, they can be staged together at a site; with each unit operating independently. Because the eVinci Micro Reactor is similar to a battery design power supply, it does not require on-site refuelling. This eliminates additional onsite facilities and structures that would typically be required for refuelling. After the end of its fuel cycle, the eVinci Micro Reactor will be transported back to the factory for refuelling or for long-term storage.

9. Design and Licensing Status

The eVinci development program has completed the conceptual design phase. Westinghouse drives the maturity of the eVinci Micro Reactor product through evaluating both Technology Readiness Level (TRL) and Manufacturing Readiness Level (MRL). The current TRL and MRL of the eVinci Micro Reactor technical areas are estimated to be at a level 5.

Westinghouse is currently working with both the U.S. Nuclear Regulatory Commission (NRC) and the Canadian Nuclear Safety Commission (CNSC) to license the design and technology. At this point, pre-application discussions and activities have highlighted the licensing efforts to date:

- Request to submit application for Vendor Design Review to Canadian Nuclear Safety Commission (CNSC)
- Submitted eVinci Micro Reactor Licensing Modernization Project (LMP) Demonstration “table top” report as part of a project to evaluate the LMP process. This report was provided to NRC for information.
- Provided the U.S. NRC a Regulatory Issue summary on process for scheduling and allocating resources for review of a new licensing applications.

10. Fuel Cycle Approach

The eVinci Micro Reactor core will use either the High-Assay Low Enriched Uranium (HALEU) – Uranium Oxycarbide (UCO) in a tristructural isotropic (TRISO) fuel or other fuel that is encapsulated.

After 3 years of full power operation without refuelling, the eVinci Micro Reactor will be disconnected and transported back to the factory in its original canister for either refuelling and redeployment or for long-term storage, which can be accomplished in the eVinci Micro Reactor canister it itself.

11. Development Milestones

2021	Electric demonstration
2024	First nuclear demonstration

ANNEX I
Summary of Global SMR Technology Development and Deployment

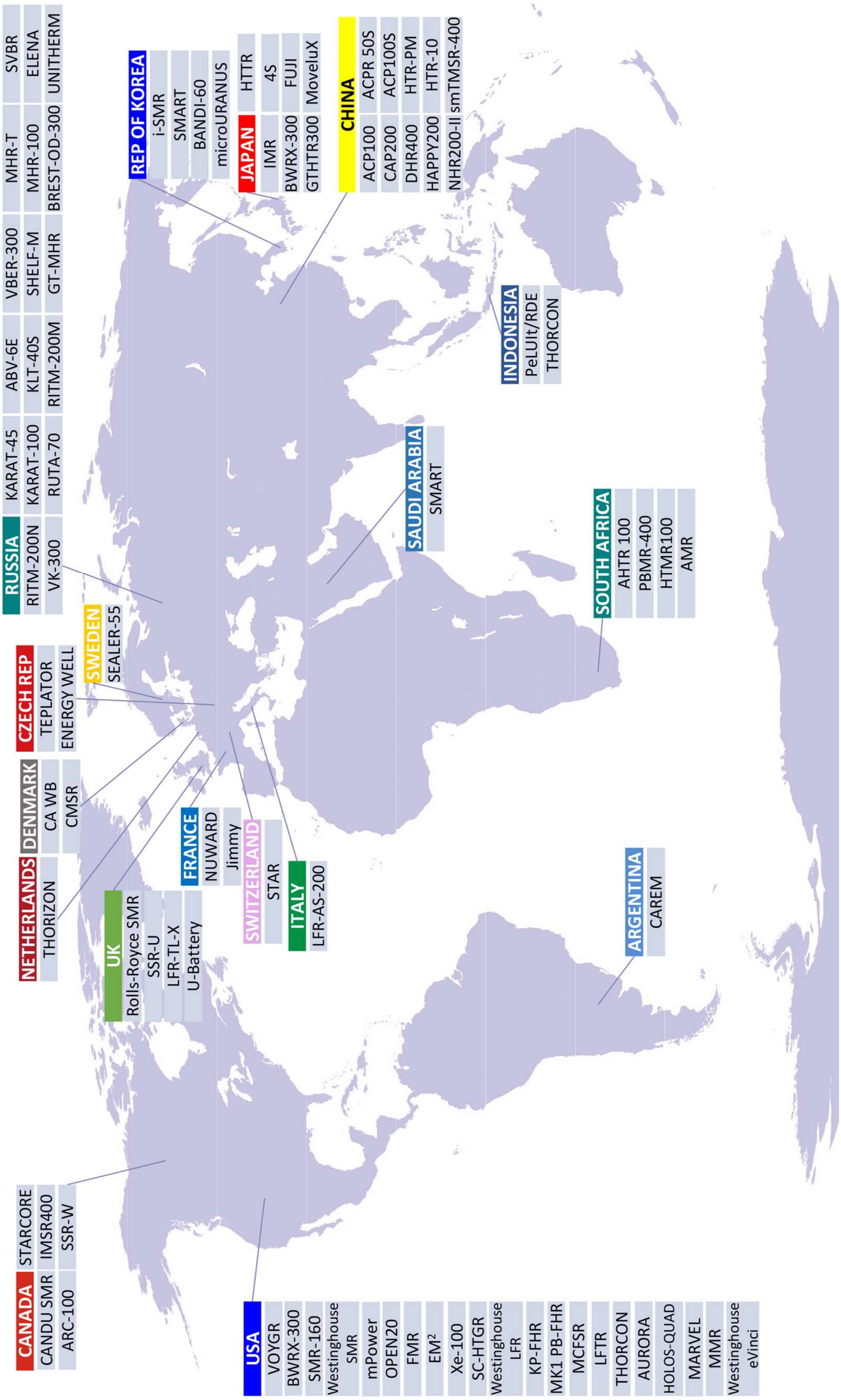


Figure I-1 Global Map of SMR Technology Development

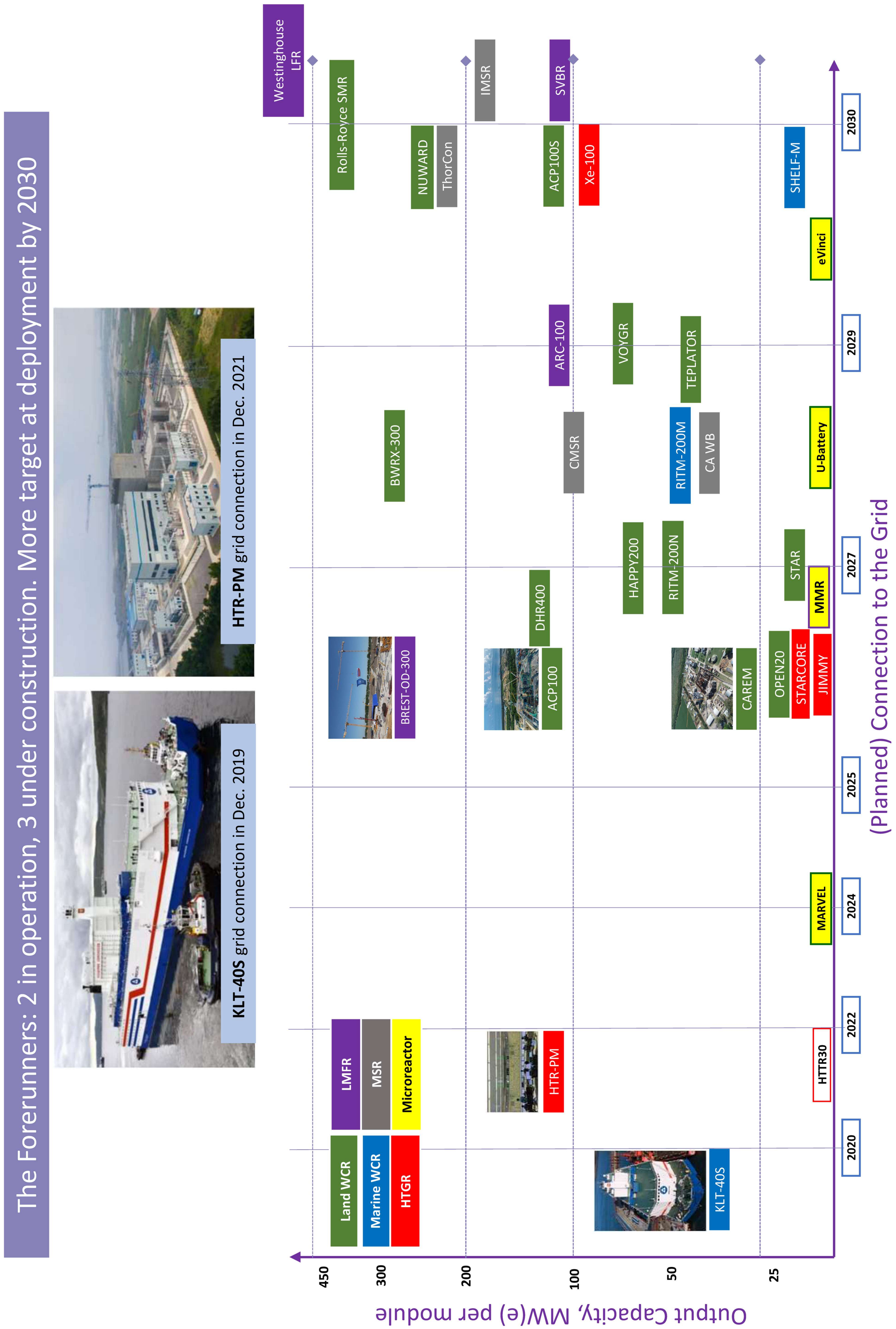


Figure I-2 General Timeline of Deployment as of 2020

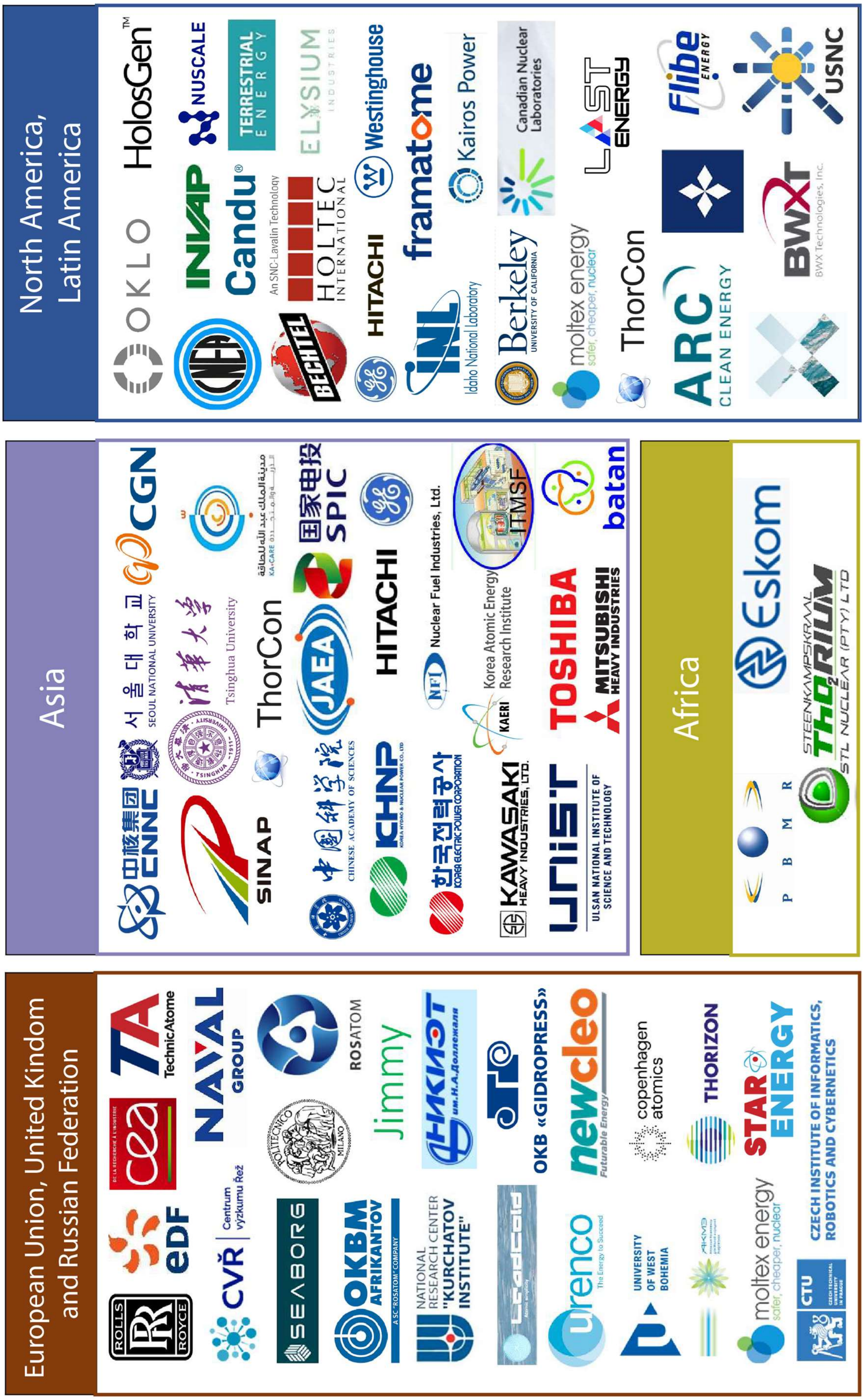


Figure I-3 Government and Private Sectors on SMR Technology Development

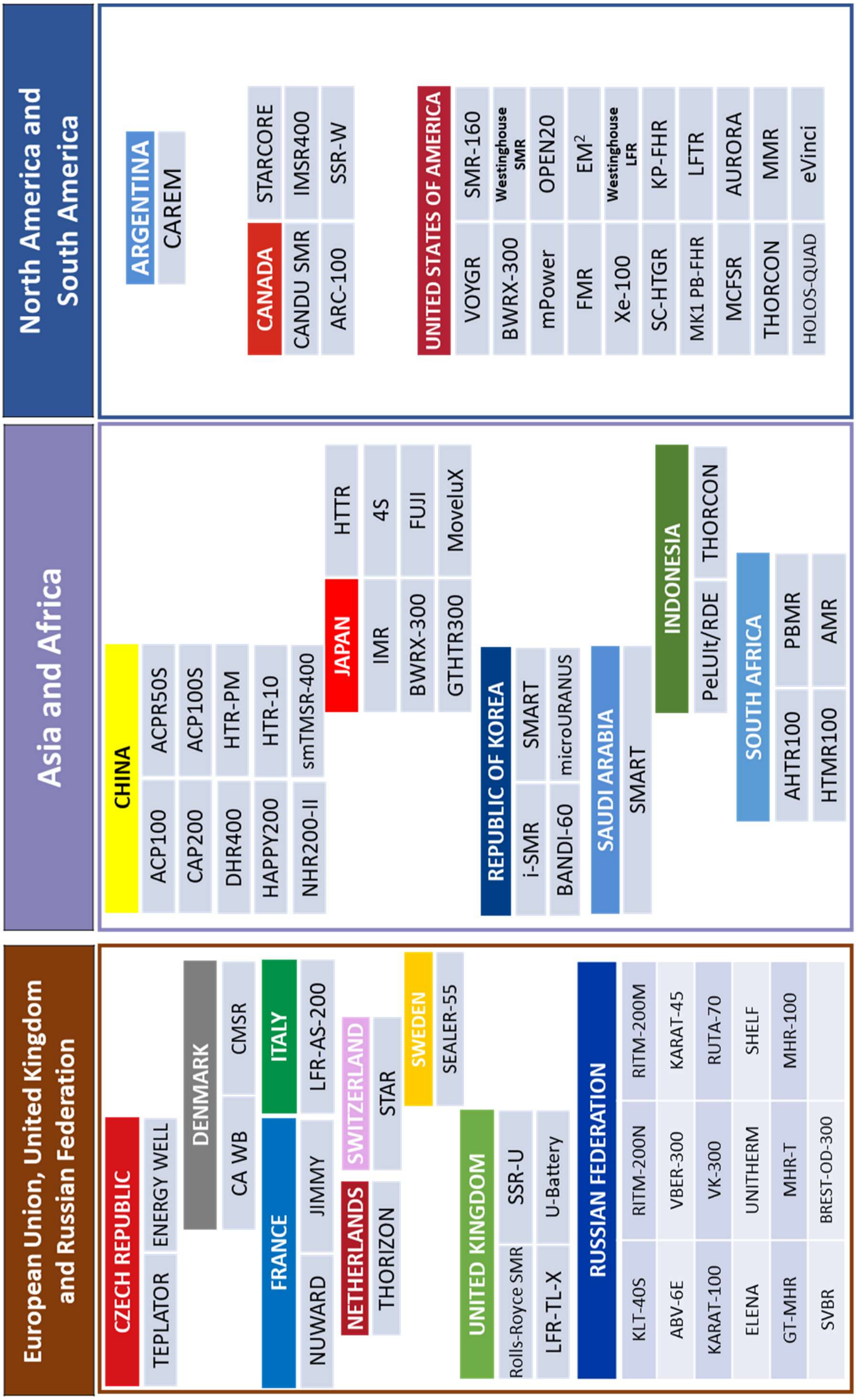


Figure I-4 SMR design and technology across the world's regions

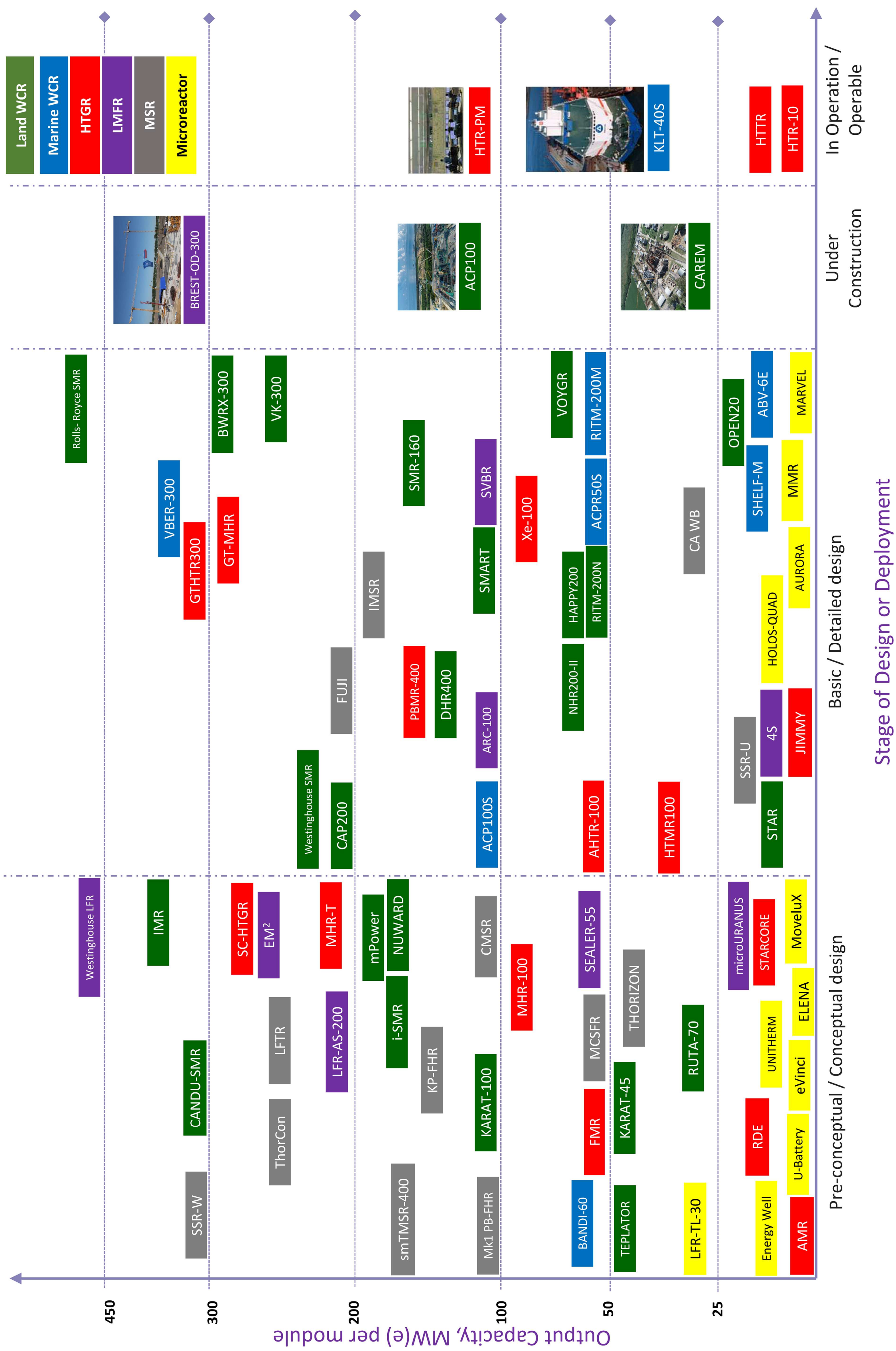


Figure I-5 Stage of Design or Deployment of SMRs in terms of their Output Capacity

ANNEX II
Power Range of SMR Designs

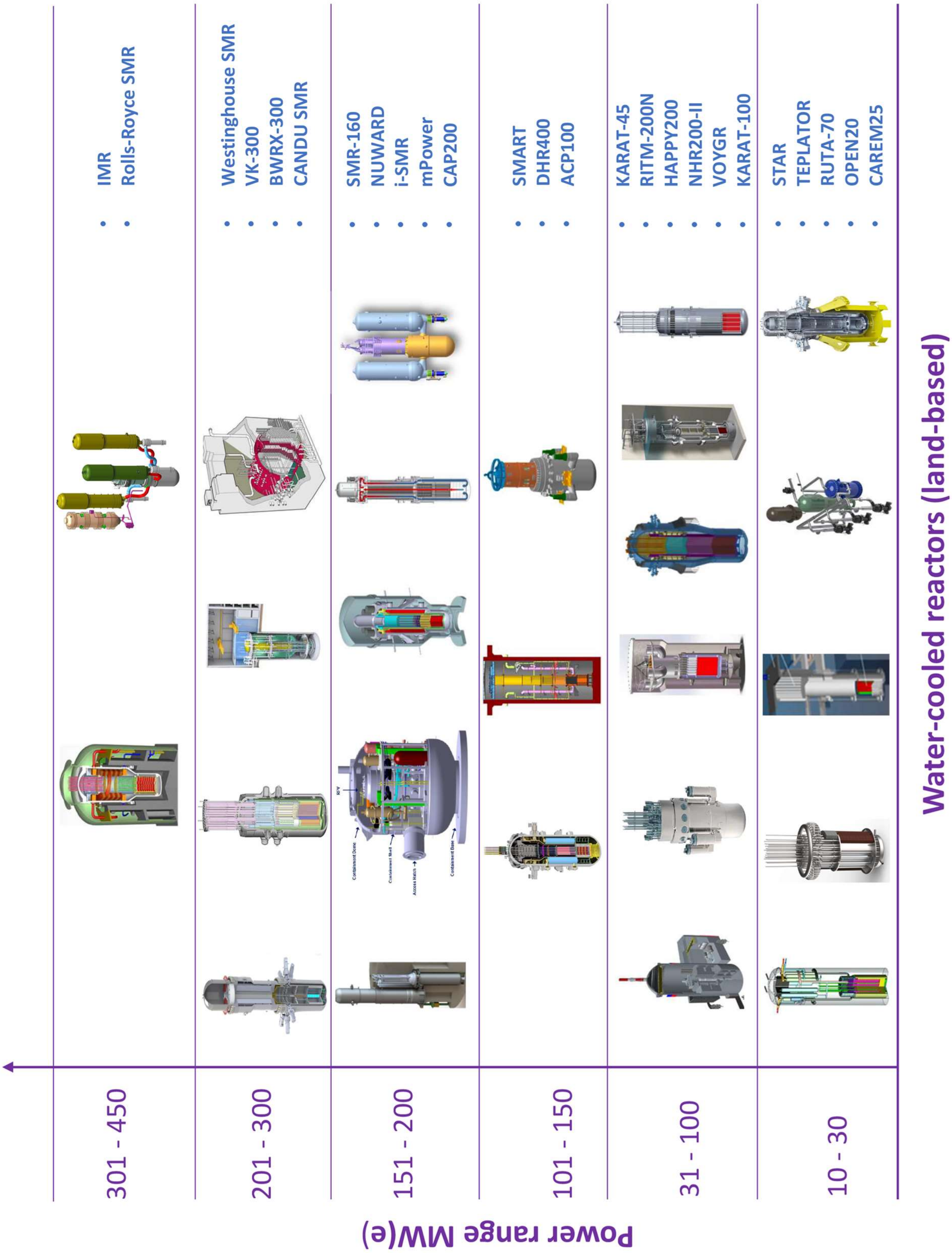


Figure II-1 Power Range of Land-based Water-cooled SMRs

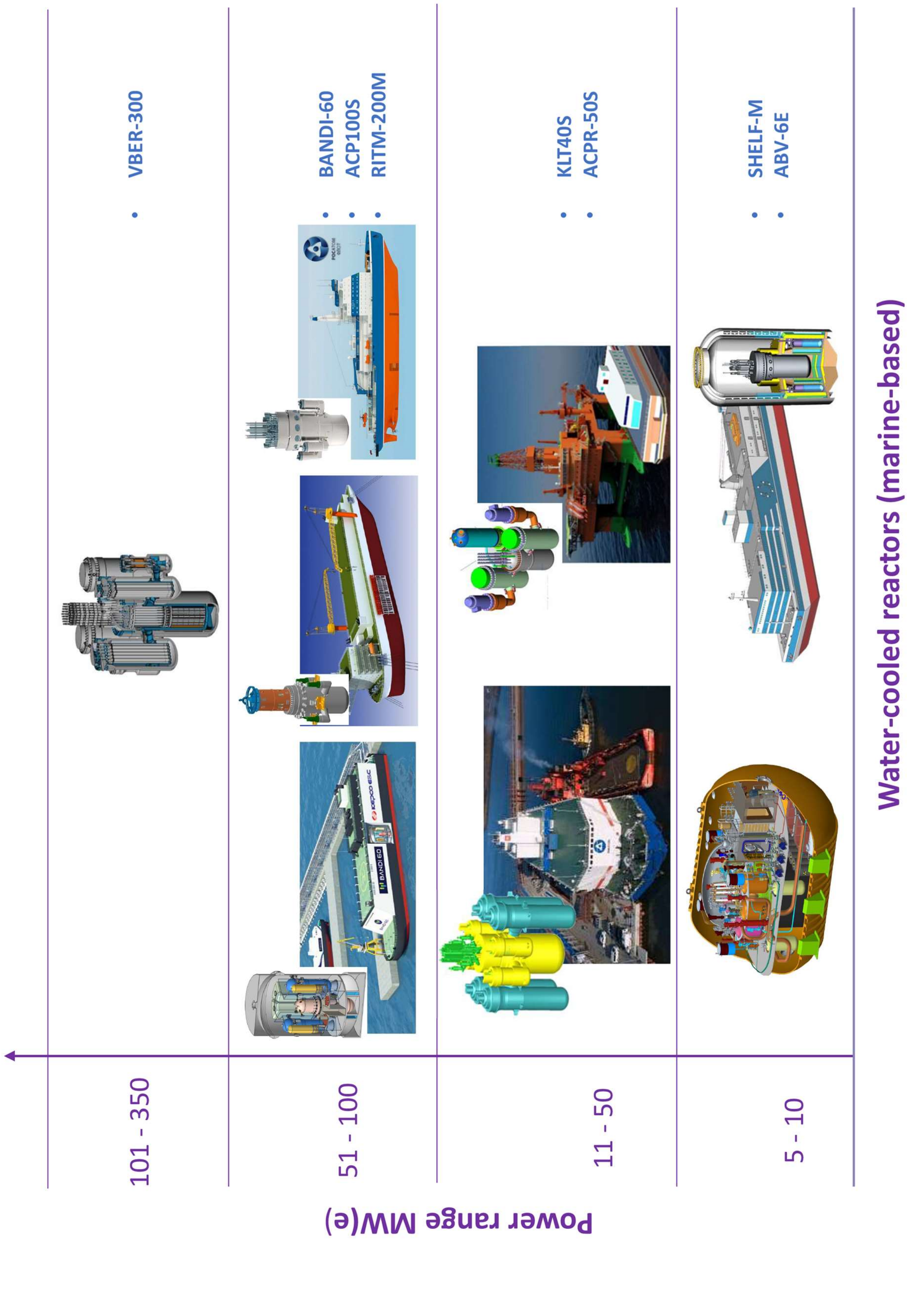


Figure II-2 Power Range of Marine-based Water-cooled SMRs

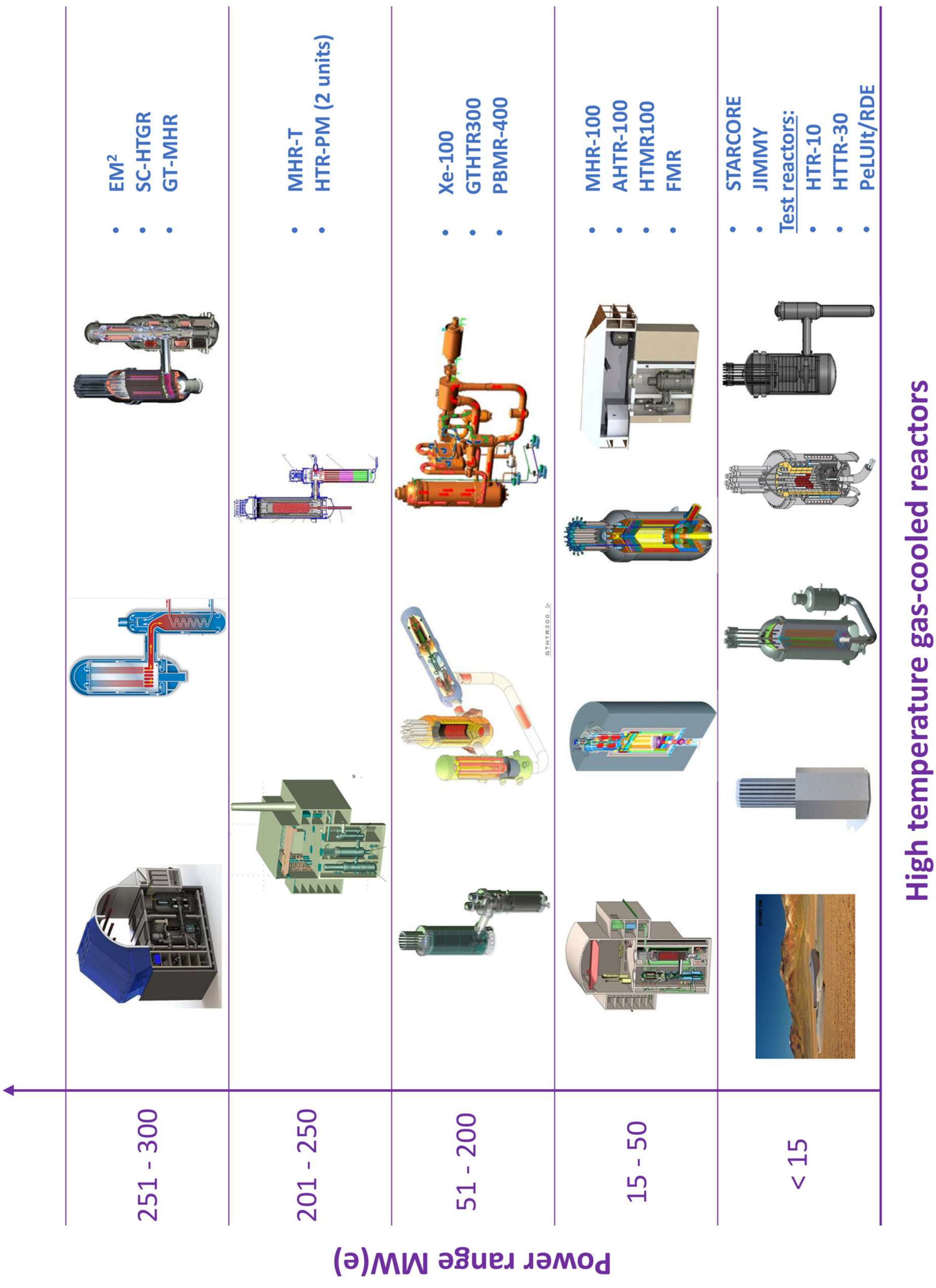


Figure II-3 Power Range of HTGR-type SMRs

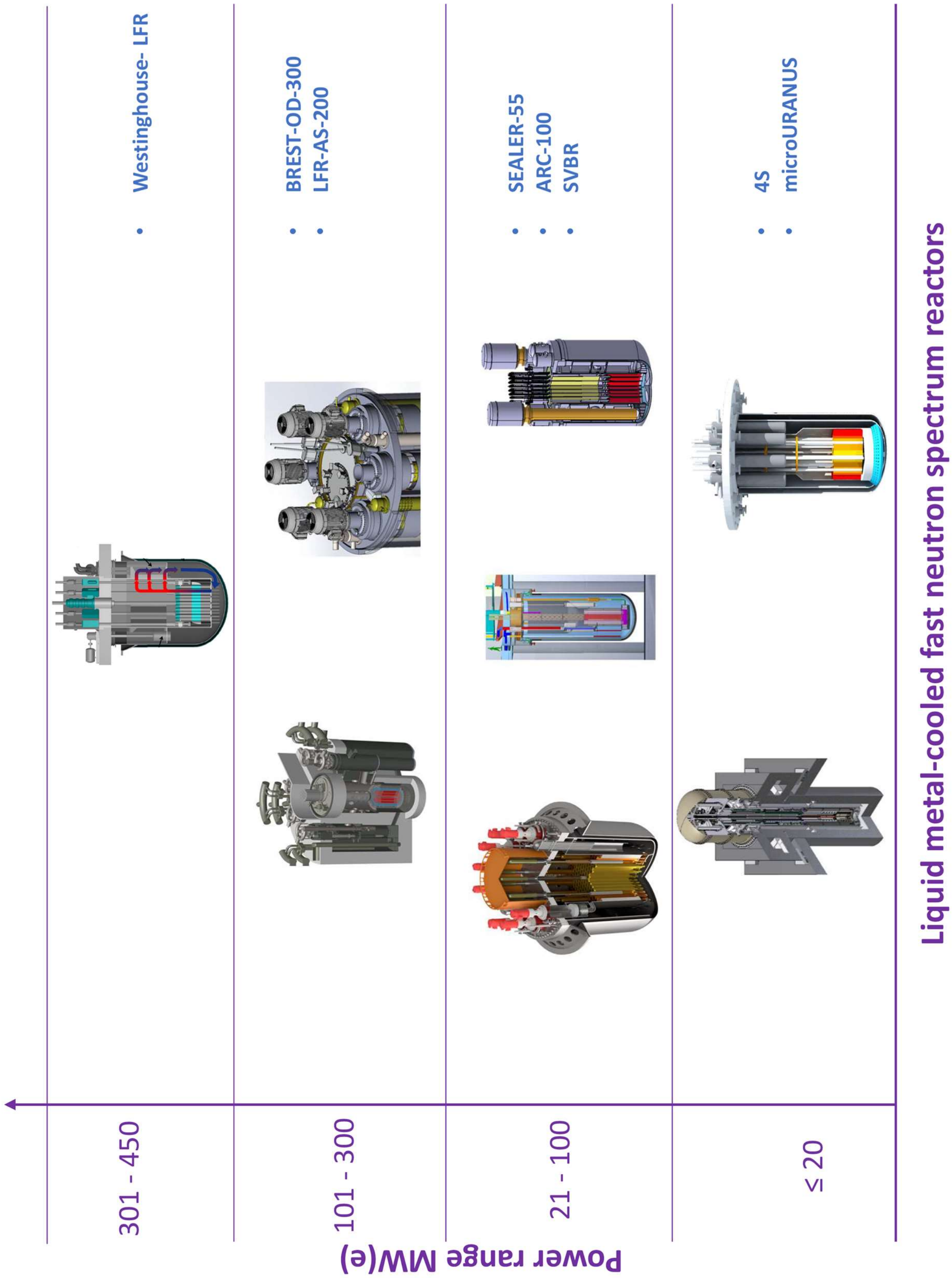


Figure II-4 Power Range of Liquid metal-cooled Fast Neutron Spectrum-SMRs

Figure II-5 Power Range of Molten Salt-SMRs

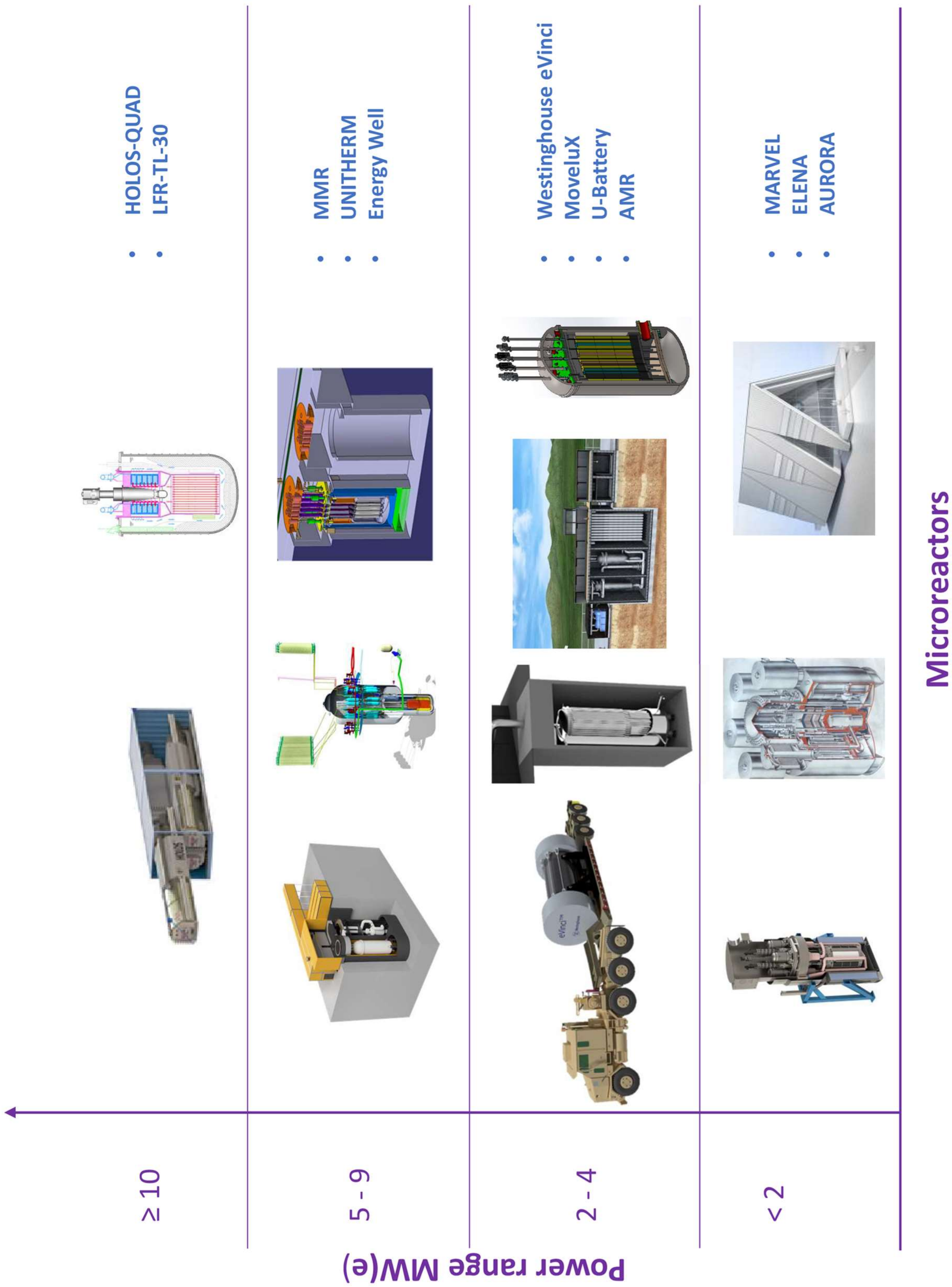


Figure II-6 Power Range of Microreactor

ANNEX III

Comparison of Main Technical Characteristics among several SMR Designs

Table III-1: Comparison of Main Characteristics among Land-based Water-cooled SMR Designs

	CAREM	ACP100	NUWARD	i-SMR	SMART	Rolls-Royce SMR	VOYGR	BWRX-300
Country of Origin	Argentina	China	France	Republic of Korea	Republic of Korea and Saudia Arabia	UK	USA	USA and Japan
Design organization(s)	CNEA	CNNC (NPIC)	EDF	KHNP&KAERI	KAERI, K.A.CARE	Rolls-Royce SMR Ltd.	NuScale Power Corporation	GE Hitachi & Hitachi GE Nuclear Energy
Reactor Type / Primary Circulation	Integral PWR / Natural Circulation	Integral PWR / Forced Circulation	Integral PWR / Forced Circulation	Integral PWR / Forced Circulation	Integral PWR / Forced Circulation	3-loop PWR / Forced Circulation	Integral PWR / Natural Circulation	BWR / Natural Circulation
Fuel type/assembly array	UO_2 pellet / hexagonal	UO_2 pellet / 17x17 square	UO_2 pellet / 17x17 square	UO_2 pellet / 17x17 square	UO_2 pellet / 17x17 square	UO_2 pellet / 17x17 square	UO_2 pellet / 17x17 square	UO_2 pellet / 10x10 array
Number of fuel assembly	61	57	76	69	57	121	37	240
Coolant	Light water	Light water	Light water	Light water	Light water	Light water	Light water	Light water
Moderator	Light water	Light water	Light water	Light water	Light water	Light water	Light water	Light water
Thermal output, MW(t)	100	385	2 x 540	540	365	1358	250	870
Electrical output, MW(e)	30	125	2 x 170	170	107	470	77 (gross)	270 - 290
Core inlet temp., ºC	284	286.5	280	295.5	296	295	249	270
Core outlet temp., ºC	326	319.5	307	320.9	322	325	316	288
Enrichment, %	3.1	< 4.95	< 5	< 5	< 5	< 4.95	≤ 4.95	3.81 (avg) / 4.95 (max)
Refuelling cycle, months	14	24	24 (half core)	24	30	18	18	12 - 24
Reactivity control	Control rods	Control rods + Gd_2O_3 solid burnable + soluble boron acid	Control rods + solid burnable	Control rod + burnable absorber rods + moderator temperature	Control rods + soluble boron	Control rods + moderator temperature	Control rods + soluble boron	Rods + solid burnable absorber (B_4C, Hf, Gd_2O_3)
Reactor Vessel's height/diameter, (m)	11 / 3.2	10 / 3.35	15 / 5	23 / 5	18.5 / 6.5	7.9 / 4.2	17.7 / 2.7	26 / 4
Design status	Under construction	Under construction	Conceptual Design	Conceptual Design	Detailed Design	Detailed Design	Equipment Manufacturing in Progress	Detailed Design

Table III-2 Comparison of Main Characteristics among Marine-Based Water-cooled SMR Designs

	KLT-40S	ACPR50S	RITM-200M	BANDI-60	ACP100S
Country of Origin	Russian Federation	China	Russian Federation	Republic of Korea	China
Design organization(s)	JSC “Afrikantov OKBM”,	CGNPC	JSC “Afrikantov OKBM”	KEPCO E&C	CNNC (NPIC)
Reactor Type	PWR	loop-type PWR	Integral PWR	PWR	Integral PWR
Fuel type/assembly array	UO_2 pellet in silumin matrix	UO_2 pellet / 17x17	UO_2 pellet / hexagonal	UO_2 pellet / 17x17	UO_2 pellet / 17x17
Number of fuel assembly	121	37	241	52	57
Coolant	Light water	Light water	Light water	Light water	Light water
Moderator	Light water	Light water	Light water	Light water	Light water
Thermal output, MW(t)	150	200	175	200	385
Electrical output, MW(e)	35	50	50	60	125
Core inlet temp., ºC	280	299.3	282	290	286.5
Core outlet temp., ºC	316	321.8	318	325	319.5
Enrichment, %	18.6	< 5	< 20	4.95	< 4.95
Refuelling cycle, months	-	30	Up to 120	48 - 60	24
Reactivity control	Control rods	Control rods + solid burnable poison	Control rods	Control rods	Control rods + Gd_2O_3 solid burnable poison + soluble boron acid
Reactor Vessel’s height/diameter, (m)	4.8 / 2.0	7.2 / 2.2	8.6 / 3.45	11.2 / 2.8	10 / 3.35
Design status	Connected to the grid in Pevek in December 2019. Entered full commercial operation in May 2020.	Detail Design	Basic Design Completed	Conceptual Design	Basic Design

Table III-3 Comparison of Main Characteristics among High Temperature Gas-cooled SMR Designs

	HTR-PM	GTHTR300	GT-MHR	HTMR100	Xe-100	SC-HTGR	EM^2
Country of Origin	China	Japan	Russian Federation	South Africa	USA	USA	USA
Design organization(s)	INET, Tsinghua University	JAEA	JSC “Afrikantov OKBM”	STL Nuclear (Pty) Ltd.	X-energy, LLC	Framatome Inc.	General Atomics
Reactor type	Modular pebble bed HTGR	Prismatic HTGR	Modular Helium Reactor	Pebble-bed HTGR	Modular HTGR	Prismatic HTGR	Modular high temperature gas-cooled fast reactor
Fuel materials	TRISO spherical elements with coated particle fuel	UO_2 TRISO ceramic coated particle	Coated particle fuel in compacts, hexagonal prism graphite blocks	TRISO particles in pebbles; LEU/Th	UCO TRISO/pebbles	UCO TRISO particle fuel in hexagonal graphite blocks	UC pellet / hexagon
Coolant	Helium	Helium	Helium	Helium	Helium	Helium	Helium
Moderator	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite	N/A
Thermal output, MW(t)	2 x 250	< 600	600	100	200	625	500
Electrical output, MW(e)	210	100 - 300	288	35	82.5	272	265
Core inlet temp., ºC	250	587 - 633	490	250	260	325	550
Core outlet temp., ºC	750	850 - 950	850	750	750	750	850
Enrichment, %	8.5	14	14-18% LEU or WPu	10%	15.5	14.5 (avg) 18.5 (max)	~14.5 (LEU)
Core Discharge Burnup (GWd/ton)	90	120	100-720 (depends on fuel type)	80 - 90	165	165	~130
Refuelling cycle, months	Online refuelling	48	25	Online fuel loading	Online fuel loading	½ core replaced every 18 months	360
Reactivity control	Control rods	Control rods	Control rods	Control rods in the reflector	Control rods	Control rods	Control rods
Reactor Vessel’s height/diameter, (m)	25 / 5.7 (inner)	23 / 8	29 / 8.2	15.7 / 5.6	16.4 / 4.88	24 / 8.5	12.5 / 4.6
Design status	In operation	Basic design	Preliminary Design completed	Basic Design	Basic Design	Preliminary Design	Conceptual design

Table III-4 Comparison of Main Characteristics among Liquid Metal-cooled Fast Neutron Spectrum SMR Designs

	BREST-OD-300	4S	SVBR	LFR-AS-200	ARC-100	Westinghouse LFR
Country of Origin	Russian Federation	Japan	Russian Federation	Italy	Canada	USA
Design organization(s)	NIKIET	Toshiba Energy Systems & Solutions Corporation	JSC AKME Engineering	*new*cleo srl	ARC Clean Energy	Westinghouse Electric Company, LLC
Reactor Type	LMFR	LMFR (pool type)	LMFR	LMFR (pool type)	LMFR (pool type)	LMFR (pool type)
Fuel type/assembly array	Mixed uranium plutonium nitride	U-Zr alloy	UO_2 / hexagonal	MOX / hexagonal	U-Zr alloy	Oxide, with provision for transition to Nitride
Number of fuel assembly	169	18	61	61	99	325
Coolant	Lead	Sodium	LBE	Lead	Sodium	Lead
Thermal output, MW(t)	700	30	280	480	286	950
Electrical output, MW(e)	300	10	100	200	100	~ 450 (Net Avg.)
Core inlet temp., °C	420	355	340	420	355	390
Core outlet temp., °C	535	510	485	530	510	650
Enrichment, %	up to 14.5	< 20	< 19.3	19% (avg) / 23,2% (max) in Pu	13.1 (avg)	≤ 19.75%
Core Discharge Burnup (GWd/ton)	61.45	34	60	100	77	Up to 100
Refuelling cycle, months	36 - 78	N/A	7 - 8 years	16	240	8 – 15 years (depending on fuel)
Reactivity control	Reactivity compensation, emergency protection, and automatic control members	Axially movable reflectors / fixed absorber	Control rods	Ex-core, reversed-flag type B4C rods, rotating B4C Rods	Control rods	Control and shutdown rods; thermal/pressure activated, nonrod-based passive shutdown
Reactor Vessel's height/diameter, (m)	17.5 / 26	24 / 3.5	8.2 / 4.53	6.2 / 6	16.7 / 7.9	Approx. 9.0 / 6.7
Design status	Under construction since July 2021	Detailed design	Detailed design for potential construction in 2025	Conceptual design	Conceptual design	Conceptual design

Table III-5 Comparison of Main Characteristics among Molten Salt SMR Designs

	IMSR400	CA Waste Burner	ThorCon	FUJI	SSR-U	KP-FHR	MCSFR
Country of Origin	Canada	Denmark	USA, Indonesia	Japan	UK	USA	USA
Design organization(s)	Terrestrial Energy, Inc.	Copenhagen Atomic	ThorCon International	International Thorium Molten-Salt Forum	Moltex Energy	Kairos Power	Elysium Industries
Reactor Type	MSR	MSR	Thermal MSR	MSR	MSR	Modular, pebble bed, high temperature, salt-cooled reactor	MSR – Fast Chloride
Fuel type/assembly array	Fluoride fuel salt	$^{7}LiF\text{-}ThF_4\text{-}(TRU)F_3$	UF_4	Molten salt with Th and U	Molten salt fuel within vented fuel tubes	TRISO / pebble bed	Molten Chloride Salt
Coolant	Fluoride salt	Fuel salt	Molten salts	Molten fluoride	Molten eutectic fluoride coolant salt	Li_2BeF_4 Fluoride salt	$NaCl\text{-}XCl_2\text{-}UCl_{3/4}\text{-}PuCl_3\text{-}FPCl_y$ fuel salt
Moderator	Graphite	Heavy water	Graphite	Graphite	Solid commercial grade graphite moderator	Graphite	-
Thermal output, MW(t)	2 x 440	0.05 - 0.25	557	450	40	320	125/500/1000/3000
Electrical output, MW(e)	2 x 195	0.1 - 0.25	250	200	16	140	50/200/400/1200
Core inlet temp., °C	620	600	560	565	725	550	650
Core outlet temp., °C	700	650 - 700	704	704	795	650	750
Enrichment, %	< 5	TRU or RGPu	2.3 (startup) 4.95 (makeup)	2.0 (0.24%U_{233} + 12.0%Th)	6	19.75	10% Pu fissile/(Pu+U total) or ~15% enriched HALEU
Refuelling cycle, months	84	N/A	12	Continuous	240	Online refuelling	Online refuelling
Reactivity control	negative temperature coefficient; online fuel addition	D_2O level adjustment	Negative temperature coef.	Control rod, or pump speed, or fuel concentration	Strong fuel temperature coef., liquid neutron absorber thermometer	Control rods, boron	Fuel expansion in/out of core; Fertile fuel addition; Passive fuel draining
Reactor Vessel's height/diameter, (m)	18.0 / 4.1	12 / 2.4	10.3 / 7.8	5.4 / 5.34	7 .5 / 6 (low pressure)	7.2 / 3.9	9.0 / 4.0
Design status	Detailed Design	Detailed design / Equipment manufacturing in progress	Basic design completed	Basic design completed	Basic Design	Conceptual Design	Conceptual Design

Table III-6 Comparison of Main Characteristics among Microreactor Designs

	eVinci™	MMR™	HOLOS-QUAD	MARVEL	MoveluX™	AMR
Country of Origin	USA	USA	USA	USA	Japan	South Africa
Design organization(s)	Westinghouse Electric Company LLC	Ultra Safe Nuclear Corporation	HolosGen	Idaho National Laboratory	Toshiba Corporation	STL Nuclear (Pty) Ltd
Reactor type	Heat pipe cooled Reactor	HTGR	HTGR	Liquid Metal Cooled Thermal reactor	Heat-Pipe cooled and calcium-hydride moderated reactor	HTGR (advanced prismatic)
Coolant	Heat pipes	Helium	Helium	NaK eutectic	Sodium heat-pipe cooled	Helium
Moderator	Metal hydride	Graphite	Graphite	Hydrogen in fuel	Calcium hydride (CaH_2)	Graphite
Thermal output, MW(t)	7 - 12	15 and 30	22	0.075 - 0.1	10	10
Electrical output, MW(e)	2 - 3.5	> 5 and > 10	10	0.015 - 0.027	3 - 4	3
Core inlet temp., ºC	N/A	300	590	480	680	450
Core outlet temp., ºC	N/A	630	855	548	685	750
Fuel type/assembly array	TRISO or another encapsulation	FCM / Hexagonal	TRISO-UCO / Hexagonal	Uranium Zirconium Hydride	Silicide (U_3Si_2) / Hexagonal	T RISO particles/ LBE eutectic/ SiC tubes
Enrichment, %	5 - 19.75	HALEU 19.75%	19.95	19.75	4.8 - 5.0	10 - 20
Refuelling cycle, months	> 36	240	96	Up to 20 years	Continuous	96
Reactivity control	Ex-core control drums	Control rod insertion, negative temperature coefficient	Redundant independent banks of Control drums & Shutdown rods	> 60	In-Ga Expansion Module	Control and shutdown rods in core, and the reflector
Reactor Vessel's height/diameter, (m)	N/A	13.25 / 3.5	5 (reactor fits 12.19 m long ISO container)	2.55 / 1.22	2.5 / 6.0	5 .96 / 2.78
Design status	Conceptual Design Completed	Basic Design	Detailed Design	Equipment manufacturing in progress	Conceptual Design	Pre-conceptual design

ANNEX IV

Non-Electric Applications using Small Modular Reactors

The interest in non-electric applications of nuclear power is growing, driven by a series of factors, including environment, economics, and security of energy supply.

There is a massive need for low carbon heat and low carbon fuels, as well as the technologies to produce them using nuclear power, such as SMRs, in order to support the decarbonization of the 'hard-to-abate sectors' i.e. those that cannot be electrified, such as metal refining, synthesis of chemical feedstock, cement and steel production, and heavy duty transport. In particular, nuclear power technologies can provide high-temperature heat needed by industrial processes. One of the low carbon options to be considered to achieve this is the capitalization of the nuclear heat provided by SMRs.

The emerging SMRs and their use for non-electric applications come with additional challenges, as their deployment requires a 'level playing field' in terms of policies, support to innovation, financing, as well as clear market signals to favour low carbon technologies and the associated licensing process.

SMR concepts being currently developed worldwide are of major reactor technologies that can support a broad range of applications in addition to electricity generation, including applications in the industry. SMRs bring the possibility and the opportunity to have their design optimized for non-electric applications. This involves single or multi-purpose use – generating commodities like hydrogen or potable water, providing heat for industrial applications and district heating, together with or without electricity generation for the grid.

Nuclear cogeneration is the generation of electricity and useful heat jointly that applies to nuclear power reactors in general, including SMRs. The followings are certain benefits that come from considering nuclear cogeneration:

- Enhanced efficiency: the efficiency of a nuclear power plant operating in cogeneration mode can exceed 80%, compared to 33% in the case of traditional large light water reactors operating just for electricity generation. It also offers additional uses of nuclear energy by capitalizing on heat for district heating, seawater desalination, hydrogen production or process heat for industrial applications.
- Optimum use of energy by avoiding unwanted transformations of energy.
- Improved flexibility, allowing switching between the two outputs – electricity and heat.
- Cogeneration gives nuclear power the flexibility to function in an energy system where a growing proportion of electricity comes from intermittent renewables, allowing plants to switch from electricity generation to other applications when demand is being met by wind, solar, or other energy sources.
- Reduced environmental impact: less heat waste is discharged in the environment; less water is needed for cooling; and using of nuclear heat can eliminate the need for fossil fuels for heat.

On the type of technologies, light water reactors are best suited to district heating and desalination, owing to their low working temperature range. While other technologies with higher-working temperatures, such as liquid metal cooled reactors (550°C), molten salt reactors (800°C), high-temperature gas-cooled reactors (850°C), and very-high temperature reactors (900°C to 1000°C) are better suited for industrial processes. Among the industries requiring high temperatures there are petroleum refining, coal gasification, steel making, and hydrogen production using high-temperature electrolysis, steam methane reforming or thermochemical processes. Furthermore, these nuclear reactors can be used for desalination and district heating when used as cogeneration systems. The figure below presents an overview of these applications together with illustrative SMR designs.

District heating

A heat distribution network would require steam or hot water of 80 - 150°C that can be supplied by a nuclear reactor, with a typical pipeline distribution range of 10 - 15 km. There are well proven projects for using nuclear combined heat and power for district heating using large scale nuclear reactors in several Member States. However, the market is still limited and there are deployment issues for this low share, such as the long distance between nuclear sites and urban areas, local governance, economic feasibility, institutional structures, and the difference of characteristics of national energy systems. These challenges associated with using large commercial reactors for district heating are now being addressed by SMR designs developed particularly for this purpose, for example in China and Finland.

Water desalination

Nuclear desalination has been demonstrated as being technically sound and economically affordable. There is accumulated experience worldwide in using large-scale nuclear reactors coupled to water desalination plants. Water desalination is an energy-intensive process. It can be achieved through thermal desalination processes - requiring heat to separate the distillate from high salinity water, reverse osmosis that uses membranes and high pressure to separate salts from the water, or hybrid processes, requiring both electricity and heat. The temperature range required is 70 - 110°C. Today, a series of factors, such as environment, energy reliability, energy supply security, the need of potable water and a significant growth in energy demand, are driving a high interest in the development and expansion of nuclear energy options for this application, where SMRs are showing high potential due to their versatility in use.

Industrial heat applications

Some of heat-intensive industrial processes can benefit from nuclear heat, including enhancing brown coal quality, coal liquefaction, coal gasification, and enhancing oil recovery. The main requirements are that the cogeneration nuclear plant has to be located close to the industrial user and to have a high reliability that nuclear can provide. This makes SMRs particularly attractive for these applications. The transport of the heat to the end user can become a challenge as the longer the distances the higher the associated costs with distribution and transport. This is the reason for which in the case of using the heat from nuclear reactors, the end user should be co-located or located close enough to the nuclear power plant. SMRs allow co-location with industrial end users due to the potential to adopt reduced emergency planning zone size, providing that there is acceptance of it by nuclear regulators. There has to be also a governance in place between industrial users and the utilities operating the nuclear reactor that supplies the process heat.

Hydrogen production

Nuclear energy can support the increasing needs for hydrogen production, through providing clean electricity, as well as heat. The IEA report on Nuclear Power and Secure Energy Transitions (International Energy Agency, Nuclear Power and Secure Energy Transitions. From today's challenges to tomorrow's clean energy systems, June 2022) released in June 2022 indicates that the economics for hydrogen production would be more favourable if the nuclear reactor is co-located with the hydrogen users, hence minimizing transportation costs. SMRs are well positioned to support production of hydrogen in the vicinity of end users.

While nuclear cogeneration and non-electric applications can be supported by the different reactor technologies of different sizes, their attractiveness and applicability may increase when it comes to SMRs, based on the following aspects:

– The nature of industrial heat market is highly fragmented, with the majority of potential heat need included in the 1 - 300 MW(t) range and 50% requiring less than 10 MW(t). This would bring an additional challenge for large scale nuclear reactors as the economics of using large capacities to capitalize on a small amount of heat needed is not favourable; instead, SMR designs are more suitable to support industrial heat applications.

– Moreover, nuclear heat used for industrial applications has to have a reliable and available source, and the source has to provide maximum availability. The SMR technologies are bringing the possibility of less frequent refuelling and can be designed from the beginning as multipurpose or to support solely producing a commodity (hydrogen or potable water).

– The SMR thermal ratings are about one-third or less of the large existing reactors. This helps lower investment requirement for the deployment and thus greatly improve the affordability to more countries, private investors, and industrial users. The modularity also makes them readily scalable with the scale and increment of energy demand by most industrial heat process plants.

– SMRs bring a relevant technology for load following and the overall power at the site level can be fractioned, facilitating an optimal harnessing of the energy produced and allowing it towards electricity production, conversion into commodities or storing for a later use. Nuclear power plants composed by multiple small modular reactors have as key advantage over a single equivalent large reactor (in terms of capacity) the possibility to operate some of the SMR fleet at full capacity while directing the thermal power of some modules for non-electric applications only.

– SMR designs allow for optimization of the use of reactors operating in cogeneration mode and generating multiproduct, the optimization being based on criteria such as cost of products, return of investment and energy efficiency.

– Coupling nuclear reactors with non-electric applications can also provide energy system storage – i.e., storing energy in the form of heat or as an energy vector such as hydrogen. This is the basic concept of hybrid energy systems.

SMRs are flexible systems that can be adapted for non-electric applications, according to their key operating parameters, i.e., exit working fluid temperature. They can be integrated effectively into the energy systems supporting communities from largest cities to most remote locations, bringing in the opportunity to support applications beyond solely electricity generation.

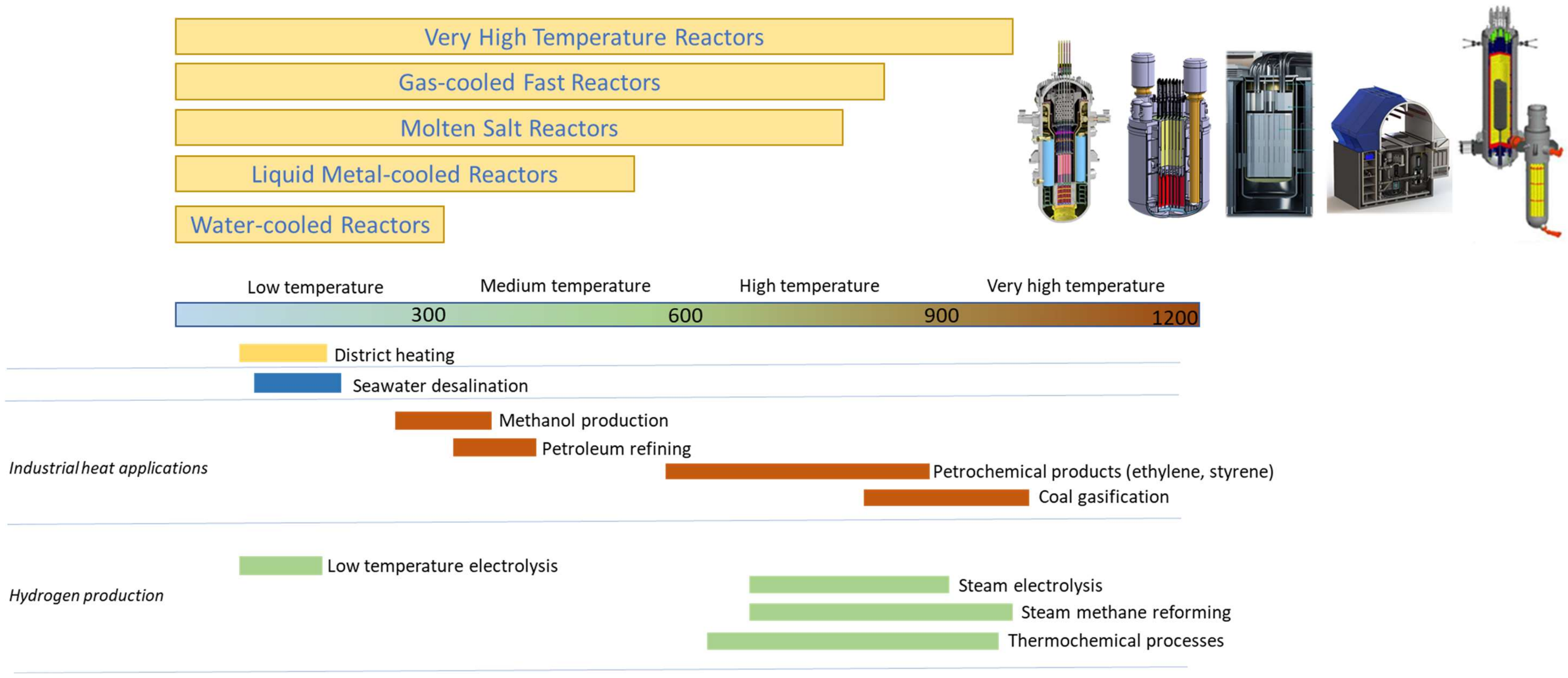


Figure IV-1 Exit Working Temperature of SMR Technologies and the Corresponding Non-Electric Applications

ANNEX V

Economic Challenges in Deployment of Small Modular Reactors

Introduction

SMRs have unique design, safety and economic features that make them attractive to potential project developers and end-users but also investors, governments, and communities worldwide. The technology can be applied to produce low-carbon electricity, heat (district heating, for example), hydrogen and desalinated water, in addition to grid, load-following and ancillary services. More than 80 SMR designs and concepts are currently under development and have varying degrees of readiness levels. For each of these projects, development costs need to be understood, as well as construction and operation expenses, which still need to be appropriately estimated, analysed and optimised. Specific revenue models are also needed for demonstrating the business case and secure access to funding, financing, and low cost of capital for the promoters of the technology. Finally, the macroeconomic impact associated with SMR design development, manufacturing, construction and operation (including periodical maintenance) has to be quantified and communicated to gain the support of the government and society at large.

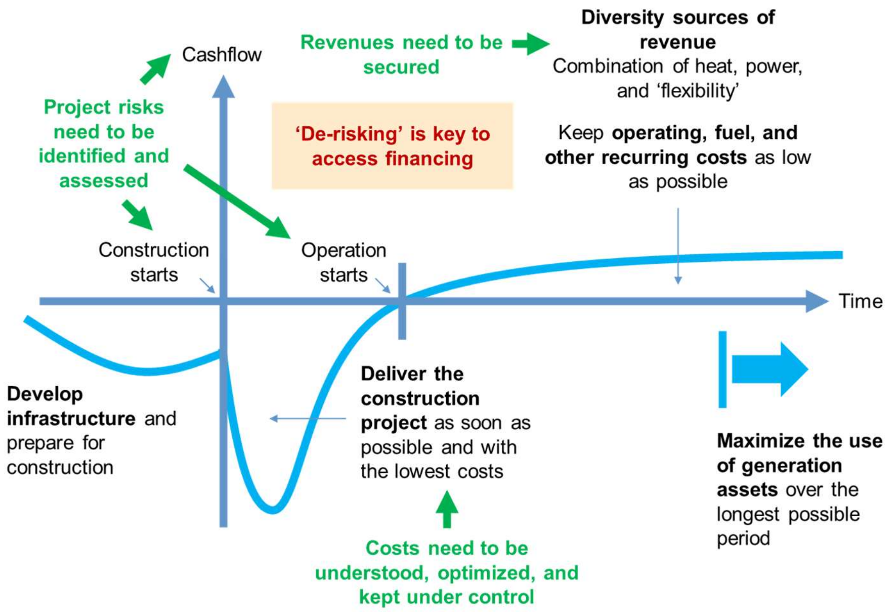


Figure V-1: Economic challenges facing SMR

This annexe provides an overview of the economic challenges facing SMR during the key phases of a typical project (**Figure V-1**) – from the project development phase to construction and commissioning to the operation phase – with a focus on costs and cost drivers, funding and financing issues, and economic impact.

Costs and cost drivers

Power generation systems, including SMR, are usually characterised by the metric of Levelized Cost of Energy (LCOE). LCOE represents the per-MWh cost (in discounted real dollars) of building and operating a generating plant over an assumed economic lifetime. The key economic indicator is determined by setting to zero the net present value (NPV) of cash in- (from the sale of electricity and, possibly, other products and services) and out-flows (CAPEX and OPEX) incurred over the key phases of a power generation project (**Figure V-2**).

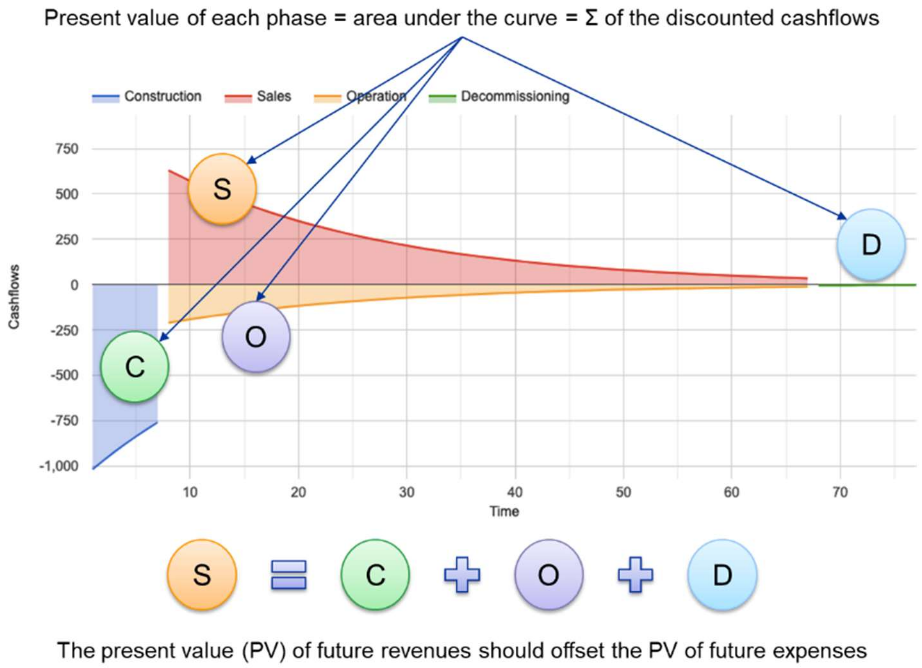


Figure V-2: LCOE estimation basis

Constructing NPPs usually involves high capital costs with a long construction period, followed by an operating phase with relatively low and stable recurring expenses, translating into an LCOE primarily driven by capital expenditures and the discount rate [1].

Another key driver is the amount of generated power, which is directly linked to the capacity factor. By producing multiple by-products and services (a combination of electricity, heat and flexibility), SMRs can maximize the utilization of the generation asset over its lifetime and secure earnings, including when operating in a load-following mode.

SMRs are based on the repeated manufacture of a modular and standard design of components and systems in a factory setting, potentially leading to a decrease in capital expenditures, a shortening of the construction and commissioning period, and a low cost of capital. The in-factory fabrication model is widely applied in the airline and shipbuilding industries and is a synonym for learning, cost reduction and improved quality [2]. To replicate this model, SMR developers need to embrace manufacturing best practices from other industries while sustaining demand for standardised SMR components and systems to enable learning and diffusion of know-how and achieve significant cost reductions. They also need to deal with a regulator focusing primarily on safety, and not necessarily on the speed of delivery [3].

A study commissioned by the Department of Energy and Climate Change of the UK suggests that SMRs could achieve LCOE parity with large units at 5 GW of cumulative deployment, assuming a production rate of 10 SMR units per year [4].

Funding and financing issues

Nuclear power projects have particular characteristics and risk profiles, which makes it difficult for their promoters to secure financing and benefit from a low cost of capital:

- Implementing new nuclear power plant projects involves a long, complex and context-dependent regulatory process.
- Nuclear projects are also subject to country-specific risks, including political risks, a major concern for investors and lenders that are not easily assessed or mitigated.

SMRs share many of the attributes of GW-scale reactors, including exposure to nuclear- and country-specific risks (**Figure V-3**), making them less attractive to risk-averse debt and equity providers than renewable generators. However, SMRs are smaller, less complex, easier and faster to build and commission, which could translate into improved access to financing. Furthermore, the modular character associated with the technology brings some flexibility, e.g., in deciding the timing and the magnitude of an increment in the installed capacity, which can make the technology more appealing to investors. Finally, SMRs have the potential to produce a combination of electricity, heat and flexibility, diversifying its sources of income and mitigating market risks, which can also improve its attractiveness.

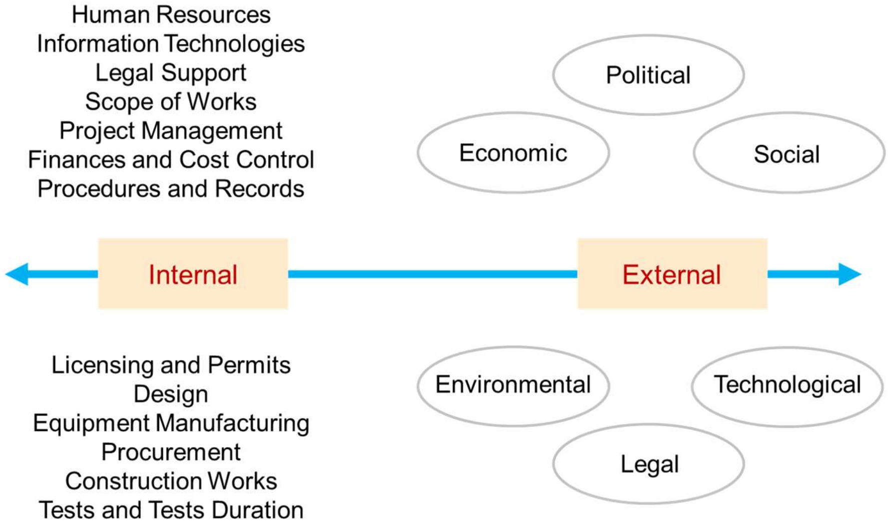


Figure V-3: Risks: Internal vs external

Enabling the production and commercialisation of SMRs will require developing and pursuing de-risking strategies to channel funding to the project and keep costs under control while securing revenues for nuclear power plant operators. Another key enabler is a policy framework recognising the value nuclear can bring to an evolving energy system, low-carbon, highly coupled, and probably renewables-dominated. Governments have a crucial role in creating such an enabling environment and overcoming obstacles to introducing SMRs and other low-emitting energy sources.

As for large units, SMRs can be funded by the State (public financing). Alternatively, a utility can build, own and operate the infrastructure, borrowing funds, raising equity capital, or simply relying on its own balance sheet (corporate financing). Project financing (recourse financing) can also be an option on the condition that the government accepts to cover nuclear- and country-specific risks. Finally, projects conducted overseas can benefit from the involvement of Export Credit Agencies (ECAs), which tend to increase the lenders' confidence.

Revenues can be secured through various models, e.g., the Contract for Difference (CfD) [5] and Regulated Asset Base (RAB) [6] models in the UK, depending on the market's value rewarding mechanisms, i.e., how electricity, heat and ancillary services are sold and priced. Today's markets primarily reward output, i.e., the MWh of energy produced, which can be traded on the futures or spot markets. The provision of capacity is also rewarded through the procurement (by the transmission system operator) of balancing services but to a much lesser extent.

Economic impacts

Nuclear power projects are often described as labour intensive, paying higher wages compared to other technologies, and having long-lasting economic impacts through direct, indirect and induced effects (**Figure V-4**), which can be estimated and quantified [7]. This impact was historically significant in countries with a thriving nuclear industry and a high degree of localisation, as emphasised by a joint report of the NEA and the IAEA that studied the employment effects of nuclear energy programs [8] and a recent study on "multipliers" of the International Monetary Fund (IMF) [9]. As for the large reactors built in the 1970s, 1980s and 1990s, SMR projects are expected to stimulate manufacturing and engineering services in all countries participating to the supply chain, as well as construction, operation and maintenance activities in the end-user country, impacting the labour market and generating economic growth across a wide range of economic sectors beyond these perimeters.

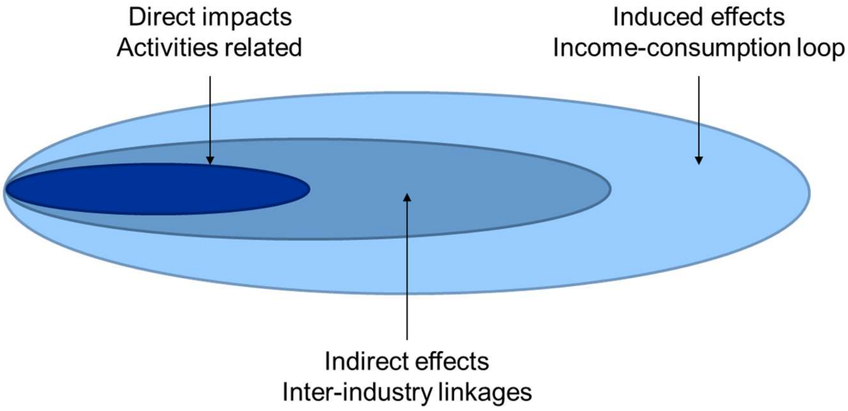


Figure V-4: Macroeconomic impacts: direct, indirect and induced effects

SMR projects can also potentially impact countries adding SMRs to their energy mixes, even in the absence of manufacturing and construction activities. By providing electricity at an affordable - and predictable - price, SMRs can, in fact, trigger induced effects within the income-consumption loop (**Figure V-4**), resulting in benefits to a variety of actors in the economy.

Table V-1 provides an example of an economic impact assessment associated with developing and deploying five grid-scale 300 MWe SMR projects in Canada [10]. In addition to job creation figures, the study, prepared at the request of the provinces of New Brunswick, Ontario, and Saskatchewan, reports a positive impact on GDP of 17 billion dollars; and an increase in government revenues amounting to 5.4 billion dollars.

Phase	Employment
Project dev.	1,528
Construction	12,455
Operation	1,469
Decommissioning	1,193

Table V-1: Direct, indirect, and induced employment on an average annual basis [10]

ANNEX VI
Fuel Cycle Approach adopted in SMRs

Introduction

Design and operation of SMRs are building on experience gained from current power and research reactors in assessing and planning for spent fuel management, waste management, and decommissioning of SMRs. For SMRs to be sustainable, it is necessary to plan for decommissioning and management of spent fuel and waste early in their development; this ensures that the right infrastructure exists for waste and spent fuel management, facilitates confidence of interested parties, and avoids unwanted liabilities and legacies for future generations.

To the extent that certain SMR designs concepts, associated fuel cycle decisions and future operational procedures, are similar to those used in current operating large NPPs, the spent fuel management (SFM) and radioactive waste management (RWM) activities can most probably be implemented consistently. Advanced SMR designs will generate entirely new wastes streams and new spent fuels which must be handled safely. Early planning will help avoid any unforeseen technological, environmental, or financial difficulties.

A better understanding of the specific fuel cycle approach being proposed by the various SMR designs is very important to stakeholders in Member States. Particularly, the knowledge of the fuel cycle will facilitate discussion on the front-end requirements and fuel supply, the potential management routes for spent fuel and waste forms (discussed in the annex VII), the potential needs for a final waste repository, and Safeguards-by-Design implementation by SMR designers. It should be recognised that fuel cycle choices are often made by the state and this may place restrictions on vendors that wish to deploy an SMR in a country.

There are many designs being advanced from major technology lines to meet the development objectives of SMRs. SMRs are not only of water-cooled reactor types but include many designs utilising other coolant types or neutron spectra, including HTGRs, LMFRs and MSRs represented in this booklet. Many of these designs adopt an innovative approach for their fuel cycles. Most of the SMR designs propose longer operation cycles between refuelling. This might raise issues on fuel (and cladding where applicable) performance (e.g. at higher burnup), and other aspects such a core maintainability, on-line access to service key components and the different steps for managing the spent fuel (mainly storage, transport and recycling routes). Some of the new fuel cycles and new fuels proposed include the use of $Th/^{233}U$, reprocessed uranium (RepU), MOX, transuranic (TRU) fuels through the reprocessing of spent LWR fuel. To achieve that it is necessary to include advanced technologies as pyro-processing for SMR fuel reprocessing and recycling. It is important to highlight that some of those innovative routes are not industrially implemented yet, a few are at demonstration stage and most of them are at R&D stage, with a low technology readiness level (TRL).

Land-based water-cooled SMRs

For water-cooled SMRs, various fuel cycles are adopted depending on the design approach and objectives. Land-based water-cooled SMRs typically utilise existing LWR fuel design with ^{235}U enrichment below 5% to achieve a range of discharged burnups as low as 30 GWd/tHM for district heating reactors, and between 2.3 up to 162.4 GWd/tHM for electricity generation, to attain refuelling cycles of 18, 24, 36 or even up to 96 months. The ultimate objective is to have an optimized capacity factor of higher than 90% while minimizing the operation and maintenance (O&M) costs. Longer fuel cycles improve the overall plant availability but could be at the expense of a somewhat reduced discharge burnup.

Typical PWR-type SMR use fuel with enrichment lower than 4.95%, propose a refuelling cycle of 18-24 months and replace half of the assemblies every cycle to optimize the overall fuel economics while maximizing the discharge burnup. A 3-batch refuelling design, on the other hand, improves the discharge burnup and fuel utilization. The target refuelling outage is between 18 to 36 days. A natural circulation integral PWR design for a multi-unit plant, for example, adopts a 24-month refuelling cycle and a 3-batch in-out refuelling scheme. This means that during the refuelling process, one-third of the fuel assemblies are removed from a reactor module and placed in the spent fuel pool and one-third of fresh assemblies are introduced; while at the same time the remaining assemblies are re-shuffled in the core.

The selected option for the standard design of CANDU SMR is an open fuel cycle based on natural uranium, producing a very low residual uranium spent fuel with low heat generation. It is also designed to be able to burn RepU from LWRs, making it a valuable addition to any light-water fleet. The use of MOX and Thorium fuels is also possible with proper customization.

There are also converted PWR type SMR designs originally deployed for marine-based SMRs. These designs have enrichment close to the 20% upper limit of LEU to ensure long time operation in remote areas while still satisfying international proliferation goals. The target refuelling cycle is as long as 72 – 84 months. In the framework of transportable SMR designs, a factory-fabricated reactor vessel could be delivered to the site loaded with fresh fuel. This initial load is designed to provide the whole NPP lifetime without refuelling.

This booklet also reports BWR-type SMRs with natural circulation. During the refuelling outage 15% to 25% of the bundles in the core are replaced by fresh fuel with the result that they stay in the core for several cycles, until they are discharged. When removed from the core, spent fuel is stored in the fuel pool inside the reactor building for 6 to 8 years then it is transferred to storage casks that are stored in a facility located outside the reactor building, waiting to be disposed of as typically, for the standard BWR designs an open cycle is used for managing the spent fuel.

In general, the water-cooled SMRs provide a spent fuel pool using storage racks with storage capacity for 6 to 10 years after operation . A longer storage option of up to 20 – 30 years could be proposed for some designs, depending on owner's requirements. The spent fuel pools are connected to the ultimate heat sink through dedicated engineered safety features. The spent fuel storage racks include not only sufficient storage for many years of operation, but also to store potential defective fuel assemblies and for non-fuel core components such as control rod assemblies.

Ultimately, the fuel cycle can be tailored to customers' requirements in compliance with the regulatory requirements both in the countries of origin and host countries.

Marine-based water-cooled SMRs

The first SMR design that completed construction, was connected to the grid and is now in commercial operation is the PWR-type KLT-40S installed in the Akademik Lomonosov floating power unit (FPU), deployed in Pevek, Russian Federation. The purpose of the FPU is to supply cogeneration of electricity and process heat in difficult-to-access remote regions. Therefore, the adopted fuel cycle has to be commensurate to the purpose of deployment. In Russia, there are at least five (5) marine-based SMR designs (included in this booklet) and their variants. To achieve a long refuelling cycle of 30 – 36 months, a near 19% fuel enrichment is required and 45.4 GWd/ton core discharge burnup is achieved in KLT-40S. Refuelling is performed 14 days after reactor shutdown when the levels of residual heat releases from spent fuel assemblies have come down to a required level. No special maintenance or refuelling ships are necessary. A single batch fuel loading is done to attain a maximum operation period between refuelling. In other words, fresh fuel is loaded in all the core positions replacing the burned fuel assemblies.

The RITM-200M based optimized floating power unit (OFPU) is also designed as a transportable SMR power plant that has a refuelling cycle up to 120 months. The OFPU is delivered to the site with fresh fuel in its reactors. After completion of the fuel cycle, the OFPU returns to the country of origin together with the spent fuel in its reactor. All operations for production, post-reactor maintenance and reprocessing of spent nuclear fuel are performed in the country of origin.

HTGR-type SMRs

The objective of HTGR-type SMR designs is built on inherent safety characteristics with the focus on accident-tolerant fuel (ATF). The HTGR-type SMRs come as pebble-bed or prismatic core designs adopting tristructural isotropic (TRISO) coated particle fuel or as designs with hexagonal assemblies adopting UC or UO_2 fuel. The SMR of this type in operation is in China and was connected to the grid in December 2021. It is a 2-unit pebble bed type HTR-PM that can generate 210 MW(e). Online refuelling (pebble bed) or long refuelling cycles (prismatic) of 25 to 60 months or even no refuelling (no shuffling) are adopted by near-term deployable HTGR-type SMRs with the ^{235}U enrichment ranging from 8.5% to 19.75%. The typical discharge burnup of the HTGR-type SMRs is substantially higher than that of typical water-cooled SMRs. It ranges between 60 to 165 GWd/tHM depending on fuel type and fuel cycle. The higher enrichment is needed for the increased burnup target but also to compensate for the typically larger neutron leakage from this graphite moderated and reflected core with its low-power density but also relatively small active core diameters (needed to limit fuel accident temperatures). Although their diameters are physically larger than typical LWR SMRs, the cores are neutronically much smaller leading to the aforementioned higher net leakage.

The EM^2 high-temperature gas-cooled fast reactor uses an open fuel cycle with LEU/DU vented fuels that can exceed 30 years operation without refuelling or shuffling. This should lead to reduced cost and decreased proliferation risk, while achieving high fuel utilisation with low mass of waste streams. The core is capable of burning used LWR, plutonium, and thorium fuels. The spent fuels are stored in storage facility at the reactor site. After cooling, the spent fuel will be directly disposed of or recycled. When the closed fuel cycle scheme is applied, the spent fuel would undergo pyroprocessing for metallic fuels and would be refabricated as fresh fuel for reuse. There is no reprocessing plant based on pyro technology under operation so far. This technology is still under demonstration stage with low TRLs and technical issues to be solved as materials corrosion, etc.

Both open and close fuel cycles can be adopted with options including Th/U, Th/Pu, Pu, and actinide fuel forms . Most of these fuels have been developed, manufactured and demonstrated (or at least tested) in the past HTGR projects. In HTR-PM, for example, if recycling is adopted, the spent fuel spheres would be dismantled, and the nuclear fuel would be reprocessed (in normal reprocessing facilities, that may have to be adjusted to receive the material). Currently there are limited routes available or decided to manage graphite wastes.

Most HTGR-type SMRs will start with open cycle option and no reprocessing will be done. The generated spent fuel shows some of the favourable proliferation resistance characteristics of these reactors, one being that the total plutonium and ^{239}Pu assay (per unit of energy produced) is much lower due to better in-situ utilisation. In the case of pebble bed reactors, fuel that has not reached the target discharge burnup is recycled again through the reactor and remains in the closed safeguarded fuel handling system until it is classified as spent fuel; and then loaded into a spent fuel cask.

With higher thermal efficiency and high fuel burnup, HTGR-type SMRs support sustainability for open fuel cycles. Most of the core designs are also compatible with various more advanced fuel cycles employing fertile / fissile material conversion and recycle including Th/U, Th/Pu, Pu, and actinide fuel forms. The TRISO coated particle fuel could also be recycled when and if such process is mandated and economically viable, although the reprocessing routes are still at R&D and demonstration stages.

Liquid metal-cooled fast neutron spectrum SMRs

This booklet reports eight liquid-metal cooled fast neutron spectrum SMRs. In general, compared to water-cooled SMRs, LMFR-type SMRs have a higher enrichment of ranging from 14% to slightly less than 20% ^{235}U (some fuels also with plutonium loadings) with core discharge burnup in the range of 60 to 100 GWd/tHM to realize longer fuel cycles of up to 30 years.

Within this category, BREST-OD-300 is a lead-cooled fast neutron reactor facility that has started construction in June 2021 in Seversk, Russia with completion schedule by 2026. The project is to enable a closed nuclear fuel cycle (CNFC) for full utilisation of the energy potential of natural uranium. Mixed nitride fuel with high density and thermal conductivity ensures optimum performance of fuel in the core (core reproduction ratio ~ 1.05) to compensate for burnup. The fuel type considered for the first core and the first partial fuel reloads of the BREST-OD-300 fast reactor is nitride of depleted uranium mixed with plutonium, whose composition corresponds to that of irradiated spent fuel from VVER's following reprocessing and subsequent cooling for about 25 years. After completion of the initial stage the reactor will operate a closed nuclear fuel cycle. For the production of new fuel, it uses the products from the reprocessing of its own spent fuel.

The LFR-AS-200, with MOX fuel of 19% avg / 23.2% max in Pu and burnup of 100 GWd/ton, can be nearly self-sufficient in Pu with a breeding ration of ~ 0.9. With some modifications, it can also be transformed into Pu burner with a breeding ration of ~ 0.5.

The Westinghouse Lead Fast Reactor, with fuel enrichment up to 19.75% and core discharge burnup up to 100 GWd/ton, applies once-through fuel cycle with single-batch long-life core design. However close fuel cycle is a possible option.

Another LMFR-type SMR design – SEALER-55, with fuel enrichment of 12%, core discharge burnup of 60 GWd/ton and fuel cycle approach to be decided, adopts uranium nitrid fuel which wille be fabricated by direct ammonolysis of 12% enriched UF_6, followed by spark plasma sintering of pellets.

Molten salt SMRs

Molten salt reactors (MSRs) have different characteristics from that of solid-fuelled reactors. In MSRs, as no fuel structure or cladding is required, the fuel is not subject to failures due to high burnup or mechanical damage. The fuel is already in a molten state so there is no risk of fuel melting (with the severe consequences that may challenge the vessel integrity and possible fission products release). Molten-salt reactors reported in this booklet adopts different fuel cycles that aim for longer fuel cycle up to 150 months, online refuelling (adding of fuel containing molten salt) and continuous operation. Enrichment levels vary from those designs with less than 5% enrichment to some with a higher-level enrichment up to 19.9%.

In the IMSR400, which plans to complete vendor design review (VDR) Phase 2 in 2022 in Canada, the core-unit will be replaced at the end of its 7-year fuel cycle. The de-fuelled core-unit is then moved by overhead crane from its operating silo to a long-term storage silo inside the reactor auxiliary building. A third core-unit can then subsequently be installed in its place ready to begin operations when needed (or once the second operating core-unit reaches its end-of-cycle and is shut down).

The Copenhagen Atomics Waste Burner uses LiF-ThF_4-(TRU)F_3 or LiF-ThF_4-RGPuF_3 kickstarter fuel salt and LiF-ThF_4 blanket salt, where the ^{233}U production benefits from the high number of excess neutrons from the plutonium fissioning. As the thorium fuel cycle converges towards equilibrium the breeding process benefits from ^{233}U superior neutron economy.

Another MSR design is ThorCon. It adopts enriched uranium of 2.3% as startup and eventually 4.95% after makeup, with discharge burnup of up to 145.8 GWd/tHM. However, once HALEU becomes available in power plant level quantity ThorCon with 500 MWe can operate as a thorium converter. The ThorCon core will require 5.3 kg of 19.7% enriched uranium and 9.0 kg of thorium per day to be added to the core to realize the 8-year core life. During the 8-year fuel cycle, a portion of the fertile thorium is converted to fissile ^{233}U which then becomes part of the fuel.

A potential issue on the fuel cycle of MSR is the management of the spent molten salt. The complex mixture of fuel and molten salt which will be difficult to manage as a high-level waste. The potential reprocessing routes require pyro technologies and there is no pyro-reprocessing plant under operation so far.

Microreactors

The microreactors reported in this booklet are of different technology lines: pressurized water reactor, heat pipe-cooled reactor, high temperature gas-cooled reactor, molten salt reactor, and liquid metal-cooled fast neutron spectrum reactor. The range of electrical output is from dozens of kilowatts up to 30 MW(e). The adopted fuel enrichment is between 4.8% up to 19.95%, with continuous online refuelling or long refuelling cycle between 36 months up to 20 years. Limited information of the fuel cycle aspects of microreactors are available.

The Westinghouse eVinci uses heat pipes for heat transfer and HALEU Uranium Oxycarbide (UCO) in TRISO encapsulated fuel as one of the options that can be used. After 3 years' operation without refuelling, the microreactor will be disconnected and transported back to the factory in its original canister for either refuelling and redeployment or for storage waiting for a disposal route.

HOLOS -QUAD also uses TRISO-UCO fuel. Its spent fuel cartridges passively dissipate decay heat from reactor shutdown, and can be placed within dual purpose transport and storage shielding dry casks within 72 hours from shutdown. The spent fuel cartridge dimensions altogether with those of the dry cask are designed to comply with ISO shipping container dimensional constraints. The spent fuel cartridges will be stored at long-term storage facilities and the spent fuel in them will not be processed as the type of fuel utilized is already in a stable form for long-term storage.

Another product called the Micro Modular Reactor (MMR) uses Fully Ceramic Micro Encapsulated (FCM™) fuel which is based on very small particles containing low-enriched uranium. The TRISO fuel, which contains the radioactive by-products of fission within layered ceramic coatings, is further encased within a fully dense silicon carbide matrix. 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. FCM fuel ensures containment of radioactivity during operations and accident conditions, which means that no fission products are released from the fuel. The MMR is initially fuelled for a 20-year cycle, and may be refuelled for an additional 20-year cycle. There is no long-term fuel storage on site.

Table VI-1 Fuel Cycle Approach adopted by SMR Designs

Fuel Cycle Requirements / Approach	SMR designs by type of coolants and technology characteristics				
	Water-cooled Reactors	HTGRs	Liquid-metal cooled Fast Reactors	Molten Salt Reactors	Microreactors
Open Fuel Cycle	CAREM, CANDU SMR™, HAPPY200, i-SMR, SMART, VK-300, RUTA-70, STAR, Rolls-Royce SMR, VOYGR™, BWRX-300, Westinghouse SMR, mPower, BANDI-60, SHELF-M	HTR-PM (currently), STARCORE, GTHTR300, GT-MHR, MHR-T, MHR-100, AHTR-100, HTMR100, EM^2, FMR, Xe-100, SC-HTGR, PeLUIt, HTR-10, HTTR	ARC-100, 4S (applicable), SVBR (on the first stage), SEALER-55 (optional), Westinghouse Lead Fast Reactor,	IMSR400, SSR-W, SSR-U, KP-FHR, Mk1 PB-FHR	Energy Well, MoveluX, HOLOS-QUAD
Closed Fuel Cycle	TEPLATOR™, NUWARD, IMR, RITM-200N, KARAT-45, KARAT-100, RITM-200M	HTR-PM (future), JIMMY	BREST-300-OD, ARC-100 (to be considered when reprocessing is licensed and accepted), 4S (applicable), MicroURANUS, LFR-AS-200, SVBR (in more distant future), SEALER-55 (optional), Westinghouse Lead Fast Reactor (possible)	smTMSR-400, CMSR (offsite reprocessing), FUJI, THORIZON, MCSFR, LFTR	MoveluX (optional), LFR-TL-30
Longer Refuelling Cycle (≥ 24 months)	ACP100, CAP200, NHR200-II, NUWARD, IMR, i-SMR, SMART, RITM-200N, VK-300, KARAT-45, KARAT-100, RUTA-70, STAR, SMR-160, OPEN20, KLT-40S, ACPR50S, ACP100S, BANDI-60, ABV-6E, RITM-200M, VBER-300, SHELF-M	STARCORE, GTHTR300, GT-MHR, FMR	BREST-300-OD, ARC-100, SVBR, SEALER-55, Westinghouse Lead Fast Reactor	IMSR400, smTMSR-400, THORIZON, SSR-U	Energy Well, AMR
Enrichment < 5%	CAREM, ACP100, CANDU SMR™, CAP200, DHR400, HAPPY200, NHR200-II, TEPLATOR™, NUWARD, IMR, SMART, VK-300, KARAT-45, KARAT-100, RUTA-70, Rolls-Royce SMR, VOYGR™, BWRX-300, SMR-160, Westinghouse SMR, mPower, OPEN20, ACPR50S, ACP100S, BANDI-60			IMSR400, SSR-W, Copenhagen Atomics Waste Burner (optional), ThorCon	MoveluX

5% ≤ Enrichment ≤ 20%	RITM-200N, STAR, KLT-40S, ABV-6E, RITM-200M, SHELF-M	HTR-PM, STARCORE, JIMMY, GTHTR300, GT-MHR, MHR-T, AHTR-100, PBMR®-400, HTMR100, EM², FMR, Xe-100, SC-HTGR, PeLUIt, HTR-10, HTTR	BREST-300-OD, ARC-100, 4S, MicroURANUS, SVBR, SEALER-55, Westinghouse Lead Fast Reactor	smTMSR-400, CMSR, FUJI, SSR-U, KP-FHR, Mk1 PB-FHR, MCSFR, ThorCon (possible)	Energy Well, ELENA, UNITHERM, AMR, LFR-TL-30, U-Battery, HOLOS-QUAD, MARVEL
Enrichment > 20%	KLT-40S, RITM, SHELF, ABV-6M, STAR		LFR-AS-200 (19% avg / 23.2% max)		
Use of Thorium-cycle and/or Disposition of Plutonium	CANDU SMR™ (possible with suitable customization)	STARCORE (plan to investigate), GTHTR300 (applicable), MHR-100 (possible), AHTR-100 (flexible), PBMR®-400 (flexible), HTMR100 (optional), EM² (capable)	LFR-AS-200 (capable), SVBR	smTMSR-400, Copenhagen Atomics Waste Burner (possible), SSR-U, Mk1 PB-FHR (possible), MCSFR, LFTR	AMR (to be investigated), LFR-TL-30
Use of Spent Fuel as Fuel	TEPLATOR™	GTHTR300 (applicable), EM² (feasible)	BREST-300-OD (DU mixed with plutonium), MicroURANUS, SVBR	SSR-W, smTMSR-400, Copenhagen Atomics Waste Burner, FUJI, MCSFR	

ANNEX VII
Spent Fuel, Waste Management and Disposal Plans adopted for SMRs

Introduction

Design and operation of SMRs are building on experience gained from current power and research reactors in assessing and planning for spent fuel management, waste management, and decommissioning of SMRs. For SMRs to be sustainable, it is necessary to plan for decommissioning and management of spent fuel and waste early in their development; this ensures the right infrastructure exists for spent fuel and waste management, facilitates confidence of interested parties, and avoid unwanted liabilities and legacies for future generations.

There are many different technologies being advanced and promoted throughout the world to fill the SMR demand. The adopted approaches depend on the particular SMR design, which may give rise to spent nuclear fuel (SNF) and radioactive waste (RW) either with well-known properties or, on the contrary, to SNF and RW for which new processing steps need to be developed. The approach to managing these will depend in part on a country's already existing spent fuel and radioactive waste management plans and practices and may need to be adjusted in case additional processing steps are needed.

One central lesson learned from prior deployment of innovative nuclear technologies remains the need to fully understand the back-end liabilities that may arise, and to have credible management paths available – otherwise the promise of sustainable deployment cannot be upheld.

Of the 83 SMR designs reported in this booklet, many of them are still in the conceptual design phase. Given the early stage of development, the key question is how far can SMR designs ensure that SNF and RW arising throughout the lifecycle of the plant – construction, operation and decommissioning – can be kept to a minimum; and can be managed as much as possible with tried and proven technologies. This is particularly relevant for designs which may generate new spent fuels and radioactive waste streams or for which disposition paths may not be readily available.

The existing national policy and strategy of a country are also significant in determining spent fuel (SFM) and radioactive waste management (RWM) approaches. Establishing responsibilities and funding mechanisms, as well as setting out technological and programmatic decision making on SFM and RWM are all essential considerations for the developers of SMRs. This becomes most relevant for countries which have no prior experience with nuclear power generation and had no need to have a strategy for SFM and RWM.

The safe, secure and sustainable management of spent fuel and radioactive waste arising from the operations of nuclear power reactors, as well as from associated nuclear fuel cycle activities and ultimately from the decommissioning of these nuclear power reactors is key to the future of nuclear energy. The nuclear fuel cycle approach (addressed in the previous annex) is a major factor influencing the type of spent fuel and waste and associated properties that need to be managed; and that in addition to the specific options for reactor design and operations and ultimate decommissioning of the SMR, also influence what type of waste and waste forms are generated (i.e., waste is not limited to decision of direct disposal or reprocessing).

Technical solutions already exist ranging from reprocessing and recycling, to conditioning of the types of spent fuel currently in use, to the knowhow how to dispose spent fuel and high-level waste in deep underground repositories. Partitioning and transmutation are advanced processes which have the potential to further reduce the impact of nuclear waste to be disposed of in future. A number of facilities in operation are effectively managing and disposing of low-level waste (LLW) and low- and intermediate-level short-lived waste (LILW-SL).

The emergence of SMRs presents an opportunity to demonstrate a new paradigm, in which effective SFM and RWM should be considered and incorporated early in the conceptual design stage. Solutions for managing spent fuel and radioactive waste arising from SMRs will be one of the important factors for Member States to consider when selecting a technology. This approach will help address uncertainties related to the back end of the fuel cycle, reduce costs and enhance public acceptance of nuclear power. The experience built up over decades, from many types of reactors (experimental, research and power), provides a most valuable resource for considering the SFM and RWM aspects of SMRs. The advances in computer codes, particularly as applied in the simulation and modelling of a wide range of phenomena in the design, performance and long-term conditions of new reactors systems, is a major enabling factor in assessing SFM and RWM during the design stage. However new fuel materials are planned for some SMR designs which will require validation of computer codes to prove their suitability. This requires the availability of experimental data of the new fuel materials.

Some SMR designs have features that could reduce the tasks associated with spent fuel management. These designs adopt longer refuelling cycles of 3 to 7 years, in comparison to 1 to 2 years for conventional power plants. Some designs are even intended to operate for up to 30 years without refuelling. However, even in such cases, there will be some spent fuel left, which will have to be properly managed. For most of the innovative non-water cooled SMRs, the designers propose to either use the existing infrastructure or to adapt it for the new spent fuel and radioactive waste stream. Countries with established nuclear power programmes have been managing their spent fuel for decades. They have gained extensive experience and have proper policies, strategies and infrastructure in place that might enable the management of spent fuel arising from SMRs based on technologies of the current fleet. Since specific plans and limits on a final repository may already be in place it may impose restrictions and a barrier to some of the new fuel cycle options and its specific waste forms.

Land-based water-cooled SMRs

Land-based water-cooled SMR developers in general adopt SFM and RWM plans similar to that of operating advanced water-cooled reactors. Facilities for waste treatment and storage are provided. For instance, in SMART, that is planned for deployment in Saudi Arabia, the liquid radioactive waste will be processed through demineralizer package to minimize the shipment volume of solid waste; while the gaseous radwaste system performs holdup and release in a controlled manner. In water-cooled SMRs, operation without soluble boron in the primary coolant allows a significant reduction in the environmental discharges and concomitant simplification of the waste treatment systems.

Advances are also being made in the dry storage technologies. Holtec International, the developer of the SMR-160, has developed a Multi-Purpose Canister called MPC-37 which among other inventories could also be used to store the spent fuel coming from the SMR-160 in future. This canister is designed for the HI-STORM UMAX modules, an underground vertical storage cask design.

For decommissioning, SMR design organizations also plan systematic dose reduction to personnel in decontamination and decommissioning. In integral PWRs, the longer distances from the fuel, and the additional material and components between the active core and the RPV, can decrease the fast neutron fluence on the RPV by a significant factor compared to a loop-type PWR. This essentially eliminates vessel embrittlement, the need for surveillance coupons for periodic in-service inspection of the RPV. It will also reduce the activation of the steel components significantly.

Marine-based water-cooled SMRs

In the case of Russian's floating nuclear power plants (FNPP), onsite refuelling is not required when deployed in a host state. The spent fuel is initially stored on board the FNPP and then will be taken back to the supplier for processing in the Russian Federation. In the RITM-200M, 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.

In the case of ACPR50S, liquid radwaste system is designed to prevent or minimize the creation of radioactive liquid effluent, and this achieved, wherever possible, by internal recycling. Gaseous radwaste system is designed to minimize the radioactivity associated with the resulting environment discharge. Solid radwaste system is designed to collect, preliminarily treat and temporarily store the solid wastes during the operation. And the wastes will be sent to disposal site for final disposal.

In the case of BANDI-60, a floating power plant, the spent fuels are transported from and to an inland fuel fabrication and storage facility using a dedicated fuel transport ship. The back-end fuel cycle option for spent fuel is dependent on the plant owner's policy and requirements.

HTGR-type SMRs

Due to higher thermal efficiencies and increased burnup, HTGRs produce about 40% less high-level waste per unit of energy produced, including significantly less plutonium, especially just one quarter of the ^{239}Pu content compared to a single-pass typical LWR cycle (mono-recycling). The storage and disposal requirements largely depend on the volume, activity and fission product decay heat, that could be up to 50 times lower (per volume) for HTGRs as compared to LWRs. In this form, proponents are exploring options different from conventional packaging due to the lower specific source term and heat production generation) and with radioactivity already contained in coated particles with a stable silicon carbide layer (that will last for more than a million years).

In comparison, the HTGR spent fuel volumes are much larger requiring much more space. If disposed of as part of a larger waste programme (designed for LWRs), the disposal of HTGR spent fuel spheres or elements may therefore take up a relative larger part of the facility than expected. For example, the fuel volumetric content of uranium in a pebble fuel sphere is significantly less than 1%. So even though spent fuel spheres may provide a stable multi-barrier containment with very low heat generation, volume reduction may be attractive. Separation of the coated particles from the graphite matrix (or fuel block) then become a more attractive option since the coated particles present excellent radioactivity containment characteristics also for long term storage or disposal. Reprocessing is also an option. Although many studies have also been performed to dispose of, condition or reprocess the large volumes of graphite from past gas-cooled reactor projects, none has been implemented on commercial scale and direct disposal may be the preferred option.

In general, HTGR-type SMRs have a dedicated waste handling system at the reactor site to store low- and medium-level liquid and solid radioactive waste generated in all operating modes. For decommissioning, spent fuel spheres are removed from the spent fuel tanks and loaded into the spent fuel transport casks for final disposal.

The two fast spectrum high temperature gas-cooled reactors – EM^2 and FMR, both have radioactive waste management systems to collect, process and store radioactive materials from plant operating systems, and storage facilities prior to being disposed or recycled.

Liquid metal-cooled fast neutron spectrum SMR

The fast neutron spectrum reactors are well known for their potential to substantially reduce the burden of generated nuclear waste through the burning of Pu and Minor Actinides recycled as new fuels. This is still a major consideration in pursuing fast spectrum SMRs but other aspects such as very long fuel cycle is also important. The sodium-cooled 4S SMR simplified-plant design contributes to waste reduction during operation and decommissioning. Some of the LMFR-type SMR designs also progressively apply the radiation-equivalent approximation (in relation to natural occurring raw materials background doses) in its radioactive waste disposal studies. This requires the development of fuel that can accommodate the recycling of the minor actinides. For the design ARC-100, long term storage is planned in containers that meet the requirements for the deep geological repository design of the Nuclear Waste Management Organization (NWMO).

Molten salt SMRs

In all the (thermal spectrum) MSRs included in the booklet, the gaseous fission products are actively removed and stored (to decay). Several designs propose the removal of the fuel salt (after cooling) to a central facility for reprocessing or conditioning and disposal. So far there are no routes for spent salt clean up that requires pyro processing technologies and disposal. The residual fuel waste will be transported to geological repositories when the plant is in the decommissioning stage. Actinides such as U/Pu/Th/MA separated at the off-site reprocessing facility are recycled to MSR and the FPs and salt will be stored to cool down at the disposal facility. In this way (in theory) the lifetime of the waste can be reduced to the few hundred years to decay (excluding the few very long-lived fission products).

Microreactors

In the case of microreactors, spent fuel and radioactive waste management approaches mostly follow that of the reactor family and coolant-type. For example, the Fully Ceramic Micro-encapsulated (FCM™) fuel from the MMR design may be treated similar to the HTGRs coated particle fuels. Research and development on the spent fuel and radioactive waste management of some unique fuel forms should be further pursued.

Table VII-1 Waste Management and Disposal Plan adopted by SMR Designs

Waste Management and Disposal Plan Categories	SMR designs by type of coolants and technology characteristics				
	Water-cooled Reactors	**HTGR-type SMRs**	**Liquid metal-cooled fast neutron spectrum SMRs**	**Molten Salt SMRs**	**Microreactors**
Volume Reduction and Conditioning	Volume reduction of all waste types are the common principle.	Coated particle separation from graphite will reduce volume by up to a factor of 100.	- Simplified designs contribute to waste reduction. - Development of fuel to include recycled minor actinides to reduce spent fuel lifetime.	- Actinides such as U/Pu/Th/MA separated at the off-site reprocessing facility to be recycled. - Fission products and salt are stored to decay or disposed of.	Small volumes relative to rest of nuclear programme for nuclear power nations.
Waste Processing	For land-based water-cooled SMRs: similar to that of operating advanced WCRs, using the available technology solution.	- Low and intermediate level waste from plant operation will be conditioned by different process technologies. - Possible graphite recycling, ^{14}C separation process	The required processes have been studied by advanced countries as part of their (large) fast reactor programmes.	Gaseous fission products are removed during operation and stored onsite to decay.	All radioactive waste generated from operation will be transferred to the designated waste area, to be categorized and packaged for removal from the NPP site.
Storage Approach, Spent Fuel Pool Cooling Mechanism	- Approach are very similar to that of the current fleet of large LWRs. - In general, designers have plans to safely store, handle and dispose of all the spent fuel including the on-site storage.	- With higher thermal efficiencies the radiotoxicity and decay heat will be lessened by 50% for HTGRs as compared to LWRs. - Dry storage with natural convection after short material active cooling. - Facilities for long-term storage of spent fuel and solid radwaste are in the NPP complex	Spent fuel will be stored until reprocessing and fuel cycle closure becomes economically viable.	- Development of special fuel cask or fuel is cooled within the reactor (tank or pot). - KP-FHR with TRISO pebble fuel, the waste is packaged in multi-purpose canisters for dry interim storage and subsequent off-site transportation for direct geologic disposal or recycling.	Typically, microreactors have life-time core loading and therefore no storage in the plant.
Spent Fuel Take-back Option, Market potential	- For marine-based water-cooled SMRs deployed in host countries, spent fuel will be taken back to the vendor's country, e.g. Russian Federation. - For embarking countries, spent fuel processing can be done in the reactor supplier's country.	- Spent fuel take back is to date not considered in HTGRs. - With TRISO fuels, the core has low power density. If affordable, accident tolerant fuel can be used.	Spent fuel take back is to date not considered in LMFR-type SMRs.	For some MSR designs, if deployed in host countries, spent fuel and reactor modules can be taken back to the vendor's country,	Most designers propose fuel to be handled by central facility.

ANNEX VIII

Enabling Design Features to Facilitate SMR Decommissioning

Introduction

Decommissioning of nuclear power facility is an activity at the last phase of the nuclear facility lifecycle, which is unavoidable. It does not produce operational economic income and generally requires an extended period for its completion. Therefore, successful implementation of decommissioning is one of the major factors for realising low cost of nuclear power generation and will depend on various considerations including the initial design. In addition, taking decommissioning into account from the design phase will increase public acceptance of the development of new reactors.

Design and operational regimes of SMRs might benefit from accumulated decommissioning experiences, best practices and lessons learned to optimize the decommissioning process. Specific innovative features, such as modular design, selection of advanced low impurities construction materials or easier decontamination access should make the decommissioning process more effective, faster, limiting/optimizing secondary waste and less costly. However, some challenges will remain, including development of specialized dismantling tools and remote handling equipment or multi-facility decommissioning of SMRs at the same site. This should be considered at the design phase to have the SMRs technology better adapted for the future decommissioning needs.

In addition to the SMRs technical aspects, the long-term liability in terms of decommissioning might be well supported through the proper decommissioning planning and costing, record keeping of documents important for decommissioning, digitalization for decommissioning etc. Circular economy opportunities might be open considering possible reuse of SMR modules or their locations for new reactors, especially if the preferrable immediate dismantling strategy will be applied. What will continue to be crucial in the next decade is the effective integration of decommissioning operations and management of specific waste arisings from SMRs.

A large number of factors were found which should have been considered or reflected in the design stage in decommissioning of NPPs and research reactors. In addition to that should decommissioning experience and lessons learned from the current nuclear facilities such as nuclear power plants or research reactors support new designs of SMRs. Some considerations implemented on the design stage could lead to minimising an economical loss or contribute to the safe and more effective activities during the decommissioning phase.

Decommissioning Aspects of SMR Design

The IAEA is engaged in efforts with Member States to develop an understanding of steps being taken in the design phase of SMRs to address decommissioning needs.

The launch of a Survey was decided in 2019 during the first Consultancy meeting on SMRs Design for Decommissioning. The first draft was provided by the invited experts. In 2020, during the second Consultancy meeting on SRMs Design for Decommissioning, the draft was updated. In June 2021, the survey was finalized after some tests within the Agency, with the assistance of the NPTDS (Nuclear Power Technology Development Section). Then, 128 requests were sent to designers, vendors and other stakeholders. 22 replies from 18 Member States were received by 3Q 2021.

The analysis done by the Agency shows that:

(i) The most relevant topics (priority in choosing the first proposal answer): 68% of the repliers will periodically update their preliminary decommissioning plans, 55% of the repliers have decommissioning specialists in their organizations, 42% of the repliers have examined the advantages and disadvantages of their coolants and moderators to limit the waste production.

(ii) Consensual answers: the periodical updates of the preliminary decommissioning plans, the estimation of the amounts of radioactive waste generated during decommissioning, the consideration of waste management cost in the overall cost of SMR lifecycle, the consideration of standard tools or specifically developed techniques as part of design, the access to the lessons learned, the feedback of lessons learned from power reactors and research reactors, and the number of decommissioning specialists in the organizations.

(iii) Due to the diversity of the projects, answers without a clear trend: the design of tools or robots for maintenance or decommissioning, the design of paths to remove reactor modules in one piece, and the transportation of modules to external dismantling workshops.

The survey provides a positive initial overview of SMRs design for decommissioning. The analysis of the replies shows relevant information and interesting optional comments. Furthermore, it was suggested that the second "advanced" questionnaire will be administered in 2022 based on the additional ideas and suggestions raised by participants of Technical Meeting in November/December 2021 at the IAEA.

Additionally, there was discussion around the possibility of considering the aspect of maintenance of systems needed after shutdown (ventilation, handling systems, pipes, valves, etc.), fuelling/defuelling, radiological shields to remove big components or structures (or to protect workers during maintenance) in the design stage. System and tools that will be used during decommissioning must be identified to keep them operational. Also mentioned was the importance of the costing estimation and its update even after shutdown as well.

Decommissioning Lessons Learned and Aspects Related to SMR Designs

The general rule for the effective decommissioning of all nuclear facilities including SMRs is to benefit from the "decommissioning-friendly" design. In case of SMR, the nuclear island could be considered as exchange module which could be transported to the factory of origin. This approach could reduce costs and scope of activities on the locality, increase nuclear and radiation safety and acceptance of the SMR use. In general, the adopted decommissioning approaches depend on the particular SMR design and more importantly on a country's existing spent fuel and radioactive waste management plans and practices.

Most SMRs are still in the conceptual design phase. The existing national policy and strategy of a country are also significant in determining radioactive waste management (RWM) approaches. Establishing responsibilities and funding mechanisms, as well as setting out technological and programmatic decision making on RWM, are all essential considerations in whole decommissioning and back-end part of SMRs existence.

Technical and organizational aspects of SMRs specific problems might range from the facility design details including modularity, selection of construction materials etc. to how to be better prepared in terms of decommissioning planning and how to apply or adapt existing decommissioning strategies in line with the radioactive waste management. There might be also considered potential for standardization of some decommissioning activities (e.g., full-scope decontamination or dismantling of primary equipment) after the operational lifetime will be over.

There were different lessons learned accumulated from the past activities of nuclear facilities decommissioning which were incorporated, e.g., to IAEA-TECDOC-1657 "Design Lessons Drawn from the Decommissioning of Nuclear Facilities". Design for decommissioning is also one of the aspects required in many new Safety Standards.

General Discussion on SMRs for Decommissioning

The knowledge of the chemical compositions (including impurities) of the used material is required to calculate the radiological inventories of activated waste. Because the cost of chemical analyses is lower for non-activated samples, the sampling should be done prior to operating the reactor. The sampling can also be done by the steel manufacturer to reduce the cost and at the same time to provide analyses to measure with a high sensitivity a wide range of nuclides. As an example, Co-60 will impact the dose rate of the workers and the shielding that will be used during the decommissioning and the waste production. To limit Co-60 arisings, the concentration of metallurgy Cobalt should be limited.

Other 'difficult to measure' radionuclides could directly need to be also considered if they impact the classification of the waste. Some of them are produced by neutron activation of impurities ignored by the steel manufactures.

An important outcome of the experience of NPPs under decommissioning and NPPs still in operation, but approaching the decommissioning phase, is that an accurate, detailed and complete radiological activation inventory of structure and components needs to be completed. It allows an optimization of dismantling activities considering waste management strategy and radioprotection measures. In particular, it could overcome the recourse, after the shut-down, to perform laborious activation characterization campaigns, avoiding workers exposure and additional cost, allowing to perform confirmation measurements directly during the on-going dismantling activities. In addition, the knowledge of structures and components activation during the operation could be useful for maintenance program and also in case of need of some component/structure removal for replacement.

Basically, the activation of structure and components of the reactor could be obtained by the numerical simulations that require:

- 3D model of the reactor, with a detailed representation of structures inside and outside the vessel, and different temperatures of the primary coolant;
- 3D neutron sources models;
- 3D multigroup neutron flux map on the structures of interest taking into account as much as possible different neutron source distributions (different power distributions) reflecting different loading pattern, average control rods position, average absorber concentration etc, or at least significant power distribution variations;
- Calculation of the activities for each structure or component of interest, taking into account the exact chemical composition of material paying attention to impurities concentrations;
- Validation and continuous improvement of the calculation scheme and model comparing result with measurement of detectors/samples such as for example that ones of vessel fluence monitoring program.

It could be useful to consider these aspects from the earliest phases of design, construction and operation of SMRs (and of course in general for any other future reactors). During the design phase, it could be useful to foresee an integration of the power and vessel fluence monitoring program implementing suitable detectors/samples in locations of interest inside and outside the vessel properly oriented to the activation inventory evaluation. During the construction it would be necessary to collect and record all the information regarding chemical composition of all structures and components materials with particular attention to impurities concentration. Also during the operation phase, it could be important to continuously tune the calculation scheme and model considering the different operating conditions including the shutdown phases.

Concerning the cost and the funding, the holistic view of the life cycle must be considered (design, building, operating and decommissioning). A work has been done by OECD/NEA in this regard to propose a methodology for a comprehensive approach to determine cost covering full life cycle, including for SMRs.

The decommissioning strategy depends, besides other factors, on the existing / planned waste routes and the design of nuclear facility. If the waste routes are not known or final disposal not existing, the immediate dismantling will be practically impossible. The use of particular / specific material will impact the waste route. The design and accessibility of components will determine the straightforwardness of the decommissioning initiative. It is important to keep in mind that cost, strategy and available waste routes are all interconnected factors when tackling decommissioning projects.

Concluding Remarks

According to the IAEA’s Advanced Reactors Information System (ARIS), there are over 80 SMR designs and concepts globally at various developmental stages. Some of them are on the advanced stages of construction and claimed as being near-term deployable. The SMR related research and development are also being conducted in the existing and newcomer nuclear energy countries.

The initiative to incorporate features that consider decommissioning needs into SMR design is a great opportunity for the nuclear industry to further increase the appeal of these revolutionary technologies. Addressing such issues from the conceptual design stage will allow for an easier and much more efficient decommissioning when that time arrives.

The ‘plug and play’ nature of small modular reactors may allow for the ‘unplugging’ and storage of an entire module upon decommissioning, thus reducing much of the energy and material demands of the decommissioning process. Modularisation could enable opportunities for improving SMR decommissioning. A novel topic potentially improving SMR decommissioning is the link between modularisation and circular economy.

ANNEX IX
Experimental Testing for Design Verification and Validation

Introduction

SMRs from major types of technologies, including WCRs, HTGRs, LMFRs, MSRs and microreactors may adopt innovative features that involve or introduce novel physical phenomena. Advanced reactor designs including SMRs propose design solutions that have not been adopted before and which sometimes are not yet supported by extensive operational experience and experimental data. A number of designs propose design solutions that have to be supported by extensive operational experience and experimental data. Considering those factors, new scientific modelling combined with high-performance computing must be developed to solve complex new mechanisms, phenomena, and processes occurring in advanced reactors. Design tools and computer codes for safety analysis require verification and validation (V&V) for the purpose of licensing or design certification. Experimental testing is conducted for design verification and validation, particularly for computer codes used for safety analyses to support licensing and/or design certifications. Design verification is performed to ensure the design complies with customer requirements, technical requirements, regulatory requirements, and codes and standards.

Water-cooled SMRs

Integral PWR is a major type adopted for SMRs, where the primary system is integrated in the RPV. Several near-term deployable SMRs are of iPWR-type. Several tests are required for validation, design certification and deployment of SMRs. The impediments in the deployment of SMRs need to be resolved through testing and qualification of components, R&Ds, training and international collaboration. Integral effects test (IET), separate effects test (SET) and component effect test (CET) facilities are fundamental for the licensing purposes. The SET facilities and related testing are important to demonstrate both the safety and the performance issues of innovative components (e.g., internal CRDMs, internal pressurizers, internal SGs. SET and IET are the major tests required for the validation of innovative technology and design, which includes – thermal-hydraulic test, computer models and code validation used to predict the thermal efficiency, performance, and safety of a reactor etc. The integral thermal-hydraulic test examines the system interaction in the design. While the data from the SETs are being used to develop and verify safety analysis model, the IETs is being used to verify the capability of the analysis method and predict the integrated innovative safety systems. The SET and IET performed for different SMR design with focus on iPWR-type is discussed below.

CAREM developed by CNEA, Argentina

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 was designed for reactivity adjustment and control rods calibration. It will also be used to verify the behaviour of the fast shutdown rods (FSS).

ACP100 developed by CNNC, China

Seven verification tests were completed for CRD line cold and hot test, CRD line anti-earthquake test, Internals vibration test research, Fuel assembly critical heat flux test research, passive ECCS integration test, CMT and passive RHR system test, and passive containment heat removal test.

SMART developed by KAERI, Republic of Korea

The advanced design features of SMART were verified by a comprehensive technology validation program that includes safety and performance tests. The safety tests consist of core CHF tests, SETs and IETs of the safety systems, thermal-hydraulic experiments, and digital man-machine interface system (MMIS) tests. The performance tests covered fuel assembly out-of-pile tests, performance tests of the major components including RPV dynamic tests, RCP moc-kup test and SG irradiation test, and MMIS control room tests.

VOYGR^{TM} developed by NuScale Power Inc., United States of America

NuScale has designed, built, and operates a one-third scale prototype testing facility that replicates the entire NPM and its safety systems including the reactor building cooling pool. It provides an electrically heated core to bring the system up to operating temperature and pressure. This facility is used to test the NPM and gather data for thermal hydraulic codes, safety analysis code, and reactor design validation. NuScale has 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 testing programs have been audited by the NRC.

HTGR-type SMRs

HTR-10 developed by INET, Tsinghua University, China

The HTR-10 is the fundamental test reactor essential to develop the HTR-PM. 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. The following engineering experiments are needed prior to the commissioning: (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.

GTHTR300 developed by JAEA, Japan

The test results using the HTTR will be utilized for realization 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 GTHTR300, which is meant for commercial unit, shall demonstrate its ability to operate in normal cogeneration mode or with electric or hydrogen system operating alone in such a case as forced shutdown of either system. 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.

Liquid Metal-cooled Fast Neutron Spectrum SMRs

BREST-OD-300 developed by NIKIET, Russian Federation

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.

SEALER-55 developed by LeadCold, Sweden

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 3 MW, produced in 7 heated rod assemblies with 37 rods each. The height of the prototype is 1:1 with respect to the SEALER-55 to validate residual heat removal capability by dip-coolers, whereas the diameter of the vessel is scaled by 1:2.5. The prototype is under engineering design, with the intent to have it in operation by 2024.

Molten Salt SMRs

IMSR400 developed by Terrestrial Energy Inc., Canada

TEI 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 Slat 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.

Microreactors

MARVEL developed by Idaho National Laboratory, United States of America

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 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. 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 modelling and simulation parameters; (iii) streamline manufacturing methods; (iv) de-risk supply chain; and (v) train operators.

HOLOS-QUAD developed by Holos Gen, United States of America

The design has been scrutinized for approximately 4 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 under this program included technical and economic feasibility executed with high-fidelity codes developed by the national laboratories. The results of these activities were published through numerous peer-reviewed technical papers addressing, thermal-hydraulic, heat transfer, load-following, transient analyses, loss of coolant, loss of electrical power, flooding, passive decay heat removal and structural aspects of the design. As part of validation activities, costing analyses of the design's SSCs' life cycle were executed including upfront cost for decommissioning, spent fuel transport and storage for 50 years.

Table IX-1 Examples of Tests and Tests for integral-PWR Design Developments and Licensing

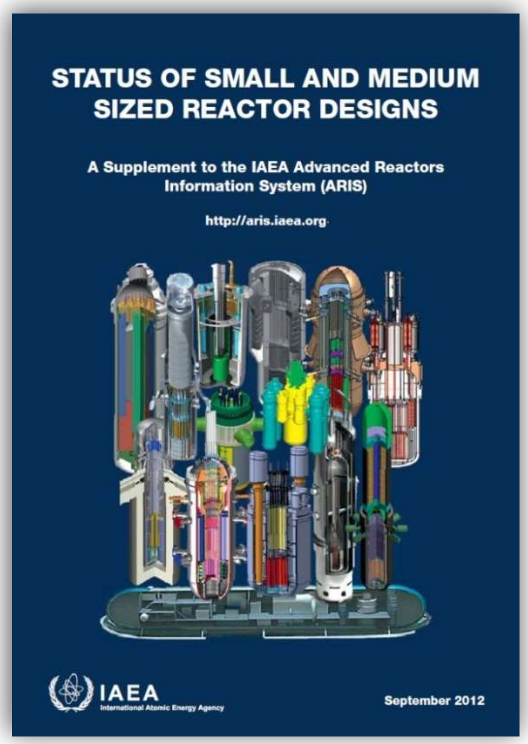
SMR Designs, Power Rating	Specific Features	Component Effect Tests	Separate Effect Tests	Integral Effect Tests
International Reactor Innovative and Secure (IRIS) 335 MW(e) / 1000 MW(t) **Notes:** *The design description of IRIS was reported in the previous editions of ARIS Booklet from 2012 to 2020.*	• The integral reactor coolant system consists of 8 helical-coil SGs, internal CRDM and integral pressurizer installed within the RPV. • Heat removal from inside the vessel depressurizes the primary system by condensation; effective heat removal by SG's emergency heat removal system (EHRS). • SGs safety valves eliminated • Passive EHRS • Passive containment cooling. • Auxiliary building is fully seismic isolated.	• Full scale test of steam generator helical-coil tubes. • EHRS passive safety systems test scaled on power/volume.	• The containment compartments are simulated in SPES3 by separate tanks connected, representing the dry-well, two pressure suppression systems, two long-term gravity makeup systems, the reactor cavity, and ADS quench tank. • The three ADS trains are simulated in SPES3 by two trains. Each train consists of a safety valve, a line to the quench tank and a line to the dry well.	IRIS integral test facility: • Full scale in height – temperature – pressure • Scaled 1:100 in power/volumes • Testing of most accident scenarios. • Validation of codes and behaviour of passive safety systems and containment-vessel coupling. • >700 measurement points, new instrumentation developed.
CAREM ~30 MW(e) / 100 MW(t)	Integral PWR – primary coolant system within the RPV, self-pressurized, in-vessel hydraulic CRDMs, primary coolant flow relies entirely on natural convection and safety systems relying on passive features.	Fuel element, already tested in the RA-6 reactor.	• High-pressure and high-temperature rig test for the innovative hydraulic CRDMs. • Thermal limits and CHF tests. • Low pressure loop: hydraulic losses and flow vibration test. • CHF facility at Pica, Argentina for neutronic code validation.	Thermal-hydraulic dynamics test in conditions similar to CAREM25 operational states (1:1 in height and pressure, temperature – 1:335 in power) – natural circulation and self-pressurization test.

SMR Designs, Power Rating	Specific Features	Component Effect Tests	Separate Effect Tests	Integral Effect Tests
ACP100 125 MW(e) / 385 MW(t)	• Integrated reactor design technology • Passive severe accident prevention and mitigation action, such as for containment hydrogen eliminator, cavity flooding etc. • A multipurpose designed for electricity production, heating, steam production or seawater desalination • Vertically mounted RCP • CRDM of magnetic force type • Reactor building and spent fuel pool are located below ground level • Passive decay heat removal system.	Nuclear fuel fabrication plant at Yibin manufactured a shortened CF2 fuel assemblies and control rod samples for R&D project at Nuclear Power Institute of China (NPIC).	Fuel assembly CHF test: • In typical unit at uniform heat flux • In typical unit at non-uniform heat flux • In non-typical unit at non-uniform heat flux • Control rod drive line (CRDL) anti-seismic test. Passive safety systems tests: • Passive ECCS integration test • CMT and PRHRS test • Flow-induced vibration test • Test indicates that the maximum stress was more less than high cycle fatigue allowable stress under 100% rated flow condition • CRDL cold and hot test	• Core cooling and residual heat removal system test facility (CREST) (1:1 in height and pressure, temperature; 1:37 in volume; 1:100 in power). • CNNC/NPIC has built up comprehensive testing facilities which fulfil the needs of design, and results of the test research on the crucial technology provided necessary basis for finial design and safety evaluation of the reactor.

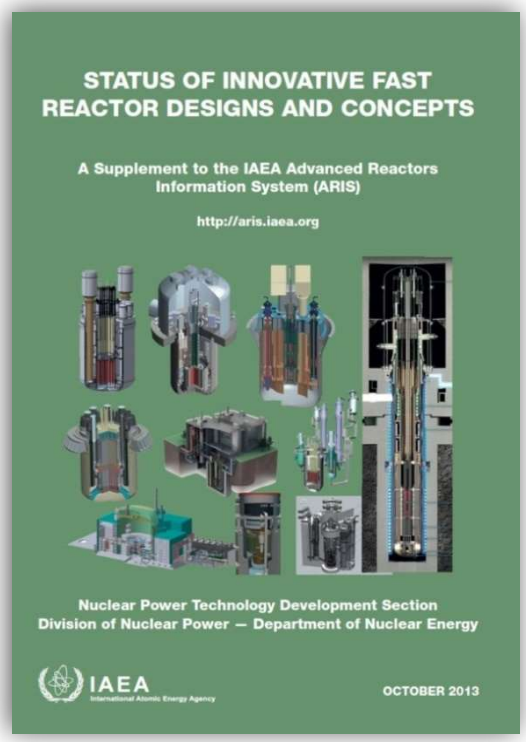
SMR Designs, Power Rating	Specific Features	Component Effect Tests	Separate Effect Tests	Integral Effect Tests
SMART 107 MW(e) / 365 MW(t)	• RPV contains all of the primary components such as core structures and SGs. • A passive decay heat removal system in the secondary side • Horizontally mounted RCPs. • Intended for sea water desalination and electricity supply in newcomer countries with small grid.	Basic fundamental thermal-hydraulic experiments include: • Boiling heat transfer characteristics in the helically coiled SGs. • Containment cooling system using heat pipe. • Verification and performance test of hydraulic valve characteristics measurement test. • The performance tests for key parts of main coolant pump • Irradiation test for SG tube material (A690)	Fuel Performance Tests: • Fuel component test. • Fuel assembly out-of-pile test. • Freon CHF test. • Water critical heat flux (CHF)/mixing test. Safety validation tests: • Safety injection bypass test (1:5 scale). • SG and passive residual heat removal system HX heat transfer test (scaling ratio height/volume–1:2.8, 1/473) Thermal-hydraulic performance tests: • RPV assembly flow distribution test (1:5 scale) • Flow mixing header assembly performance test (1:5 scale) • Internal pressurizer/level measurement test (1:6 scale) • Tests for various I&C systems • Software validation of key design tools and methods: core physics, core thermal-hydraulics, safety analysis etc. • Test of flow instability in steam generator • Test of self-pressurizer performance • Two-phase critical flow test with on-condensable gases	• Thermal-hydraulic integral effect test was carried out at VISTA-ITL test facility. • Experiment for natural circulation in integral reactor • The integral thermal hydraulic test which examines the system interaction in the design is performed. • The high temperature and high pressure integral thermal hydraulic test facility established performed various comprehensive thermo-hydraulic tests.

SMR Designs, Power Rating	Specific Features	Component Effect Tests	Separate Effect Tests	Integral Effect Tests
VOYGR™ 77 MW(e) / 250 MW(t)	• Natural circulation cooled; decay heat removal using containment; built below ground. • RPV is placed within a containment vessel made of high-strength stainless steel, submerged underwater in a below-grade pool shared by all modules. • 100% turbine bypass capability; dry cooling option; enhanced features for load-following; island mode capability during loss of off-site power; flexibilities for process heat applications.	Prototypic full-scale thermal-hydraulic test conducted at Stern Laboratory in Canada to obtain data for validating sub-channel code for design certification with the NRC: • CHF data for correlation development. • Fuel bundle sub-channel exit temperatures to determine mixing coefficients. • Fuel bundle single and two-phase pressure drops. • Full-length NuScale fuel design. • 5 × 5 electrically heated array. • Uniform and symmetric cosine power peaking test sections. • Demonstration of a proof-of-concept inspection system test for the helical coil SG using available supplier tooling successfully completed.	To obtain the necessary test data, NuScale embarked on two major test programs utilizing specialized facilities at Società Informazioni Esperienze Termoidrauliche (SIET) in Piacenza, Italy. • A full-length SG tubes test to investigate important physical phenomena and processes that occur in the SG tubes and on the primary side tube bank at high pressure and temperature with focused on the secondary side performance • The second test focuses on overall primary and secondary side performance and consists of a prototypic tube bank operated at prototypic primary and secondary flow conditions. A full-scale upper module mock-up of the reactor is built. It includes top of the reactor module down to the elevation of reactor vessel head. All major components are mocked up including: • The upper portion of the containment vessel • Major piping such as steam, feed water, and chemical/volume control • CRDMs • Major valves such as isolation valves and emergency core cooling valves • Module service platform.	The NuScale integral system test (NIST-1) facility located in Corvallis, Oregon is a 1/3 scale in volume/height and 1:1 in pressure and temperature prototype that replicates the power module and reactor building cooling pool. It provides an electrically heated core to bring the system up to operating temperature and pressure. • Stability testing ensures that throughout the operating conditions, stable natural circulation is assured. • Large-scale real-time integral effects data for LOCAs, long-term core cooling, and non-LOCA transients to validate design and analysis tools for design certification. • A 12–module control room simulator to supports human factor engineering studies, control room staffing exemption, control room design and plant performance studies.

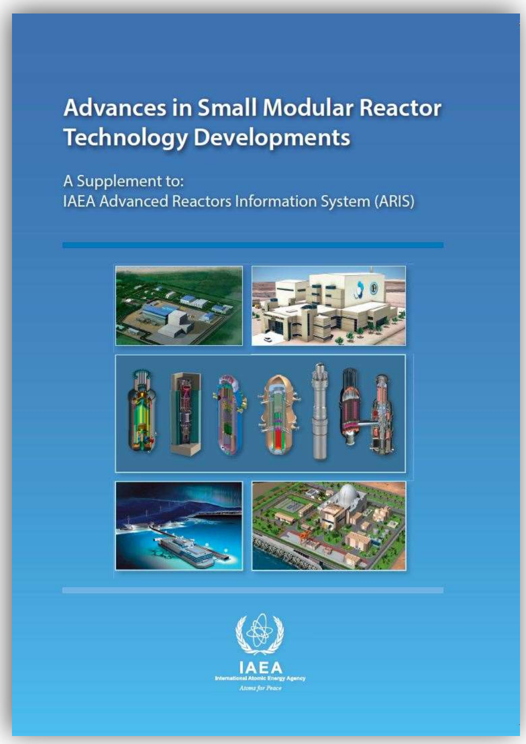
ANNEX X
Bibliography



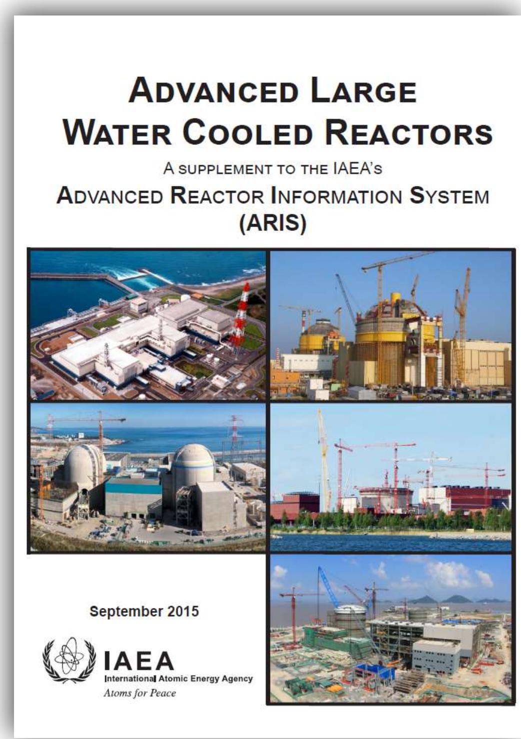
- Contained status, design description and main features of 32 selected SMR designs;
- Sorted by type/coolant: iPWR, PHWR, GCR, and LMFR;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), FBNR (Brazil), CNP-300 (China), Flexblue (France), IMR (Japan), SMART (Republic of Korea), ABV-6M (Russian Federation), SHELF (Russian Federation), RITM-200 (Russian Federation), VK-300 (Russian Federation), VBER-300 (Russian Federation), WWER-300 (Russian Federation), KLT-40S (Russian Federation), UNITHERM (Russian Federation), IRIS (International Consortium), mPower (USA), NuScale (USA), Westinghouse SMR (USA), EC6 (Canada), PHWR-220 (India), AHWR300-LEU (India), HTR-PM (China), PBMR (South Africa), GT-MHR (USA), EM^2 (USA), CEFR (China), 4S (Japan), PFBR-500 (India), BREST-OD-300 (Russian Federation), SVBR-100 (Russian Federation), PRISM (USA), G4M (USA).
- **Published September 2012**



- Contained status, design description and main features of 22 selected fast reactor designs;
- Sorted by type/coolant: SFR, GFR, and HLMC, MSFR;
- Sorted by Country of Origin;
- Included: CFR-600 (China), ASTRID (France), FBR-1&2 (India), 4S (Japan), JSFR (Japan), PGSFR (Republic of Korea), BN-1200 (Russian Federation), MBIR (Russian Federation), PRISM (USA), TWR-P (USA), MYRRHA (Belgium), CLEAR-I (China), ALFRED (Europe/Italy), ELFR (Europe/Italy), PEACER (Republic of Korea), BREST-OD-300 (Russian Federation), SVBR-100 (Russian Federation), ELECTRA (Sweden), G4M (USA), ALLEGRO (Europe), EM^2 (USA), MSFR (France).
- **Published October 2013**



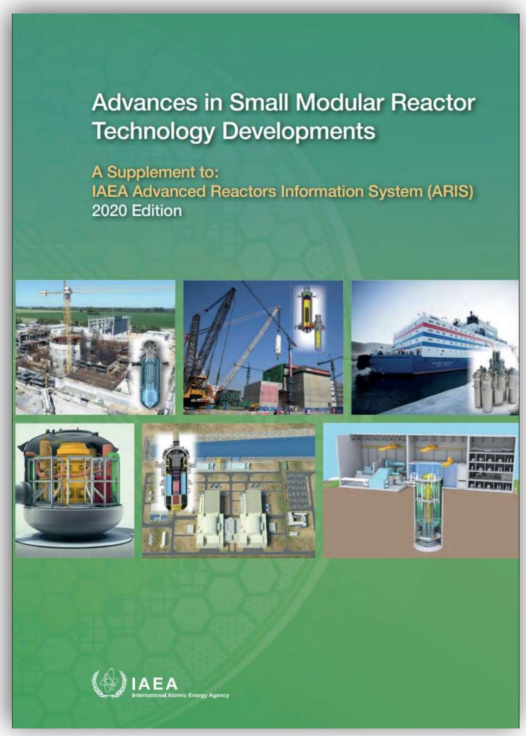
- Contained status, design description and main features of 31 selected SMR designs;
- Sorted by type/coolant: iPWR, AHWR and HTGR;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), ACP-100 (China), Flexblue (France), IMR (Japan), SMART (Republic of Korea), ABV-6M (Russian Federation), SHELF (Russian Federation), RITM-200 (Russian Federation), VK-300 (Russian Federation), VBER-300 (Russian Federation), KLT-40S (Russian Federation), UNITHERM (Russian Federation), IRIS (International Consortium), mPower (USA), NuScale (USA), Westinghouse SMR (USA),SMR160 (USA), AHWR300-LEU (India), HTR-PM (China), PBMR (South Africa), GT-MHR (Russian Federation),VVER-300 (Russian Federation), RUTA-70(Russian Federation), ELENA (Russian Federation), DMS (Japan), HTR-PM (China), GTHTR300 (Japan), MHR-T (Russian Federation), MHR-100 (Russian Federation), PBMR-400 (South Africa), HTMR-100 (South Africa), SC-HTGR (USA), Xe-100 (USA)
- **Published September 2014**



- Contained overview of status and main features of 18 selected large water-cooled reactor designs;
- Sorted by Country of Origin/Vendor;
- Included: ACPR-1000 (China), CAP-1400 (China), CPR-1000 (China), HPR1000 (China), APR1400 (Republic of Korea), APWR (Japan), AP1000(Japan), ABWR(Japan), VVER1000(Russian Federation),, VVER1200(Russian Federation),.VVER1500(Russian Federation), IPHWR(India), EPR(France), KERENA(France), ATMEA1(France) , EC6 (Canada), ABWR(USA), ESWR(USA)
- **Published September 2015**

- Contained status, design description and main features of 48 selected SMR designs;
- Sorted by Land based and Marine based LWRs, HTGR, Fast spectrum SMRs and Molten Salt SMRs;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), ACP100, CAP150, CAP200 (China), AHWR-300 (India), IRIS (International Consortium), DMS and IMR (Japan), SMART (Republic of Korea), UNITHERM, KARAT-45, KARAT-100, ELENA, RUTA-70 (Russian Federation), NuScale, mPower, Westinghouse SMR, SMR-160 (United States of America)
  ACPR50S (China), Flexblue (France), KLT-40S, RITM-200, VBER-300, ABV-6E, SHELF (Russian Federation),
  HTR-PM (China), GTHTR300 (Japan), GT-MHR, MHR-T, MHR-100 (Russian Federation), PBMR-400, HTMR-100 SMR (South Africa), SC-HTGR, Xe-100 (United States of America)
  LEADIR-PS (Canada), 4S (Japan), BREST-OD-300, SVBR-100 (Russian Federation), G4M, EM2 (United States of America)
  Integral Molten Salt Reactor (Canada), MSTW (Denmark), ThorCon (International Consortium), FUJI (Japan), Stable Salt Reactor (United Kingdom), SmAHTR, Liquid Fluoride Thorium Reactor, Mk1 PB-FHR (United States of America)
- **Published August 2016**

- Contained status, design description and main features of 56 selected SMR designs;
- Sorted by Land based and Marine based LWRs, HTGR, Fast spectrum SMRs and Molten Salt SMRs;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), ACP100, CAP150, CAP200 (China), AHWR-300 (India), IRIS (International Consortium), DMS and IMR (Japan), SMART (Republic of Korea), UNITHERM, KARAT-45, KARAT-100, ELENA, RUTA-70 (Russian Federation), NuScale, mPower, Westinghouse SMR, SMR-160 (United States of America), ACPR50S (China), Flexblue (France), KLT-40S, RITM-200, VBER-300, ABV-6E, SHELF (Russian Federation), HTR-PM (China), GTHTR300 (Japan), GT-MHR, MHR-T, MHR-100 (Russian Federation), PBMR-400, HTMR-100 SMR (South Africa), SC-HTGR, Xe-100 (United States of America)
  LEADIR-PS (Canada), 4S (Japan), BREST-OD-300, SVBR-100 (Russian Federation), G4M, EM2 (United States of America)
  Integral Molten Salt Reactor (Canada), MSTW (Denmark), ThorCon (International Consortium), FUJI (Japan), Stable Salt Reactor (United Kingdom), SmAHTR, Liquid Fluoride Thorium Reactor, Mk1 PB-FHR (United States of America)
- **Published September 2018**



- Contained status, design description and main features of 72 selected SMR designs;
- Sorted by Land-based and Marine-based WCRs, HTGRs, Fast neutron spectrum SMRs, Molten Salt SMRs and Microreactors;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), ACP100 (China), CANDU SMR (Canada), CAP200, DHR400, HAPPY200 (China), TEPLATOR^{TM} (Czech Republic), NUWARD (France), IRIS (IRIS Consortium), DMS and IMR (Japan), SMART (Republic of Korea and Saudi Arabia), UNITHERM, VK-300, KARAT-45, KARAT-100, RUTA-70, ELENA (Russian Federation), UK SMR (UK), NuScale (USA), BWRX-300 (USA and Japan), SMR-160, Westinghouse SMR, mPower (USA)
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  Energy Well (Czech Republic), MoveluX (Japan), U-Battery (UK), Aurora, Westinghouse eVinci, MMR (USA)
- **Published September 2020**

ANNEX XI
Acronyms

ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ACC	Air-cooled Condenser
ACME	Advanced Core-cooling Mechanism Experiment
ADS	Automatic Depressurization System
ADV	Atmospheric Dump Valve
AFA	Alumina Forming Austenite
AGR	Advanced Gas-cooled Reactor
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
ARIS	Advanced Reactor Information System
ASEC	Air Heat Sink for Emergency Cooldown
ASIV	Amphora-Shaped Inner Vessel (in LFR design)
ASME	American Society of Mechanical Engineers
AST	Auxiliary Standby Transformer
ATF	Accident Tolerant Fuel
ATR	Advanced Test Reactor
ATS	Automation System
ATWS	Anticipated Transient Without Scram
AUSC	Advanced Ultra-Supercritial
BCR	Back-up Control Room
BDBA	Beyond Design Basis Accident
BOL	Beginning of Life
BOP	Balance of Plant
BPV	ASME Boiler and Pressure Vessel code
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium (Canadian Pressurized Heavy-water Reactor)
CBR	Core Breeding Ratio
CCF	Common Cause Failure
CCS	Containment Cooling System
CCWS	Component Cooling Water System
CDF	Core Damage Frequency
CED	Contract Effective Date
CEDM	Control Element Drive Mechanism
CES	Containment Enclosure Structure
CHE	Containment Hydrogen Control and Filtration Exhaust System
CHR	Containment Heat Removal System
CI	Conventional Island (Turbine-Generator Building)
CIS	Containment Isolation System
CMT	Core Makeup Tank
CNPP	Cogeneration Nuclear Power Plant
CNPP	Cogeneration Nuclear Power Plant
CNSC	Canadian Nuclear Safety Commission
CNV	Cylindrical Containment Vessel
CPRSS	Containment Pressure and Radioactive Suppression System
CPS	Control and Protection System
CRA	Control Rod Assemblies
CRDM	Control Rod Drive Mechanism
CS	Containment Structure
CSG	Compact Steam Generator

CSMR	CANDU SMR
CSS	Control and Support Safety System
CSS	Control Safety System
CST	Coolant Storage Tank
CSTS	Condensate Storage and Transfer System
CTAH	Coiled Tube Air Heaters
CTS	Chemical Technological Sector
CV	Containment Vessel
CVCS	Chemical and Volume Control System
CWS	Chilled Water System
D/G	Diesel Generator
DAS	Diverse Actuation System
DBA	Design Basis Accident
DBC	Design Basis Condition
DBE	Design Basis Earthquake
DC	Direct Current
DCA	Design Certification Application (a licensing term in the United States)
DCIS	Distributive Control and Information System
DCS	Distributed Control System
DEC	Design Extension Condition
DGR	Deep Geological Repository
DHRS	Decay Heat Removal System
DID	Defence in Depth
DLOFC	Depressurized Loss of Forced Cooling (in HTGR)
DRACS	Direct Reactor Auxiliary Cooling System
DU	Depleted Uranium
DVI	Direct Vessel Injection
EAB	Exclusion Area Boundary
EBI	Emergency Boron Injection
EBT	Emergency Boration Tank
ECCS	Emergency Core Cooling System
ECDS	Emergency Cooling Down System
ECT	Emergency Cooldown Tank
EDG	Emergency Diesel Generator
EFPD	Effective Full Power Day
EHRS	Emergency Heat Removal System
ENTSO	European Network of Transmission System Operators for Electricity
EOC	End-of-cycle
EOL	End of Life
EPZ	Emergency Planning Zone
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Feature
ESWS	Essential Service Water System
ETS	Energy Transport System
FA	Fuel Assembly
FBR	Fast Breeder Reactor
FCD	First Concrete Date
FDT	Fuel salt Drain Tank
FE	Fuel Element
FEED	Front-End Engineering Design
FGCS	Fission Gas Collection System
FHR	Fluoride-salt cooled High Temperature Reactor
FHS	Fuel Handling System
FMCRD	Fine Motion Control Rod Drive (in BWR)
FNPP	Floating Nuclear Power Plant

FOAK	First of a Kind
FPC	Fuel Pool Cooling and Cleanup System
FPGA	Field Programmable Gate Arrays
FPHE	Formed Plate Heat Exchanger
FPU	Floating Power Unit
FPVS	Fission Product Vent System
FRPS	First Reactor Protection System
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
FSS	Free Surface Separation
FSS	First Shutdown System
FSST	Fuel Salt Storage Tank
GCB	Generator Circuit Breaker
GDA	Generic Design Assessment
GDCS	Gravity Driven Cooling System
GDWP	Gravity Driven Water Pool
GFR	Gas-cooled Fast Reactor
GHT	Gas Holding Tanks
GTG	Gas Turbine Generator
GV	Guard Vessel
HALEU	High-Assay Low Enriched Uranium
HE	Heat Exchanger
HEU	High Enriched Uranium
HFE	Human Factors Engineering
HFIR	High Flux Isotope Reactor
HGDPV	Hot Gas Duct Pressure Vessel
HHTS	Hybrid Heat Transport System
HIPS	Highly Integrated Protection System
HLMC	Heavy Liquid Metal-Cooled
HMIS	Human-Machine Interface system
HPB	Helium Pressure Boundary
HPCF	High Pressure Core Flooder (in BWR)
HRSG	Heat Recovery Steam Generator System
HTF	Heat Transfer Fluid
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
HTS	Heat Transport System (in CANDU and PHWR)
HVDC	High Voltage Direct Current
HWR	Heavy Water Reactor
HX	Heat Exchanger
I&C	Instrumentation and Control
IC	Isolation Condenser
ICS	Isolation Condenser System
IHP	Integrated Head Package
IHX	Intermediate Heat Exchanger
IPIT	Intermediate Pressure Injection Tanks
IRACS	Intermediate Reactor Auxiliary Cooling System
IRWST	In-Containment Reactor Water Storage Tank
IST	Integrated System Test
IVR	In-Vessel Retention
LBB	Leak Before Break
LBE	Lead-Bismuth Eutectic coolant
LBLOCA	Large Break Loss of Coolant Accident
LC	Lead Coolant

LCA	Lower Containment Area
LCOE	Levelized Cost of Electricity
LEU	Low Enriched Uranium
LFR	Lead-cooled Fast Reactor
LFTR	Liquid-Fluoride Thorium Reactor
LLSF	Low Level Safety Functions
LOCA	Loss of Coolant Accident
LOFC	Loss of Forced Cooling
LOHS	Loss of heat sink
LOOP	Loss of Offsite Power
LPFL	Low Pressure Core Flooder (in BWR)
LWR	Light Water Reactor
MA	Minor Actinides
MCP	Main Coolant Pump
MCR	Main Control Room
MCSFR	Molten Chloride Salt Fast Reactor
MHT	Main Heat Transport
MMIS	Man-machine Interface System
MOX	Mixed Uranium-Plutonium Oxide fuel
MSA	Moisture Separator Reheater
MSFR	Molten Salt Fast Reactor
MSK	Medvedev-Sponheuer-Karnik scale (a macroseismic intensity scale used in Russia)
MSR	Moisture Separator and Reheater
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
MSSV	Main Steam Safety Valve
MW(e)	Mega Watt electric
MW(t)	Mega Watt thermal
MWS	Makeup Water System
MWS	Metal and Water Shielding
NACC	Nuclear air-Brayton Combined Cycle
NDHP	Nuclear District Heating Plant
NFC	Nuclear Fuel Cycle
NGCC	Natural Gas Combined Cycle
NHP	Nuclear Heating Plant
NHSS	Nuclear Heat Steam Supply System
NI	Nuclear Island
NIS	Nuclear Instrumentation Systems
NPP	Nuclear Power Plant
NPS	Nuclear Power System
NSSS	Nuclear Steam Supply System
NTEP	Nuclear Thermoelectric Plant
NWMO	Nuclear Waste Management Organization
O&M	Operation and Maintenance
OBE	Operating Basis Earthquake
OCP	Outside Containment Pool
OFPU	Optimized Floating Power Unit
OPEX	Operating Expenses
OTSG	Once-Through Steam Generators
OTTO	Once Through Then Out
PAFS	Passive Auxiliary Feedwater System
PAR	Passive Autocatalytic Re-Combiners
PAS	Passive Containment Air Cooling System
PBMM	Pebbled Bed Micro Model
PC	Primary Containment

PCCS	Passive Containment Cooling System
PCHR	Passive Containment Heat Removal
PCS	System Primary Containment System
PCT	Peak Cladding Temperature
PCU	Power Conversion Unit
PCV	Primary Containment Vessel
PDHRS	Passive Decay Heat Removal System
PFB	Passive Feed and Bleed System
PGA	Peak Ground Acceleration
PHTS	Primary Heat Transport System
PHWR	Pressurized Heavy Water Reactor
PHX	Primary Heat Exchanger
PLC	Programmable Logic Controller
PLOFC	Pressurized Loss of Forced Cooling (in HTGR)
PLS	Plant Control System
PMG	Plant Main Generator
PMS	Protection and safety Monitoring System
PORV	Power-Operated Relieve Valve
PRA	Probabilistic Risk Assessment
PRHRS	Passive Residual Heat Removal System
PSAR	Preliminary Safety Analysis Report
PSGC	Passive Steam Generator Cooler
PSIS	Passive Safety Injection System
PSWS	Plant Service Water System
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAB	Reactor Auxiliary Building
RCAB	Reactor Containment and Auxiliary Building
RCCS	Reactor Cavity Cooling System
RCH	Reactor Closure Header
RCI	Reactor Coolant Inventory and Purification System
RCIC	Reactor Core Isolation Cooling (in BWR)
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RCSS	Reactivity Control and Shutdown System
RDP	Reactor automatic Depressurization System
RFA	Robust Fuel Assembly
RGPu	Reactor-grade Plutonium
RHRS	Residual Heat Removal System
RIA	Reactivity Insertion Accident
RP	Reactor Plant
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSS	Reserve Shutdown System
RTNSS	Regulatory Treatment of Non-Safety System
RV	Reactor Vessel
RVACS	Reactor Vessel Auxiliary Cooling System
RW	Radioactive Waste
RWST	Reactor Water Storage Tank
SAS	Small Absorber Sphere
SBO	Station Black-Out
SCADA	Supervisory Control and Data Acquisition

sCO_2	supercritical CO_2
SDC	Shutdown cooling
SDS	Shut Down System (in CANDU and PHWR)
SEU	Slightly Enriched Uranium
SFP	Spent Fuel Pool
SFR	Sodium-cooled Fast Reactor
SFSC	Spent Fuel Storage Canister
SFSF	Spent Fuel Storage Facility
SG	Steam Generator
SGL	Steam Generator in Liquid Region
SGPV	Steam Generator Pressure Vessel
SIS	Safety Injection System
SIT	Safety Injection Tanks
SLCS	Standby Liquid Control System
SMR	Small Modular Reactor
SNF	Spent Nuclear Fuel
SRPS	Second Reactor Protection System
SSC	Systems, Structures and Components
SSE	Safe Shutdown Earthquake
SSG	IAEA Specific Safety Guide
SSLC	Safety System Logic and Control
STSG	Spiral-tube steam generators (in LFR design)
T/G	Turbine/Generator
TC	Turbo Compressor
TCU	Thermal Conversion Unit
TEG	Thermoelectric Generator
TES	Thermal Energy Storage
TEU	Thermoelectric Unit
TGP	Turbine Generator Package
TM	Turbo Machine
TRISO	Tri-structural Isotropic particle fuel
TRL	Technology Readiness Level
TRU	Transuranic waste
UAT	Unit Auxiliary Transformer
UCA	Upper Containment Area
UCO	Uranium Oxy Carbide
UHS	Ultimate Heat Sink
UPS	Uninterrupted Power Supply (System)
URWT	Ultimate Heat Sink and Refueling Water Tank
V&V	Verification and Validation
VCS	Vessel Cooling System
VDR	Vendor Design Review (a licensing term in Canada)
WATSS	Waste to Stable Salt
WDS	Waste Disposal System
WPu	Weapon-Grade Plutonium
WWER	Water-cooled Water-Moderated Power Reactor (Russian PWR)
YSZ	Yttria Stabilized Zirconia pellets

Annex XII

Acknowledgement

<u>Organisations in Member States</u>

Comisión Nacional de Energía Atómica (CNEA), Argentina

Agencia Boliviana de Energía Nuclear (ABEN), Bolivia

ARC Clean Energy, Canada

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National Research and Innovation Agency (BRIN), Indonesia

*new*cleo srl, Italy

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Japan Atomic Energy Agency (JAEA), Japan

Mitsubishi Heavy Industries Ltd., Japan

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International Thorium Molten-Salt Forum (ITMSF), Japan

THORIZON, Netherlands

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Korea Hydro & Nuclear Power Co., Ltd. (KHNP), Republic of Korea

Ulsan National Institute of Science & Technology (UNIST), Republic of Korea

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National Research Centre ‘Kurchatov Institute’, Russian Federation

JSC AKME Engineering, Russian Federation

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PBMR SOC Ltd., South Africa

STL Nuclear (Pty) Ltd., South Africa

LeadCold, Sweden

STAR ENERGY SA, Switzerland

Moltex Energy, United Kingdom

*new*cleo, Ltd, United Kingdom

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Idaho National Laboratories (INL), United States of America

NuScale Power Corporation, United States of America

BWX Technologies Inc., United States of America

Last Energy Inc., United States of America

Westinghouse Electric Company LLC, United States of America

University of California, Berkeley, United States of America

Elysium Industries, United States of America

Oklo Inc., United States of America

HolsGen LLC, United States of America

X-energy, United States of America

General Atomics, United States of America

Ultra Safe Nuclear Corporation, United States of America

ThorCon International, United States of America and Indonesia

StarCore Nuclear, Canada, United Kingdom and United States of America

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