Status Report - HTGR-POLA (Poland)

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1. Overview

- ☐ **In Operation**
- ☐ **Under Construction**
- ☐ **Described in**

Link to PRIS	--

- ☑ **New Concept**

Number of Units	--
Year	2029

Reference Plant

Name of Site	--
Link to PRIS	--
Net Power Output (MWe)	11.5
Gross Power Output (MWe)	--

1.1 Introduction

The implementation of HTGRs in Poland started in 2012, almost parallel to a start of the new Polish Nuclear Power Program of 2010 which aims towards building at least 6 GWe split into 6 or 4 nuclear reactors build in mid 2030ies. The acceleration towards implementation of HTGRs was backed by the ratification of the Paris Agreement in 2015 aiming at reduction of the greenhouse gas emissions. Poland as one of the largest emitters of carbon dioxide in Europe is obliged to reduce emissions, and so it is also investing in renewables which together with nuclear will form the energy mix around 2050.

In 2016 the Minister of Energy appointed the "Committee for Analysis and Preparation of Conditions for Deployment of High-Temperature Nuclear Reactors" which concluded that Poland should build a fleet of industrial HTGRs with an average unit power 180 MWt, steam temperature T = 540C, pressure p = 13.8 MPa, and steam flow 230 t/h in order to fit Polish industry demand. However, due to non-existence of commercial reactors in Poland so far, the experts decided on construction of a research HTGR reactor of some smaller power as a technology demonstrator to convince regulator and the industry about its safety, economy, and practicality. It should possess as many features of the commercial reactor as possible, so that one would be able to scale it up to a commercial size with least effort. This is how an idea of HTGR-POLA reactor of 30 MW thermal power appeared.

It is important to mention that HTGR-POLA is an updated and adjusted to the Polish market requirements

design of the research-demo high temperature gas-cooled reactor based on the Japanese HTTR (High Temperature Engineering Test Reactor). It is a common design project of National Centre for Nuclear Research (NCBJ) and of the Japan Atomic Energy Agency (JAEA), which is going to be built at the site of NCBJ. Since HTTR was designed more than 30 years ago, then some of its solutions are not up-to-date. Besides, it was designed as purely research reactor to establish HTGR fundamental technologies including demonstration of inherent safety features and hydrogen production so that it operated occasionally, not continuously. The objective of HTGR-POLA design is a bigger challenge. It is not going to be a purely research reactor, but the reactor which can also serve a technology demonstrator working continuously and producing electricity and heat for targeted practical applications.

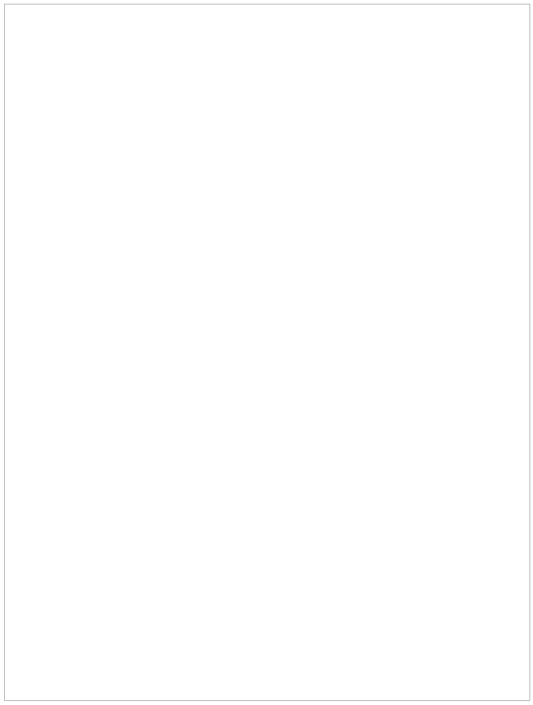


Figure 1. HTGR-POLA reactor.

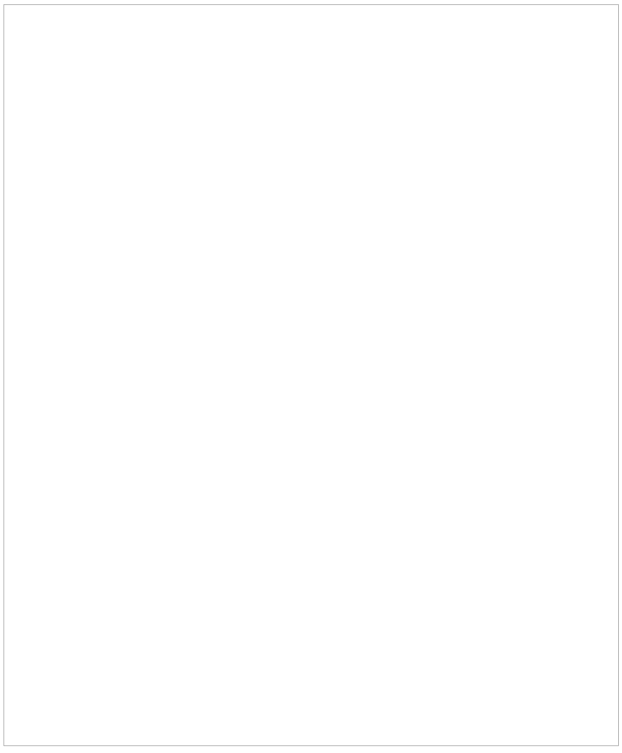


Figure 2. Plant layout.

Target application

The primary mission of the HTGR-POLA is to serve as a demonstrator of HTGR technology, addressing the industrial needs of Poland by supplying superheated steam. The design of the plant incorporates solutions for a conventional island, including the conceptualization of secondary loop facilities. The test plans consider the utilization of superheated steam for future industrial applications, such as integration with hydrogen production facilities or other end-user processes, as well as utility applications for the NCBJ campus, including electricity generation and district heating. Three primary cogeneration modes have been identified, each focused on different output capacities: electricity production with a maximum output of 11.5 MWe, superheated steam with a maximum output of 25 t/h, and communal heat with a maximum output of 16.5 MWt.

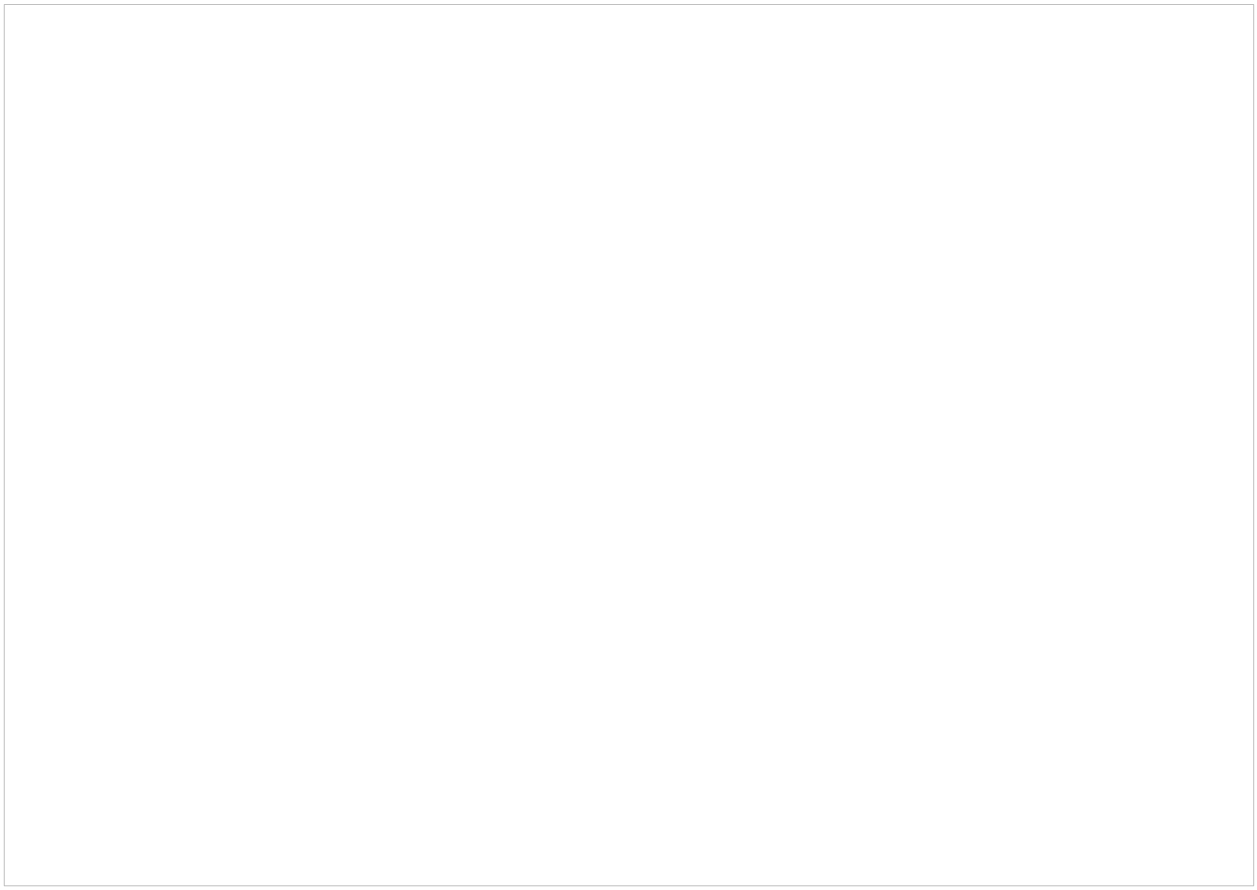


Figure 3. Simplified diagram of the HTGR- POLA NSSS.

1.2 Development Milestones

2019	During 2012-2019, preliminary studies and technological innovation (using available materials) were conducted.
2021	Completion of pre-conceptual design phase and technology validation
2022	Completion of conceptual design phase
2024	Completion of basic design phase
2025-2028	Licensing phase

2025-2028	Detailed design phase
2029-2032	Construction
2033	Commissioning

1.3 Design organization or vendor company

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1.4 Links to designer/vendor website

1.5 Design Description

Design Philosophy

The main assumptions about the research HTGR-POLA reactor are such that it should serve building competence (human resources, industry, regulator, etc.), research tasks, and also a small-scale demonstrator of HTGR technology for industrial applications. One assumes that the design should combine features of the industrial reactor as planned for the European GEMINI Plus reactor design NCBJ was involved in years 2017-2020 as well as proven properties of Japanese HTTR as a research reactor. It possesses a unique core design and a reactor design matching specific Polish requirements in research, demonstration, and applications. An important objective to maximize similarity to an industrial type First Of A Kind (FOAK) 180 MWt reactor design.

1.6 Licensing Application Support Documents

Design and Licensing Status

The pre-concept was finished in 2021 and the concept in 2022. The basic design phase is finished in 2024. Before the end of 2024 some larger pieces of the Preliminary Safety Analysis Report (PSAR) are prepared, though the rest (including site investigations) will be demanded to be finished soon in order to apply for the licensing. In the meantime the technical project is going to be started.

1.7 Reactor Unit in PRIS

2. Category Fields

Table 1: ARIS Category Fields

ARIS Category	Value	Units of Examples
Current/Intended Purpose	Demonstration	e.g. Commercial – Electric/Non-electric, Prototype/FOAK, Demonstration, Experimental
Main Intended Application (once commercial)	Non-electric (industrial applications, district heating)	e.g. Baseload, Dispatchable, Off-grid/Remote, Mobile/Propulsion, Non-electric (specify)

Reference Location		e.g. On Coast, Inland, Below-Ground, Floating-Fixed, Marine-Mobile, Submerged-Fixed (Other-specify)
Reference Site Design (reactor units per site)		e.g. Single Unit, Dual Unit, Multiple Unit (# units)
Core Coolant	He	H2O, D2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)
Reactor Core Size (1 core)	30 MWth <= Thermal power < 1000 MWth	e.g. Thermal Power < 30 MWth, 30 MWth ≤ Thermal power < 1000 MWth, 1000 MWth ≤ Thermal power ≤ 3000 MWth, Thermal power > 3000 MWth
Reactor Type	GCR	e.g. PWR, BWR, HWR, SCWR, GCR, GFR, SFR, LFR, MSR, ADS
Reactor Size	SMR	e.g. PWR, SMR
Neutron Moderator	Graphite	e.g. H2O, D2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)
NSSS Layout		e.g. Loop-type (# loops), Direct-cycle, Semi-integral, Integral, Pool-type
Primary Circulation		e.g. Forced (# pumps), Natural
Thermodynamic Cycle		e.g. Rankine, Brayton, Combined-Cycle (direct/indirect)
Secondary Side Fluid		e.g. H2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)
Fuel Form	Coated Sphere	e.g. Fuel Assembly/Bundle, Coated Sphere, Plate, Prismatic, Contained Liquid, Liquid Fuel/Coolant
Fuel Lattice Shape		e.g. Square, Hexagonal, Triangular, Cylindrical, Spherical, Other, n/a
Rods/Pins per Fuel Assembly/Bundle		e.g. #, n/a
Fuel Material Type		e.g. Oxide, Nitride, Carbide, Metal, Molten Salt, (Other-specify)
Design Status	Basic Design	e.g. Conceptual, Detailed, Final (with secure suppliers)

Licensing Status		e.g. DCR, GDR, PSAR, FSAR, Design Licensed (in Country), Under Construction (# units), In Operation (# units)
Neutron Spectrum	Thermal	e.g. Fast, Thermal

3. Parameter Fields

Table 2: ARIS Parameter Fields

ARIS Parameter	Value	Units of Examples
Plant Infrastructure		
Design Life	60	Years
Lifetime Capacity Factor		%, defined as Lifetime MWe-yrs delivered / (MWe capacity * Design Life), incl. outages
Major Planned Outages		# days every # months (specify purpose, including refuelling)
Operation / Maintenance Human Resources		# Staff in Operation / Maintenance Crew during Normal Operation
Reference Site Design		n Units/Modules
Capacity to Electric Grid		MWe (net to grid)
Non-electric Capacity		e.g. MW(th) heat at x ºC, m3/day desalinated water, kg/day hydrogen, etc.
In-House Plant Consumption		MWe
Plant Footprint	25,000	m2 (rectangular building envelope)
Site Footprint		m2 (fenced area)
Emergency Planning Zone		km (radius)
Releases during Normal Operation		TBq/yr (Noble Gases / Tritium Gas / Liquids)
Load Following Range and Speed		x – 100%, % per minute
Seismic Design (SSE)	0.3g	g (Safe-Shutdown Earthquake)
NSSS Operating Pressure (primary side)	4	MPa(abs), i.e. MPa(g)+0.1, at core outlet
NSSS Operating Pressure (secondary side)	13.8	MPa(abs), i.e. MPa(g)+0.1, at secondary side outlet

Primary Coolant Inventory (incl. pressurizer)		kg
Nominal Coolant Flow Rate (primary)		kg/s
Nominal Coolant Flow Rate (secondary)		kg/s
Core Inlet Coolant Temperature	325	ºC
Core Outlet Coolant Temperature	750	ºC
Available Tempeature as Process Heat Source		ºC
NSSS Largest Component	RPV	e.g. RPV (empty), SG, Core Module (empty/fuelled), etc.
- NSSS Largest Component Length	16200	mm
- NSSS Largest Component Diameter	4080	mm
- NSSS Largest Component Thickness		mm
- NSSS Largest Component Weight	200	tonne
Reactor Vessel Material		e.g. SS304, SS316, SA508, 800H, Hastelloy N
Steam Generator Design		e.g. Vertical/Horizontal, U-Tube/ Straight/Helical, cross/counter flow
Secondary Coolant Inventory		kg
Pressurizer Design		e.g. separate vessel, integral, steam or gas pressurised, etc.
Pressurizer Volume		m3 / m3 (total / liquid)
Containment Type		Dry (single/double), Dry/Wet Well, Inerted, etc.
Containment Volume		m3
Spent Fuel Pool Capacity and Total Volume		Years of full-power operation / m3
Non Electric Application		-
Core Delta Temperature		ºC
Fuel/Core		
Single Core Thermal Power	30	MW(th)

Refuelling Cycle	20	months or “continuous”
Fuel Material		e.g. UO2, MOX, UF4, UCO
Enrichment (avg./max.)	10-12	%
Average Neutron Energy		eV
Fuel Cladding Material		e.g. Zr-4, SS, TRISO, E-110, none
Number of Fuel “Units”		Specify as Assembly, Bundle, Plate, Sphere, or n/a
Weight of one Fuel Unit		kg
Total Fissile Loading (initial)		kg fissile material (specify isotopic and chemical composition)
% of fuel outside core during normal operation		Applicable to online refuelling and molten salt reactors
Fraction of fresh-fuel fissile material used up at discharge		%
Core Discharge Burnup	40	MWd/kgHM (heavy metal, eg U, Pu, Th)
Pin Burnup (max.)		MWd/kgHM
Breeding/Conversion Ratio		Fraction of fissile material bred in-situ over one fuel cycle or at equilibrium core
Reprocessing		e.g. None, Batch, Continuous (FP polishing/actinide removal), etc.
Main Reactivity Control		e.g. Rods, Boron Solution, Fuel Load, Temperature, Flow Rate, Reflectors
Solid Burnable Absorber		e.g. Gd2O3
Core Volume (active)		m3 (used to calculate power density)
Fast Neutron Flux at Core Pressure Boundary		n/(cm²*s)
Max. Fast Neutron Flux		n/(cm²*s)
Core Control Rod Absorber		-
Core Soluble Neutron Absorber		-
Safety Systems		
Number of Safety Trains	Active / Passive	% capacity of each train to fulfil safety function

- reactor shutdown	Active/Passive	- / - / -
- core injection	Active/Passive	- / - / -
- decay heat removal	Active/Passive	- /- / -
- containment isolation and cooling	Active/Passive	- / - / -
- emergency AC supply (e.g. diesels)	Active/Passive	- / - / -
DC Power Capacity (e.g. batteries)		Hours
Events in which Immediate Operator Action is required		e.g. any internal/external initiating events, none
Limiting (shortest) Subsequent Operator Action Time		Hours (that are assumed when following EOPs)
Severe Accident Core Provisions		e.g. no core melt, IVMR, Core Catcher, Core Dump Tank, MCCI
Core Damage Frequency (CDF)		x/reactor-year (based on reference site and location)
Severe Accident Containment Provisions		e.g., H2 ignitors, PARs, filtered venting etc.
Large Release Frequency (LRF)		x/reactor-year (based on reference site and location)
Overall Build Project Costs Estimate or Range (excluding Licensing, based on the Reference Design Site and Location)		
Construction Time (nth of a kind)		Months from first concrete to criticality
Design, Project Mgmt. and Procurement Effort		Person-years (PY) [DP&P]
Construction and Commissioning Effort		PY [C&C]
Material and Equipment Overnight Capital Cost		Million US$(2015) [M&E], if built in USA
Cost Breakdown	%[C&C]/%[M&E]	
- Site Development before first concrete		(e.g. 25/10) (30/40) (20/25) (20/10) (5/15) (----------) (to add up to 100/100)
- Nuclear Island (NSSS)		
- Conventional Island (Turbine and Cooling)		
- Balance of Plant (BOP)		

- Commissioning and First Fuel Loading		
Factory / On-Site split in [C&C] effort		%/% of total [C&C] effort in PY (e.g., 60/40)
Nuclear System		
System Steam Flow Rate		kg/s
System Steam Pressure		MPa
System Steam Temperature		ºC
System Feedwater Flow Rate		kg/s
System Feedwater Temperature		ºC
Reactor Core		
Active Core Height		m
Equivalent Core Diameter		m
Average Linear Heat Rate		kW/m
Average Fuel Power Density		kW/kgU
Average Core Power Density		MW/m3
Outer Diameter Of Fuel Rods		mm

4. Plant Layout, Site Environment and Grid Integration

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Plant layout arrangement

The plant is split into the nuclear and the conventional island. Although in the below solely the reactor building is presented as a part of the nuclear island, this part will be composed of the following buildings : reactor building, auxiliary building and spent fuel building.

The conventional island which is designed to convert energy produced by the reactor is composed of the turbine, reboiler, water treatment system, switch yard, power output and electrical building.

4.1 Site Requirements during Construction
4.2 Site Considerations during Operation
4.3 Grid Integration

5. Technical NSSS/Power Conversion System Designn

SUMMARY FOR BOOKLET

Main design features

Nuclear Steam Supply System

The nuclear steam supply system comprises a reactor pressure vessel featuring a graphite core assembly and metallic internals, a coaxial cross-duct, a steam generator, and a main helium blower with associated piping. The reactor operates at a thermal power of 30 MWt, with helium coolant transporting heat from the reactor core to the steam generator. The coolant temperature within the core increases from 325°C to an average of 750 ºC, with a coolant pressure at the blower discharge of 4.0 MPa. Within the steam generator, heat is transferred to the secondary circuit (water/steam), resulting in the generation of superheated steam at 565 ºC and 13.8 MPa. In the secondary circuit, the steam is split into two main flows: one directed to the steam turbine and the other to the reboiler, with the respective flow ratio depending on the operational mode of the plant. The reboiler subsequently generates in the tertiary circuit (water/steam) final process steam for different applications.

Reactor Core

The active core is comprised of 19 fuel columns arranged in a uniform triangular pitch, assembling into two rings surrounding a central fuel column. Each fuel column contains a stack of six fuel blocks. A single fuel block is a hexagonal prism measuring 36 cm across the flats and 60 cm in height. The fuel block incorporates a grid of fuel holes and coolant channels. There are two types of fuel blocks, i.e. standard blocks and control blocks (with a channel for reserved shutdown system). The active core height is 3.6 m, and its equivalent diameter is 1.66 m.

TRISO fuel with kernels enriched in 235U (HALEU, 10-12 wt.%, UO2) is utilized. Fuel compacts with packing factor of 15-30% are inserted into dedicated holes within the fuel blocks. Between four to six of the fuel holes may be occupied by burnable poison rods (B4C/C) instead of fuel compacts. Helium coolant flows in a top-down direction through the coolant channels in fuel blocks. The active core is radially surrounded by two rings of replaceable reflector blocks. Selected reflector blocks are designed with channels for the control rods. Above and below the active core, there are two layers of replaceable reflector blocks in each location. The external dimensions of the replaceable reflector blocks are as those of the fuel blocks, except for height.

Figure 4. Reactor core configuration.

Reactivity control

The reactor design incorporates two diverse and independent reactivity control systems: Control Rods (CR) and Reserve Shutdown System (RSS). For active operational reactivity control, 18 CR, utilizing B4C as the neutron poison, are employed. These CR are positioned around the active core in the first ring of the replaceable reflector and are inserted into the reactor core vertically from the top. Each CR is driven by its own control rod drive mechanism. In the event that a CR becomes inoperable, the RSS is activated. The RSS employs cylindrical pellets (B4C/C) that are released from hoppers into six channels within the active core. The long-term management of reactivity excess is achieved through the use of burnable poison rods (B4C/C) in fuel blocks.

An additional way to adjust the power being the consequence of the coupling between reactivity and temperature is to reduce the heat uptake by reducing the coolant flow, which leads to matching the reactor power to the altered flow.

Reactor pressure vessel and internals

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

Reactor Coolant System and Steam Generator

The coolant from the upper mixing chamber is directed through the top reflector to the active core, then flows through the lower reflector and enters the lower mixing chamber, which is located between the core supporting structures and the bottom reflector. From there, the coolant is then routed to the cross-duct and

exits the RPV. Hot helium through the inner duct of the cross-duct goes to the steam generator (SG). After extracting heat from the helium in the SG, the cooled medium is directed to the external main helium blower. From the blower, the helium flows back to the SG and subsequently moves through the outer SG shell, a then through the outer cross-duct pipe to the RPV, where it enters the riser channels leading to the upper mixing chamber.

Pressuriser

There is no pressuriser in this technology. Pressure in the primary cooling circuit is controlled by helium purification and supply systems.

Primary pumps

The helium coolant flow is driven by a single-stage, magnetic-bearing main blower. The blower is a standalone device connected to the steam generator via pipes. Both the blower and the piping are part of the reactor coolant pressure boundary. The cooled helium from the steam generator is directed to the main blower. From there, the high-pressure helium is routed back through the steam generator and cross-duct to the reactor pressure vessel.

Instrumentation and control systems

The I&C systems used in the HTGR-POLA, due to their specifications, can be divided into the levels of architecture composed of process instrumentation, logic controllers and supervision and control system. These levels correspond to different overall design requirements. All instrumentation and control systems are designed to withstand the environmental conditions existing in the rooms or locations in which they are installed, so that they perform their design functions correctly. The system structure meets the operational, design and safety licensing objectives. The visualisation and control level consists of the workstations, Main Control Room and the Reserved Shutdown Station. The I&C automation level is based on three individual subsystems. In the event of a DBE, DEC or Limiting Conditions of Operation exceeding, the Reactor Protection System serves as the overriding function of all I&C systems, thus ensuring the safe shutdown of the reactor. The Safety Automation System is utilised to safeguard the pivotal components integral to the heat transfer from the primary coolant circuit. These include the steam generator, the residual heat removal system and the insulation of confinement system. The Process System is employed for the purpose of regulating all settings associated with the technological infrastructure of the building. Its function is to integrate all systems with which the building is equipped in order to maintain the process in accordance with plant technical parameters.

The ultimate objective is to reduce the probability of operator error and their subsequent impact on the safety of the facility. In order to achieve this goal, the principles of appropriate ergonomic design are implemented, and operators are afforded as much time as is reasonably necessary to react.

5.1 Primary Circuit
5.2 Reactor Core and Fuel
5.3 Fuel Handling
5.4 Reactor Protection
5.5 Secondary Side
5.6 Containment/Confinement
5.7 Electrical, I&C and Human Interface
5.8 Unique Technical Design Features (if any)

6. Technology Readiness

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Testing conducted for design verification and validation

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

6.1 Deployed Reactors
6.2 Reactors under Licensing Review
6.3 Reactors in the Design Stage

7. Safety Concepts

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Safety features

Engineered safety system approach and configuration

Unique inherent safety properties of HTGR, which can be categorized as a safety measure are: a strong negative fuel, moderator, and reflector temperature coefficient of reactivity and the large heat capacity of bulk graphite in the core. An increase in fuel temperature leads to a decrease in the number of fission reactions and consequently in a decrease of reactor power. In potential accident conditions, when the temperature rises significantly, there will be a spontaneous shutdown of the reactor.

There is no safety injection of coolant to the reactor coolant system. Under accident conditions, the heat from the reactor core is dissipated by conduction, natural convection, and thermal radiation to the reactor cavity and directly to the components of the reactor cavity cooling system.

Safety approach and configuration to manage accident conditions

According to the defence in depth concept, the safety of the HTGR-POLA design is based on the layered approach. The approach to the accident management lies in the prevention and its mitigation through the use of active and passive systems with great emphasis of the inherent reactor features. Those combined provide the assurance of the high risk (more severe consequence) events to be put in the low probability category. The prevention of the accidents focuses on the first two layers of defence concept. The level 1 is achieved by the use of the high quality and inherently safe materials – the TRISO fuel, as well as operating with large safety margins. For the level 2 features and system the I&C system is capable of detecting and acting upon departure from safe operation for variety of Postulated Initiating Events and restore the required parameters levels. The safety approach to the management of the accident conditions is done through the mitigation. The mitigation is achieved by the activation of the level 3 of defence in depth layer

systems, like the passive safety system for the core heat removal and maintaining of the physical barrier integrity. The Reactor Cavity Cooling System reduces the accident consequences by controlling the RPV wall and core temperature, and ensures the increased ability to retain fission products in the primary coolant system. Additional feature that deals with the accident conditions exceeding the design basis, is the SG drain system, dedicated exclusively to the water ingress conditions, that prevents the chemical attack of the water (steam) on the core graphite structures. The two last defence lines (4 and 5), are focusing on the accident management, the limited operator action requirements, ability of the RCCS secondary coolant addition and emergency preparedness with off-site emergency response.

Containment system

For HTGR-POLA, the containment system is defined as a reactor functional containment, comprising a series of protective barriers that collectively limit the transport and release of radionuclides into the environment throughout the spectrum of normal operating conditions, anticipated operational occurrences (AOOs), and accident conditions. These barriers include, in order: the fuel kernel, the TRISO layers, the graphite matrix of the fuel compact, the graphite blocks, the primary system boundary, and the confinement in reactor building. The primary barrier for the retention of fission products is the TRISO fuel itself, while the structure of the confinement plays a comparatively lesser role. Radionuclide retention occurs at the source – within the fuel. The subsequent barriers serve to limit and control the release of radionuclides into the environment, ensuring that design conditions critical to safety are not exceeded in any of the reactor states.

Spent fuel cooling safety approach / system

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

7.1 Safety Philosophy and Implementation
7.2 Transient/Accident Behaviour

8. Fuel and Fuel Cycle

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Fuel cycle approach

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

Waste management and disposal plan

Radioactive waste management and disposal plan includes solid, liquid, gas wastes, and spent nuclear fuel. Spent nuclear fuel is to be temporary stored on site. After ~2 years of primary cooling in the storage

located in the reactor building is to be transferred to the temporary dry storage in the spent nuclear fuel building. Spent nuclear fuel building is designed for 60 years of reactor life. Irradiated reflector graphite blocks are to be temporarily stored in the spent nuclear fuel building. Final disposal would be depend on national plan.

8.1 Fuel Cycle Options
8.2 Resource Use Optimization
8.3 Unique Fuel/Fuel Cycle Design Features

9. Safeguard & Physical Security

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9.1 Safeguards
9.2 Security
9.3 Unique Safeguards and/or Security Features

10. Project Delivery & Economics

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10.1 Project Preparation and Negotiation
10.2 Construction and Commissioning
10.3 Operation and Maintenance