

# Update on the IAEA activities on HTGR technology development

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# Historical development of HTGR



	Test HTGRs				Prototype HTGRs		
	Dragon	AVR	HTTR	HTR-10	Peach Bottom	FSV	THTR-300
Country	UK (OECD)	Germany	Japan	China	USA	USA	Germany
Period of operation	1963-76	1967-88	1998-present	2000-present	1967-74	1976-89	1986-89
Reactor type	Tube	Pebble	Prismatic	Pebble	Tube	Prismatic	Pebble
Thermal power, MWt	21.5	46	30	10	115	842	750
He coolant outlet temp., °C	750	950	950	700	725	775	750
Coolant pressure, MPa	2	1.1	4.0	3.0	2.25	4.8	3.9
Electrical output, MW	-	13	-	2.5	40	330	300
Process heat output, MW	-	-	10	-	-	-	-
Process heat temp., °C	-	-	963	-	-	-	-
Core power density, W/cm <sup>3</sup>	14	2.6	2.5	2	8.3	6.3	6.0
Fuel particle	UO <sub>2</sub>	(Th/U, U)O <sub>2</sub> , C <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	ThC <sub>2</sub>	(Th/U, Th)C <sub>2</sub>	(Th/U)O <sub>2</sub>
Kernel coating	TRISO	BISO & TRISO	TRISO	TRISO	BISO	TRISO	BISO

HTR-PM in Shidao Bay, Shandong Province of China, the first modular HTGR nuclear power plant entered commercial operation



HTR-PM  
6 Dec 2023

Plant electrical power, MWe	200
Core thermal power, MW	250
<b>Number of NSSS Modules</b>	<b>2</b>
Core outlet temperature, °C	750
Core inlet temperature, °C	250
Steam pressure, MPa	13.2
Steam temperature, °C	567

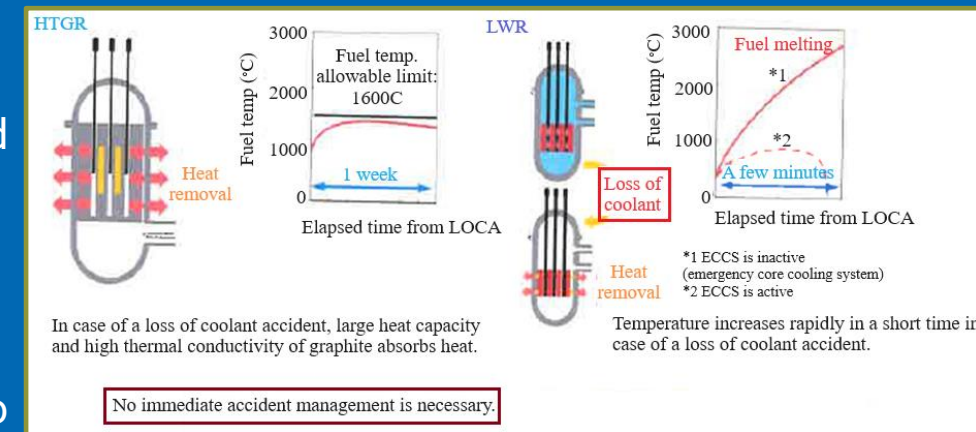
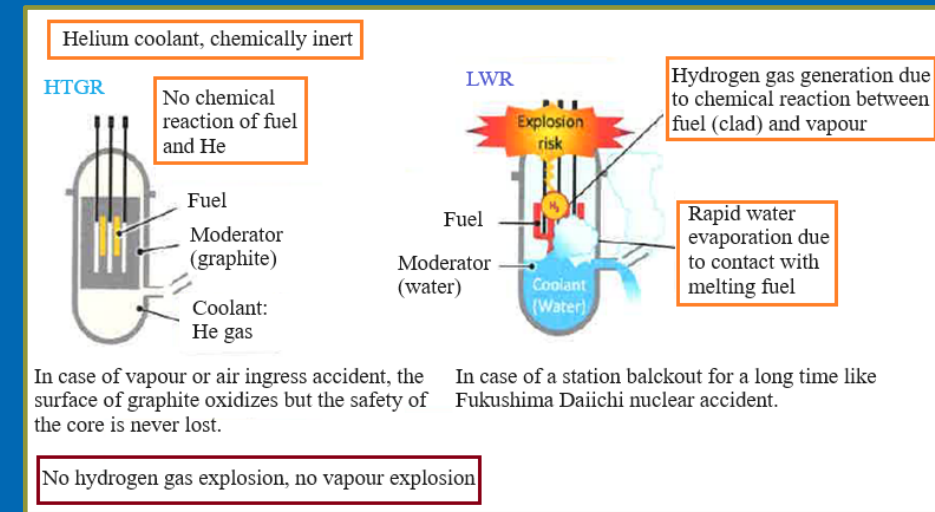
- Extensive operating experience
- Mature technology ready for commercial deployment (in next decade)
- Wealth of know-how available
- Interest from new-comer countries
  - Indonesia (BATAN) experimental power reactor

# Overview of HTGR features

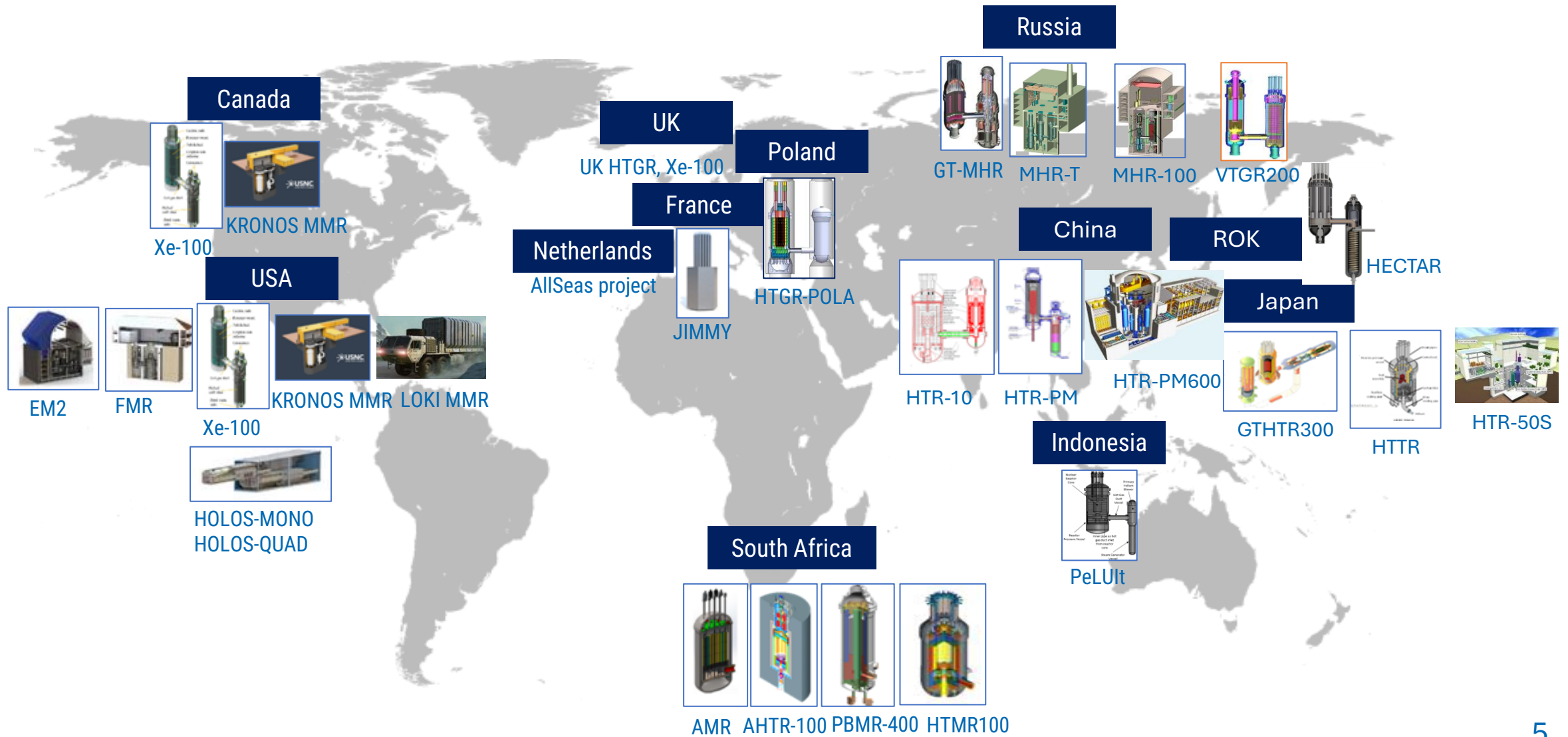
- Can supply high temperature heat energy of 750-950°C
- Enhanced or higher thermal efficiency
- The high temperature heat can be used for a variety of applications (e.g. production of synthetic fuels, methanol and hydrogen)
- Safety demonstrated in experimental reactors
- Uses spherical fuel coated with ceramics (C, SiC)
- Inert He gas as coolant
- Graphite as moderator
- Can achieve higher fuel burnup (>80 000 up to 200 000 MWd/t)
- Types: pebble-bed core (HTR-PM, China) and prismatic core (HTTR, Japan)

# Safety features of HTGRs

- Minimize the potential for severe accidents and their consequences
- No core meltdown accident or core damage (no possibility for massive FP release)
- No need for active decay heat removal / multiple trains of cooling capabilities
- Low power density (~ 30 times lower than LWRs)
- No hydrogen gas explosion, no vapor explosion
- The coated fuel particles can bear very high temperature condition over 2200°C
- The fuel system has a negative temperature coefficient
- HTGR can be designed that the fuel temperature does not exceed 1600°C in any accident to prevent fuel damage
- Graphite moderator has a very high thermal inertia
- Most transients are slow (develop over hours and days)
- In a loss of forced cooling scenario, the decay heat of fuel transfers to the reactor vessel through the core graphite structure slowly by thermal conduction and emission.

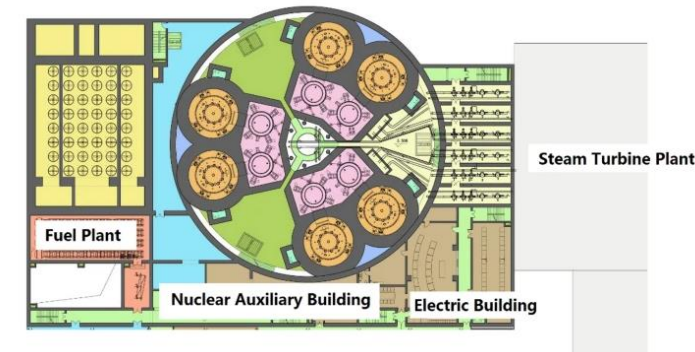
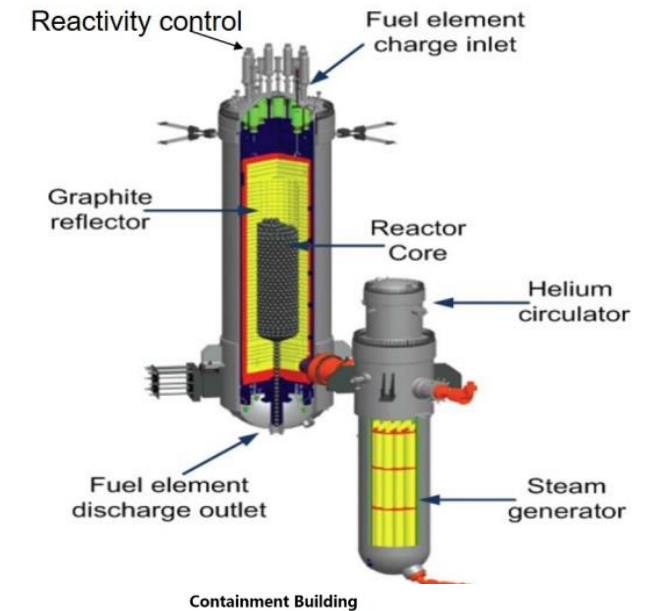


# Global developments on HTGR-SMR technology



# China

- HTGR basic research started in China in 1980s and in 1992 the government approved to construct HTR-10; its first criticality was achieved in 2000 and the full operation started in 2003.
- In 2006, HTR-PM project was listed as a major project in the national R&D programme for science and technology.
- In 2012, first concrete for HTR-PM was poured, 2016 first RPV was installed. Fuel loading started in August 2021 and the connection to the grid was done in December 2021. The commercial operation started in December 2023.
- An industrial scale fuel plant, capacity 300,000 fuel elements/ yr was built in Baotou. Construction was started in 2013 and production in 2016.
- **April 2024: China's HTR-PM started heating project operation**
- **HTR-PM600 will feature 6 modules connected to one single turbine to produce 650 MWe, building on the experience from the demonstration plant. Based on requirements from particular utilities, larger units with more modules can also be built that would enhance the economic competitiveness.**
  - Since 2013, the HTR-PM600 has been jointly developed by INET and the Chinergy Co., Ltd of the China National Nuclear Corporation (CNNC).
  - The design of a standard HTR-PM600 was completed in 2016
  - HTR-PM600 vs HTR-PM Demo Plant: same safety features, same major components, same parameters
  - main tasks are to optimize manufacturing processes and standardize production in batches to minimize costs
  - SNERDI is manufacturing the RPVs → importance of preparation of long lead items
- **Another version is HTR-PM600S for cogeneration having HTR-PM modules with some minor changes.**
- **August 2024: the Jiangsu Xuwei nuclear heat project was approved to build 2 Hualong 1 units and 1 HTR-PM600S unit**
- Further development of HTR-PM for elevated temperatures is on-going (900-950C) looking at extending qualification of fuel performance and key component development; hydrogen production studies, first coupling option is steam methane reforming



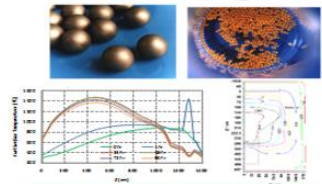


# Specifications and history of HTGR technology development in China

Source: CNNC, Chinergy Co. Ltd at IAEA International Conference on Small Modular Reactors and their Applications (2024)

## The development of HTGR in China can be divided into four stages

- Fuel
- Reactor physics
- Thermohydraulics
- Reactor structure etc.



Fundamental research (1980s)

- Know-how
- verified Inherent safety features
- Fuel tech
- Thermal application R&D



Test reactor (HTR-10, 1986)

- Fuel quantity production
- Modular reactor
- Engineering validation
- Equipment manufactures capacity
- Safety review tech
- Economic feasibility



Demonstration plant (HTR-PM, 2006)

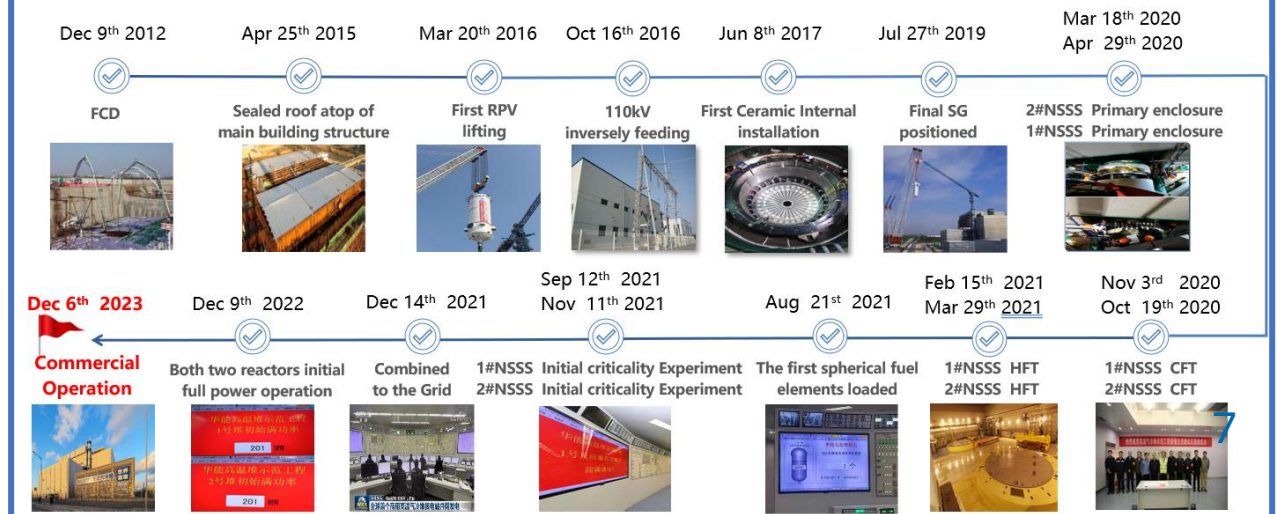
- Inherent safety features
- Mature tech
- Modular and quantity production
- Economic competitiveness
- Achievable multi-use



Commercial plant (HTR-PM600, 2014)

Plant electrical power, MWe	200
Core thermal power, MW	250
Number of NSSS Modules	2
Core outlet temperature, °C	750
Core inlet temperature, °C	250
Steam pressure, MPa	13.2
Steam temperature, °C	567

## Milestones of Construction (HTR-PM, Shidao Bay, Shandong)

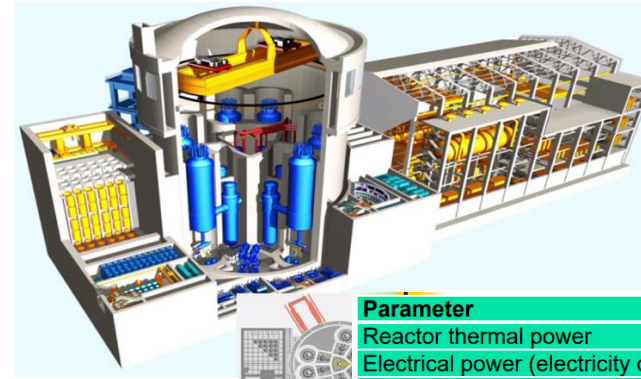


<b>Development Milestones HTR-PM</b>	
<b>2001</b>	Launch of commercial HTR-PM project
<b>2004</b>	Standard design of HTR-PM started
<b>2006</b>	HTR-PM demonstration power plant approved as one of the National Science and Technology major projects
<b>2006</b>	Huaneng Shandong Shidaowan Nuclear Power Co. Ltd., the owner of HTR-PM, established by the China Huaneng Group, the China Nuclear Engineering Group Co. and Tsinghua University
<b>2006-2008</b>	Basic design of HTR-PM completed
<b>2009</b>	Assessment of HTR-PM PSAR completed
<b>2012</b>	First pour of concrete of HTR-PM
<b>2013</b>	Fuel construction plant started
<b>2014</b>	Qualification irradiation tests of fuel elements completed
<b>2015</b>	Civil work of reactor building finished
<b>2016</b>	RPV and core barrel delivered, installation on main components
<b>2017</b>	Fuel plant achieved expected production capacity
<b>2020</b>	Startup commissioning test of primary circuit
<b>Aug 2021</b>	Operation license
<b>Sept 2021</b>	First criticality
<b>Dec 2021</b>	Grid connection
<b>Dec 2022</b>	Both 2 reactors initial full power operation
<b>Dec 2023</b>	Commercial operation

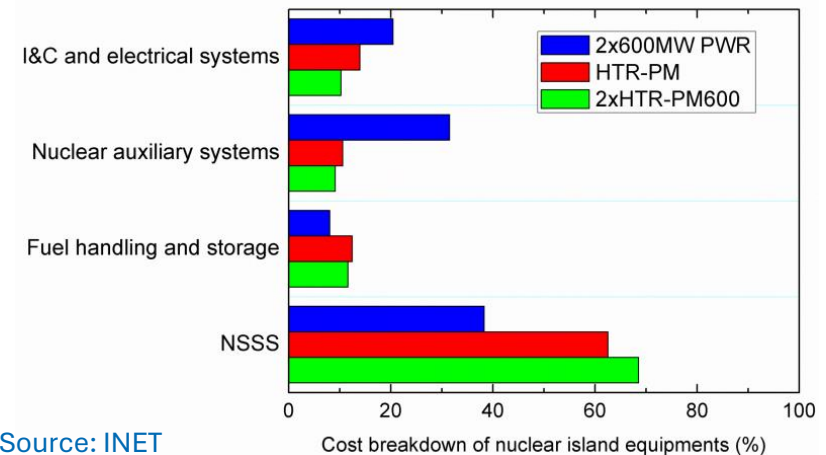
# HTR-PM600

A complete multi-module power plant design

- Design characteristics of HTR-PM as reference
- 6 modules + one steam turbine, 650 Mwe
- Cogeneration of electricity and steam
- Adoption of experience feedback from the demo plant
- HTR-PM600 vs HTR-PM demo plant:
  - Same safety features
  - Same major components
  - Same parameters
- Same site footprint and same reactor plant volume vs domestic PWR
- Proven technologies from HTR-PM:
  - Fuel element
  - Manufacture (RPV, SG, internals, control rod system, He circulator, fuel handling, SF storage)
- HTR-PM600 takes full advantages of HTR-PM outcomes (design, manufacture, civil work, commissioning, licensing, project management)



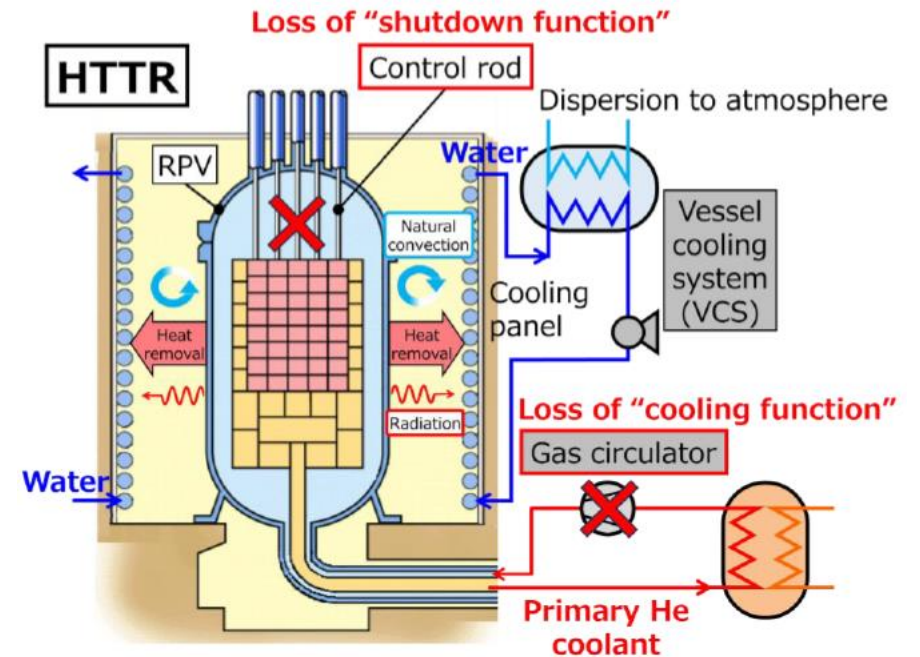
Parameter	Unit	Value
Reactor thermal power	MW	6 x 250
Electrical power (electricity only)	MW	650
Designed life time	a	40
Working mode	-	Cogeneration
Maximum process steam temperature	°C	550
Primary helium pressure	MPa	7
Core inlet temperature	°C	250
Core outlet temperature	°C	750
Fuel type	-	Coated particle spherical fuel
Enrichment of fresh fuel	%	8.5
Number of fuels per reactor core	-	~420,000
Average burn-up	GWd/tU	90
Main feed water temperature	°C	205
Main steam temperature	°C	566
Main steam pressure	MPa	13.24
Containment type	-	Low pressure ventilated
Turbine type	-	Super high-pressure steam turbine



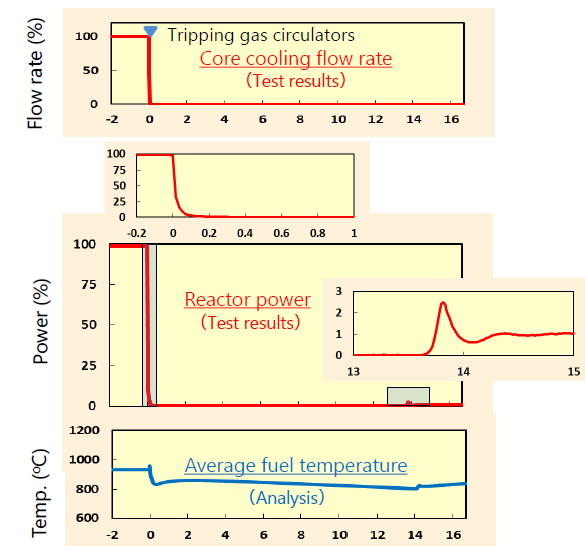
Source: INET

# Japan

- Japan's Green Growth Strategy for achieving carbon neutrality in 2050 was issued in December 2020. As part of the supporting policy, the *Government is looking at international safety demonstration for HTGR technology utilizing HTTR and facilitates the technology development for hydrogen production using HTTR.*
- The LOFC tests as the safety demonstration of HTGR were completed.
- Design and safety analysis of the HTTR hydrogen production system is ongoing.
- Control technology development test of IS process is ongoing.
- Demonstration of coupling of hydrogen plant and HTGR is expected by 2040.
- The draft application documents for licensing the HTTR hydrogen production project was completed in 2024. JAEA will submit the application documents in the early 2025 to Nuclear Regulatory Authority. The hydrogen production test using HTTR is expected by 2030.
- JAEA collaborates with Mitsubishi Heavy Industries (MHI) for basic design, detailed design, manufacturing, and construction for the **domestic demonstration HTGR**. Reactor power will be range from 150MW-250MW, supplying above 800C to a hydrogen plant.
- JAEA participates in overseas demonstration reactor projects through **Japan-UK HTGR collaboration**.
- In March 2024, the loss of forced cooling test (with all helium gas circulators stopped at 100%) was conducted.
- JAEA holds the HTGR fuel design technology and the Japanese private company, NFI, holds the HTGR fuel manufacturing technology
- JAEA plans to **establish HTGR fuel manufacturing technology in the UK** with a view to commercial HTGRs, and to use the UK's fuel as an option for procuring fuel for the Japan demonstration HTGR.
- In April 2024, NNL and JAEA signed a collaboration memorandum and license arrangement for UK Coated Particle Fuel Programme



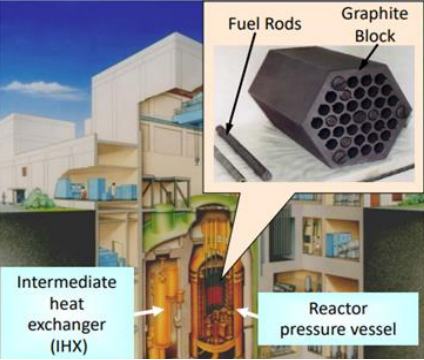
Loss of forced cooling test (All HGC stopped at 100% (30MWt)) March 2024



# Specifications and history of HTTR

Source: Shibata, T., Plans and Status toward SMR Development and Deployment in Japan, in IAEA TWG-SMR meeting (2019)


**HTTR(High Temperature Engineering Test Reactor)**  
Graphite-moderated and helium-cooled VHTR



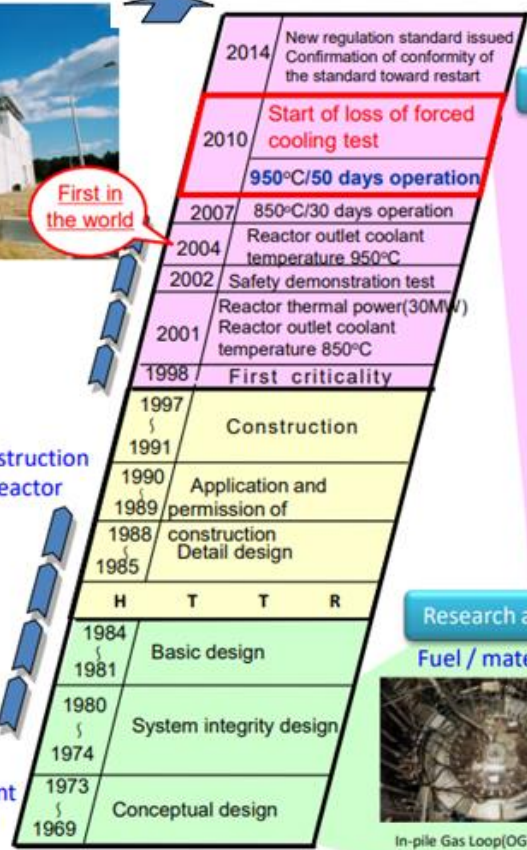
**Major specification**

Thermal power	30 MW
Fuel	Coated fuel particle / Prismatic block type
Core material	Graphite
Coolant	Helium
Inlet temperature	395°C
Outlet temperature	950°C
Pressure	4 MPa

**First criticality : 1998**  
**Full power operation : 2001**  
**50 days continuous 950°C operation : 2010**  
**Loss of forced cooling test at 9MW : 2010**



**Proposal for prototype commercial system**



**Research and development**

- 1969-1973: Conceptual design
- 1974-1980: System integrity design
- 1981-1984: Basic design

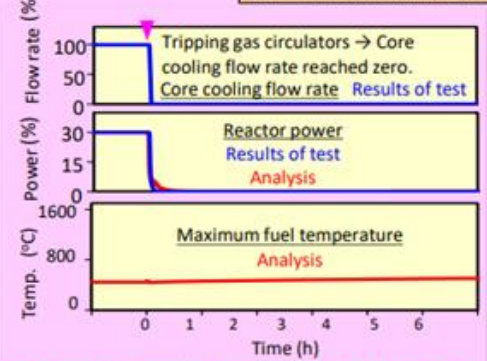
**Construction of Reactor**

- 1985-1988: Detail design
- 1989-1990: permission of construction
- 1991-1997: Construction

**Operation and Safety**

- 1998: First criticality
- 2001: Reactor outlet coolant temperature 850°C
- 2002: Safety demonstration test
- 2004: Reactor outlet coolant temperature 950°C
- 2007: 850°C/30 days operation
- 2010: 950°C/50 days operation
- 2010: Start of loss of forced cooling test
- 2014: New regulation standard issued

**Safety demonstration tests** (OECD/NEA project)



Tripping gas circulators → Core cooling flow rate reached zero.  
 Core cooling flow rate Results of test  
 Reactor power Results of test  
 Maximum fuel temperature Analysis

No CR reactivity control. No core cooling. Reactor is kept stable naturally with only physical phenomena.

**Fuel / material**: In-pile Gas Loop(OGI-1)

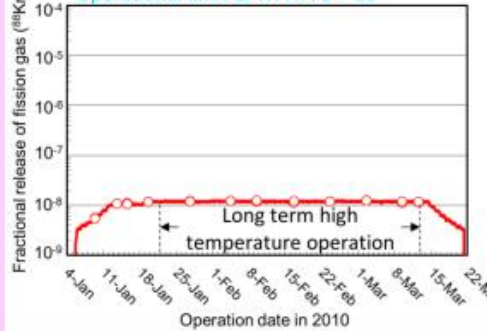
**Reactor physics**: Very High Temperature Reactor Critical Assembly (VHTRC)

**Thermo-hydraulics**: HENDEL (Helium Engineering Demonstration Loop)

**Long term high temperature operation**

**<Integrity of fuel coating>** 950°C, 50days

Operational limit of HTTR :  $1 \times 10^{-4}$

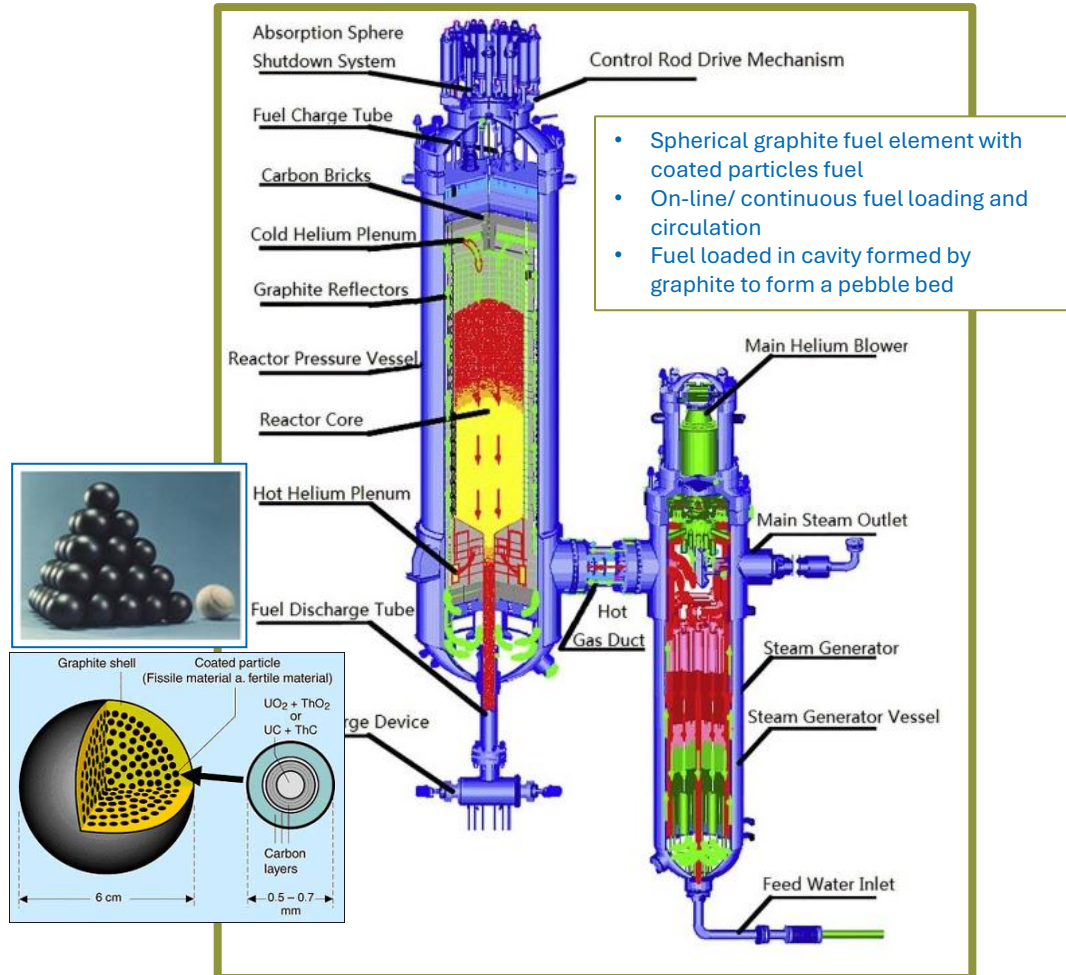


Fractional release of fission gas ( $^{88}\text{Kr}$ )  
 Operational date in 2010

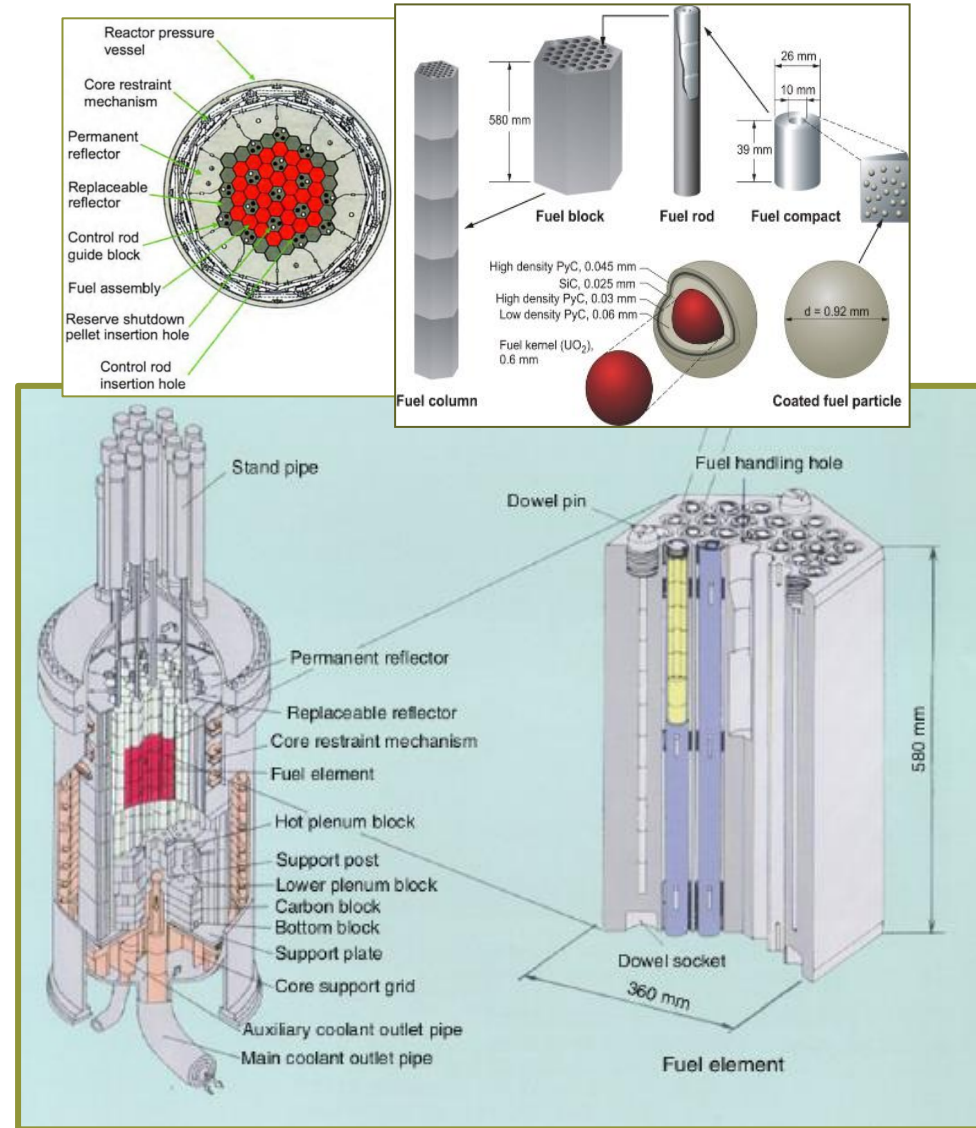
➤ R/B of  $^{88}\text{Kr}$  : 4 orders of magnitude less than the operational limit

# HTR-PM & HTTR and their fuel elements

## HTR-PM and fuel element



## HTTR and fuel element



# HTGR and non-electric applications - Highlights

## Canada

- OPG and X-Energy have signed a framework agreement to pursue opportunities to deploy Xe-100 SMRs for electricity and **high-temperature steam for industrial applications** in Ontario and Canada (e.g., oil sands, petrochemical).
- CNL is conducting a series of R&D activities on HTGR on materials, hydrogen production, modelling, nuclear fuel.

## France

- Jimmy HTGR 60 MWt, 450C is developed to support **heat applications**;
- 2024 - order contract with Toyo Tanso for supply of graphite, contract with URENCO for the supply of 10% HALEU (FOAK);
- First client: Sugar factory Cristanol, Bazancourt; 450C steam; footprint of 6 000 m<sup>2</sup> (100 x 60m)
- Building permit for TRISO fabrication facility will be submitted in 2026; licensing pending for fuel assembly facility

## Indonesia

- Target is to acquire 250 MWe from nuclear power in 2033;
- PeLUlt-40 (40MWt) for electricity and **steam for industrial applications**, including hydrogen production

## Poland

- HTGR-POLA (30-40 MWt, 750C) - technology demonstrator for **industrial applications**, aiming to supply superheated steam to address Polish industrial needs
- Completed its basic design phase, entering into the detailed design and licensing phase expected to end by 2028
- The commissioning of the reactor is aimed to be in 2033.

## Republic of Korea

- KAERI has developed the key technologies of VHTR for hydrogen production since 2017: VHTR code V&V and code improvements, materials characterization and lab scale TRISO fabrication, helium experimental loop, long term life prediction for IHX, lab scale sulphur-iodine cycle, coupling of VHTR and HTSE for **hydrogen production** (HTSE module fabrication – 6 kW, HTSE performance test, coupled analysis VHTR-HTSE)
- Linking HTGR deployment to industrial decarbonisation strategies, including **hydrogen-based steelmaking**, with strong involvement of industrial end-users (e.g. POSCO)

## Russian Federation

- HTGR technology development in Russia started in the 70s for electricity and **heat generation** for ammonia production. Several HTGR designs were developed: VG-400 (1060 MWt), VGM (200 MWt), VGM-P (215 MWt), GT-MHR (600 MWt), MHR-T (600 MWt), RDE (10 MWt) and since 2021, **VTGR-200** (200 MWt).
- The investment stage of the project for a VTGR-200 reactor plant construction (prismatic core); awaiting the approval of the justification of investments in 2025; **hydrogen production** process intended is steam methane reforming; the FOAK NPP expected to enter commercial operation in 2035

# HTGR and non-electric applications - Highlights

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## South Africa

- PBMR project, established in 1999, was placed under “care and maintenance” in 2010 but is now being considered for restart
- Cabinet decisions in 2025 facilitates reopening the PBMR fuel development laboratories, manufacture HTR fuel for export, and collaborate internationally on R&D and **non-electric applications (hydrogen, process heat, desalination)**
- STL Nuclear develops HTGR and TRISO fuel technologies, including the HTMR100 and HTMR25 pebble bed reactor designs, with international collaborations

## United Kingdom

- UK government funding is mainly directed towards advanced reactor systems, especially HTGRs
- The UKJ-HTR (UK-Japan High Temperature Reactor) Demonstration Project is a central part of the AMR RD&D programme, developed in partnership with JAEA and METI Japan.
- The project aims to deliver a prismatic block-type HTGR with 150–250 MWth, helium coolant, 750–950°C outlet temperature, and advanced fuel (up to 19.75% enrichment). The design draws on Japanese HTGR experience (HTTR, HTR50S, GTHTR300) and is tailored for UK requirements
- Extensive engagement with **end users (industry, energy clusters)** identified barriers: integration with existing systems, energy pricing, capital costs, commercial flexibility, and investor confidence
- JAEA>NNL collaboration on a pilot-scale TRISO fuel manufacturing capability; Urenco to build a HALEU enrichment facility by 2031

## United States

- DOE 50/50 cost-share demonstration award for XE Xe-100 over 7 years to build a 2-unit Xe-100 demonstration plant
- X-energy’s XE-100 is supported by DOE’s ARDP (\$1.23B), with four units planned for Dow Chemicals in Texas (Project Long Mott, by 2029) and further deployments for **Amazon datacentres** and Energy Northwest in Washington State. International projects include up to 12 units in Hartlepool (UK) and feasibility studies in Alberta (Canada).
- The US TRISO fuel qualification program aims for completion of fuel qualification test by 2027.
- Selection, irradiation, characterization and qualification of existing nuclear grade graphite is on-going; the graphite data are included in the NDMAS - DOE-ART Graphite
- The Nuclear Reactor Pilot Program (NRPP) launched in August 2025 allows SMR developers to construct and operate test reactors outside national labs, bypassing NRC licensing for non-commercial tests. 10 companies were selected, including Radiant Industries, Antares Nuclear, Valar Atomics, and others

# SMRs details captures in ARIS

## ADVANCED REACTOR INFORMATION SYSTEM

The Advanced Reactor Information System (ARIS) database is designed and maintained by the IAEA and contains design descriptions of evolutionary and innovative nuclear reactors

Explore →

### Total 119 Reactors/ 21 Member States

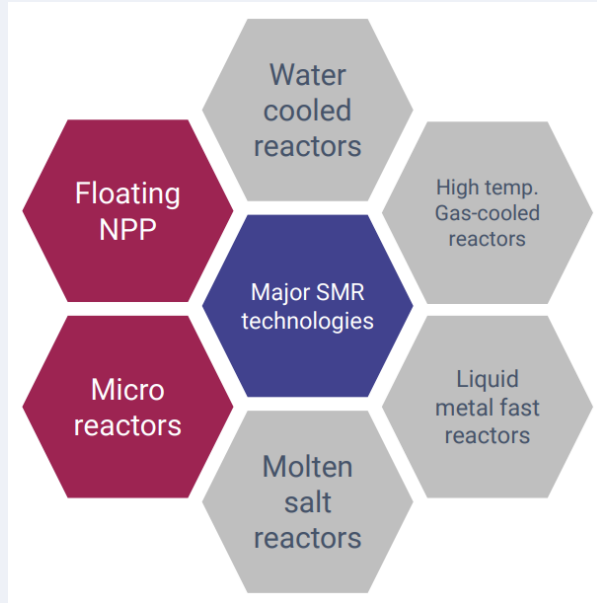
- High Temperature Gas Cooled 19
- Gas Fast Reactors 4

Web-accessible database that provides Member States with balanced, comprehensive and up-to-date information about advanced nuclear plant designs and concepts

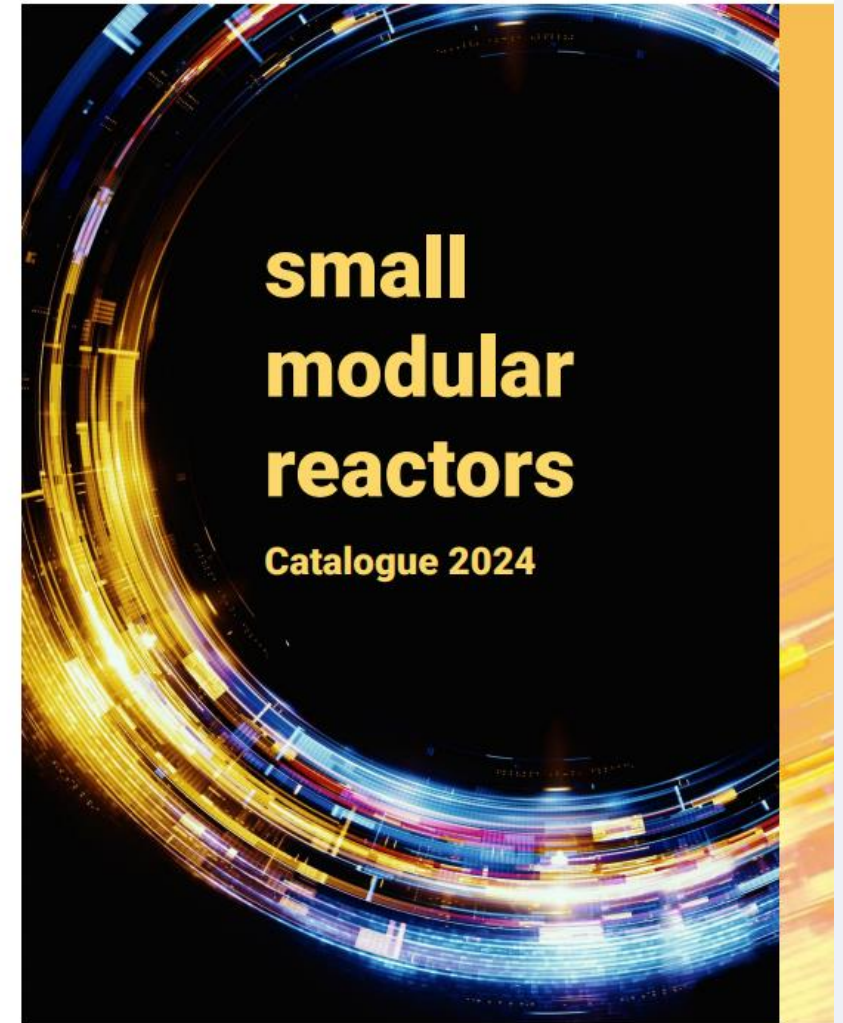
WCRs, GCRs, FRs, MSR, SMRs, Microreactors

A tool for Member States at various stages of nuclear power development, offering data on reactor designs, including evolutionary and innovative concepts, to support informed reactor technology assessments

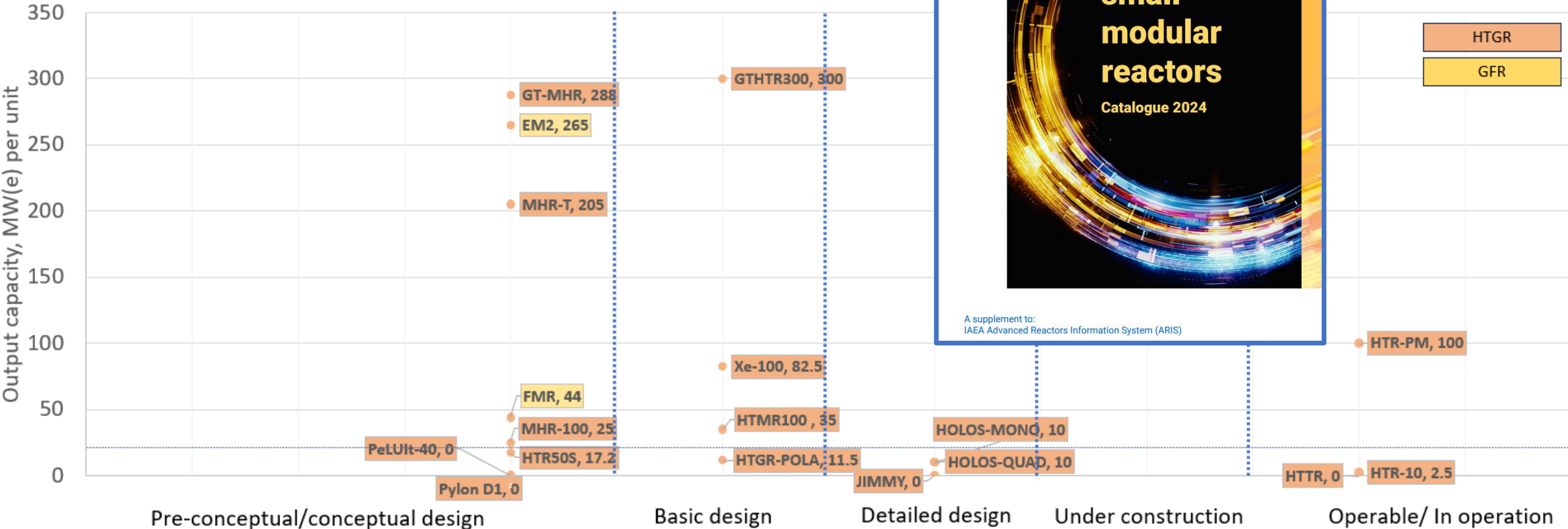
# ARIS SMR Catalogue 2024



- **Available for download from this link:**  
[https://aris.iaea.org/publications/SMR\\_catalogue\\_2024.pdf](https://aris.iaea.org/publications/SMR_catalogue_2024.pdf)
- A supplement to ARIS Database, published biannually since 2012 with the objective to provide Member States with a concise overview of the latest status of SMR designs.
- The content is provided by the responsible institute or organization and is reproduced, with permission.
- *For the 2024 SMR Catalogue, although close to 100 designs could have been listed, only active designs with demonstrated sustained development were reported in the catalogue.*



# SMR2024 Catalogue



Note: The value displayed with the design refers to the output capacity in MW(e) per unit.

# Various HTGR type SMR design information

	Thermal/ electrical capacity, MWt/MWe	Design status	Core type	Core inlet/outlet coolant temp (°C)	NSSS Operating Pressure (primary/secondary) (MPa)	Fuel enrichment	Core Discharge Burnup (GWd/t)	Design life (years)	Refuelling cycle (months)	Fuel cycle requirements/ Approach	Distinguishing features
<b>CHINA</b>											
<b>HTR-PM (Tsinghua University)</b>	2 × 250 / 210	<b>In operation</b>	Pebble bed	250 / 750	7 / 13.25	8.5%	90	40	On-line refuelling	LEU, open cycle, spent fuel intermediate storage at the plant	Inherent safety, no need for offsite emergency measures
<b>HTR-10 (Tsinghua University)</b>	10 / 2.5	<b>In operation</b>	Pebble bed	250 / 700	3 / 4	17%	80	20	On-line refuelling	Once through fuel cycle	To verify and demonstrate the technical and safety features
<b>FRANCE</b>											
<b>JIMMY (JIMMY ENERGY SAS)</b>	10-20 / N/A	Detailed design	Prismatic	300 / 700	1.5 / 3.0	19.5%		10-20	No refuelling	No refuelling, no on-site storage	Compact, low-weight design; simplicity of manufacturing, flexibility of operation; intrinsic passive safety
<b>INDONESIA</b>											
<b>PeLUIt/RDE (BRIN)</b>	40 MWt, 10 MWt	Conceptual design	Pebble bed	250 / 750	3 / 6	17%	80	40	On-line refuelling	Open cycle	No need for offsite emergency measures
<b>JAPAN</b>											
<b>GTHTR300 (JAEA Consortium)</b>	<600/ 100- 300	Basic design	Prismatic	587-633 / 850-950	7 / 7		120	60	48	Open cycle	Cogeneration of hydrogen, process heat, steelmaking, desalination, district heating
<b>HTTR (JAEA)</b>	30 MWt	<b>In operation</b>	Prismatic	395 / 850 (950 max.)	4	3 – 10 (6 avg.)	22-33	20	660 equivalent full power days		Safety demonstration test

# Various HTGR type SMR design information (cont.)

	Thermal/ electrical capacity, MWt/MWe	Design status	Core type	Core inlet/outlet coolant temp (°C)	NSSS Operating Pressure (primary/secondary), (MPa)	Fuel enrichment	Core Discharge Burnup (GWd/t)	Design life (years)	Refuelling cycle (months)	Fuel cycle requirements/ Approach	Distinguishing features
<b>RUSSIAN FEDERATION</b>											
<b>GT-MHR (JSC "Afrikantov OKBM")</b>	600 / 288	Detailed design	Prismatic	490 / 850	7.2 / -		100-720	60	25	Standard LEU or WPU / No recycling	Inherent safety characteristics; no core melt
<b>MHR-T Reactor (JSC "Afrikantov OKBM")</b>	4 × 600 / 4 × 205.5	Conceptual design	Prismatic	578 / 950	7.5 / -		125	60	30	Standard LEU / No recycling;	Multi-module HTGR dedicated to hydrogen production / high temperature process heat application
<b>MHR-100 (JSC "Afrikantov OKBM)</b>	215 / 25 – 87	Conceptual design	Prismatic	490 – 553 / 795 – 950	4 – 5	< 20%		60		Open cycle; Pu and Th cycle also possible	Cogenerations of electricity, heat and hydrogen; high- temperature heat supply to oil refinery plant
<b>SOUTH AFRICA</b>											
<b>AHTR-100 (Eskom Holdings SOC Ltd.)</b>	100 / 50	Detailed design	Pebble bed	406 / 1200	9	LEU or WPU		40	On-line refuelling	Initially open cycle	Inherent safety characteristics; no core melt;
<b>PBMR®-400 (PBMR SOC Ltd.)</b>	400 / 165	Basic design	Pebble bed	500 / 900	9	9.6% LEU or WPU		40	On-line refuelling	Uranium once through	Inherent safety characteristics; no core melt
<b>HTMR100 (STL Nuclear (Pty) Ltd.)</b>	100 / 35	Basic design	Pebble bed	250 / 750	4/16	10%	80-90	40	On-line refuelling	Different fuel cycles, including mixtures of Th and Pu, Th and U	No core meltdown, no active engineered safety systems

# Various HTGR type SMR design information (cont.)

	Thermal/ electrical capacity, MWt/MWe	Design status	Core type	Core inlet/outlet coolant temp (°C)	NSSS Operating Pressure (primary/ secondary), (MPa)	Fuel enrichment	Core Discharge Burnup (GWd/t)	Design life (years)	Refuelling cycle (months)	Fuel cycle requirements/ Approach	Distinguishing features
<b>UNITED STATES</b>											
<b>EM2 (General Atomics)</b>	500 / 265	Conceptual design		550 / 850	13.3	7.7%	130	60	360	Open fuel cycle	Fast reactor, convert and burn
<b>FMR (General Atomics)</b>	100 / 50	Conceptual design		509 / 800	7	19.75%	100	60	96	Open fuel cycle	Silicon carbide composite cladding
<b>Xe-100 (X- energy, LLC)</b>	200 / 82.5	Basic design	Pebble bed	260 / 750	6.0 / 16.5	15.5 %	165	60	On-line refuelling	Uranium once through (initially)	No core meltdown
<b>SC-HTGR (Framatom e Inc.)</b>	625 / 272	Preliminary design	Prismatic	325 / 750	6 / 16	14.5-18.5%	165	80	½ of the core replaced every 18 months	LEU once- through fuel cycle / reprocessing options for later consideration.	400 m EPZ; underground construction
<b>CANADA, UNITED KINGDOM AND UNITED STATES</b>											
<b>STARCORE (StarCore Nuclear)</b>	35 / 14	Pre- conceptual / conceptual design	Prismatic	280 / 750	7.4 / 6.7		60	40-60	> 60	LEU/Temporary storage in Silo at plant	RPV 30 m below grade in hardened silos
<b>POLAND</b>											
<b>HTGR- POLA</b>	30/ max. 10	Basic design	Prismatic	325 / 750				60		LEU/HALEU	Cogeneration operation, electrical power max. 10MW, high temp heat in steam max 25 t/h

# Challenges of HTGRs

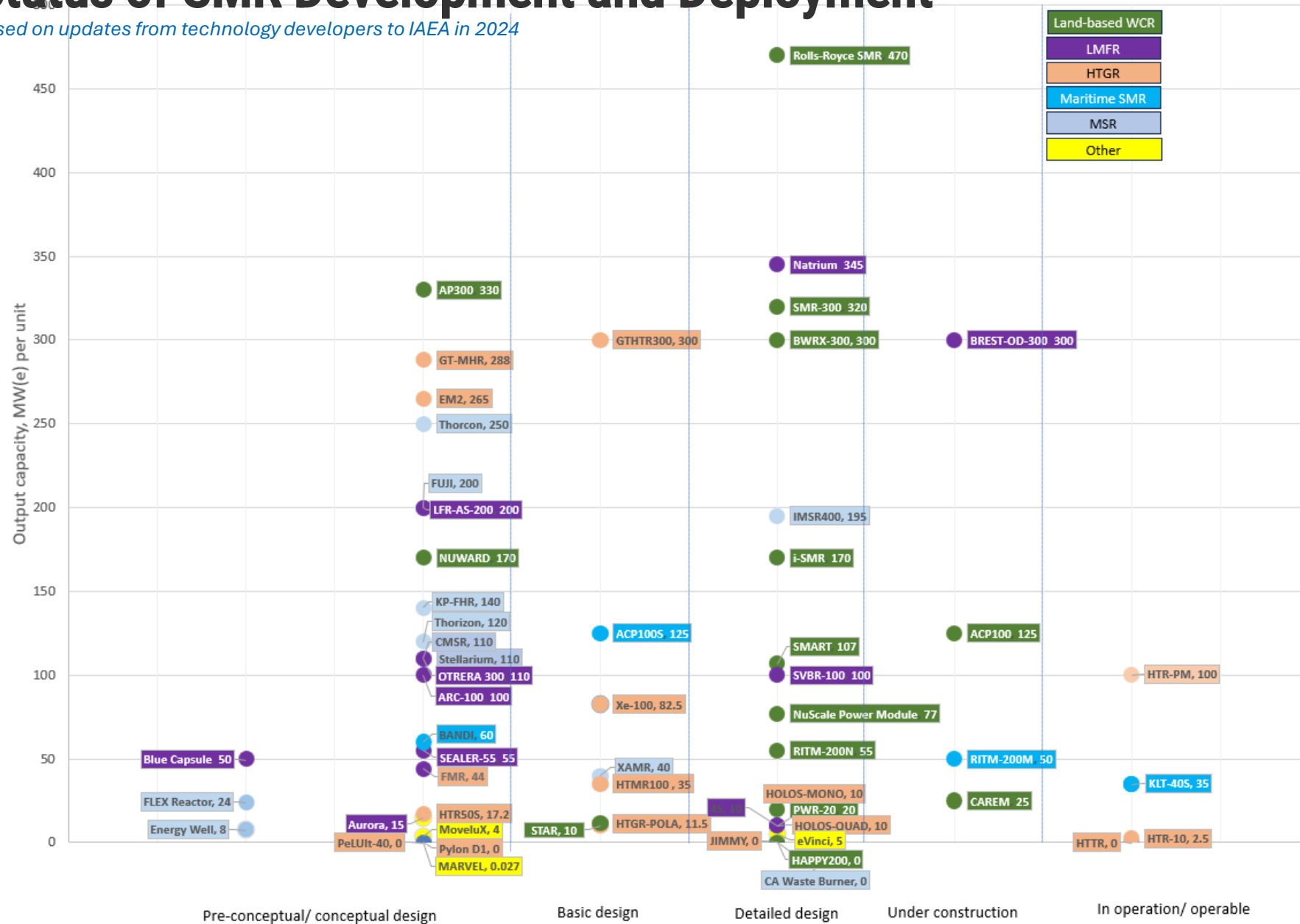
- Large core volume for the relatively low thermal power output (large pressure vessel)
- Susceptibility to water ingress after a steam generator or shutdown cooler tube rupture. Water in the core can pick up fission products, which would be released through pressure-relief valves.
- High-temperature coolant that exceeds the temperature limits of many metal alloys used in the power conversion system.
- Susceptibility to air ingress in the event of a leak in the primary coolant boundary. Oxygen can react with graphite, degrading the surface integrity and creating combustible gases.

# Challenges for HTGR deployment for non-electric applications

- Site and process-specific analysis and demonstration of the coupling of HTGRs to support non-electric applications
- Ensuring safety of the coupled system
  - Avoidance of cross-contamination
  - Management of thermal disturbances
  - Protection against external hazards
- Clarification and establishment of a regulatory scheme for the coupled system
- Cooperation between players from different sectors (nuclear and non-nuclear)
- Government commitment to decarbonisation policy and the provision of predictable and effective incentive schemes
- Cost and schedule predictability

# Status of SMR Development and Deployment

Based on updates from technology developers to IAEA in 2024



**Small Modular Reactors** are advanced reactors that produce typically up to 300 MWe, built in factories and transported as *modules* to sites for installation to shorten construction span thus reduce the cost.

Note: The value displayed with the design refers to the output capacity in MW(e) per unit.

# Gas Cooled Reactors – Technical Working Group (TWG-GCR)

- Advises the IAEA DDG-NE on specific topics of relevance to the IAEA programmatic activities in the field, since 1978
- Shares information and knowledge on national and international programme
- Contributes to the development and/ or review of selected IAEA publications, in particular from the IAEA Nuclear Energy Series, assesses existing gaps and advises on the preparation on new publications or e-learning materials
- Upon request, presents to the Standing Advisory Group on Nuclear Energy (SAGNE) the key findings of the TWG meeting
- Shares experience and advice on increasing the participating of young professionals and improving the gender balance in the nuclear sector
- Focus today on HTGRs
- **Member States** with designated member (2025-2028): Canada (TBC), China, Finland, France, Germany, Indonesia, Japan, **Jordan**, Republic of Korea, Poland, Russian Federation, South Africa, Switzerland, Ukraine, United Kingdom, United States of America
- **Observers:** European Commission, OECD/NEA



# Members, Observers & Technical Advisers TWG-GCR, new term 2025 – 2028



Members

No	Country	Name and Affiliation
1	Canada (TBC)	<b>Mr Ali SIDDIQUI</b> Canada Nuclear Laboratories (CNL)
2	China Past Chair of TWG-GCR (2021-2024) & Acting Chair for 2025	<b>Mr Yujie DONG</b> Tsinghua University, Institute of Nuclear and New Energy Technology (INET)
3	Finland	<b>Mr Jaakko LEPPÄNEN</b> VTT Technical Research Centre of Finland
4	France	<b>Mr Christoph DÖDERLEIN</b> Commissariat à l'énergie atomique et aux énergies alternatives (CEA)
5	Germany	<b>Mr Hans-Josef ALLELEIN</b> RWTH Aachen
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9	Republic of Korea	<b>Mr Chan Soo KIM</b> Korea Atomic Energy Research Institute (KAERI)
10	Poland	<b>Ms Agnieszka BOETTCHER</b> National Centre for Nuclear Research (NCBJ)
11	Russian Federation	<b>Mr Peter FOMICHENKO</b> National Research Centre Kurchatov Institute
12	South Africa	<b>Ms Vishana NAICKER</b> North-West University
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14	Ukraine	<b>Mr Mykola ODEYCHUK</b> Kharkov Institute for Physics and Technology
15	United Kingdom	<b>Ms Athanasia TZELEPI</b> National Nuclear Laboratory (NNL)
16	United States of America	<b>Mr Gerhard STRYDOM</b> Idaho National Laboratory (INL)

Observers



No	Country	Name and Affiliation
1	OECD/NEA	<b>Mr Brent WILHELM</b>
2	EC-JRC	<b>Mr Michael FUETTERER</b>

## Technical Advisers

No	Country	Name and Affiliation
1	China	<b>Mr Chen ZHIPENG</b> Tsinghua University, Institute of Nuclear and New Energy Technology (INET)
2	Japan	<b>Mr Junya SUMITA</b> Japan Atomic Energy Agency (JAEA)
3	Russian Federation	<b>Mr Grigory KODOCHIGOV</b> JSC Afrikantov OKBM

# Suggested strategic topics during the past TWG-GCR meetings

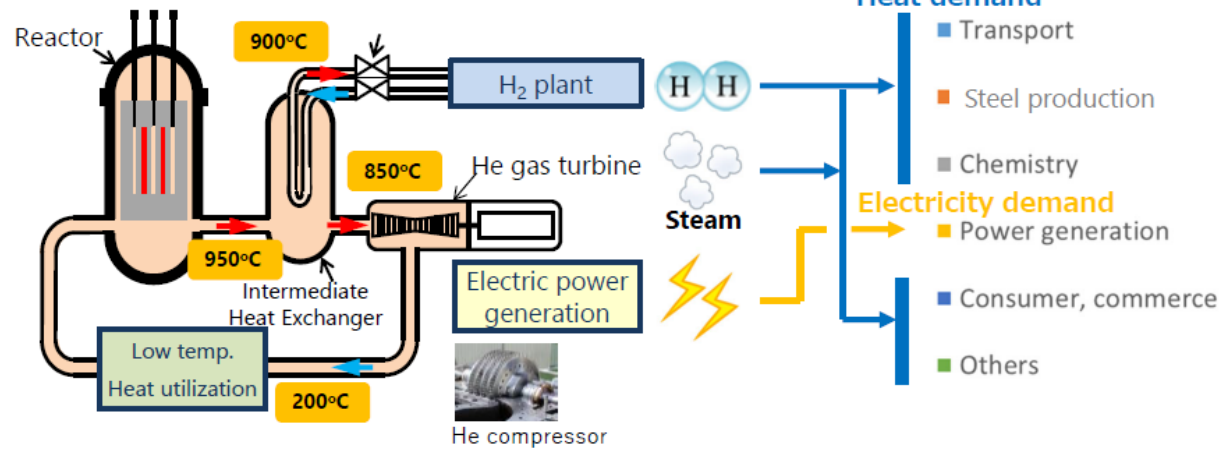
- **Develop safety standards applicable to modular HTGRs**
- **Facilitate and support R&D on HTGR for non-electric applications**
  - **Economic competitiveness** of HTGRs for cogeneration
  - Facilitation of information exchange on HTGR based cogeneration, especially with **technology providers** in order to better assess technology maturity and requirements needed for connection with HTGR
  - Develop **hydrogen roadmaps** to accelerate the hydrogen economy for carbon neutralization, using HTGR
  - Develop **safety standards for non-electric applications**, especially for the coupling infrastructure and coupling with other industries
  - Encourage harmonization of **regulatory approach for collocating HTR with various industrial processes**, for cogeneration of heat and electricity
- **Foster information sharing and collaboration (reactor technology, fuel cycle, waste management, ..)**
  - Treatment of **irradiated graphite** at industrial scale and management for its disposal
  - Establish the standard **fuel design and manufacturing** for HTGR
  - Develop approaches for **quality control at industrial level**
  - Initiate **international collaboration on separation of TRISO particle fuels from fuel matrix** and **R&D on irradiated graphite disposal**
- **Establishing a systematic approach for developing the safety analysis report for HTGR**
- **Support young generation through dedicated events and programmes**
- **Identify and improve material and component classification and reduce as much as possible the nuclear grade materials for economic reasons**
- **Address challenges of transportable HTGRs, transport after operation, safeguards and security**

# Suggested strategic topics during the past TWG-GCR meetings

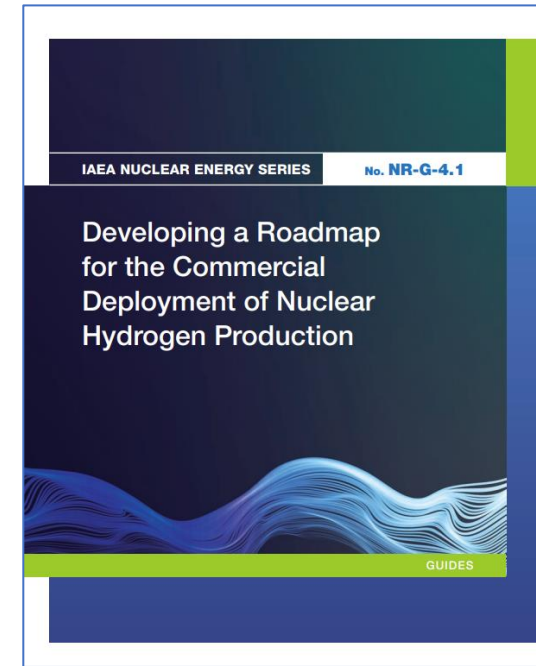
- **Disseminate modelling tools and facilitate E&T for HTGRs**
  - Continue and intensify the HTGR safety analysis code knowledge/capability training and sharing
- **Facilitate experiments, code development and data sharing for validation of thermo-hydraulic, neutronics, materials and safety codes, and uncertainties assessment**
  - Identify experimental facility requirements for establishing an HTGR design (incl. licensing) and possible experimental facilities available
  - Facilitate sharing of experimental facility required for establishing and checking an HTGR design, including the main components and systems, as well as the auxiliary systems (for e.g. the He purification system, RCCS)
  - Facilitate material testing and sharing of data
  - Development of codes for fuel design and benchmark models that apply for pebble bed and prismatic core
  - Establish a database not only for graphite but also for some other materials of interest for HTGRs where test results can be included
  - Develop user manuals to ensure effective use of results of various completed studies
  - Continuation of the ONCORE project and development of HCP code
  - Develop practice of multiphysics calculation for HTGR design purpose as regulators are used with this approach for LWR

# Hydrogen production using HTGR - Options

HTGR cogeneration system

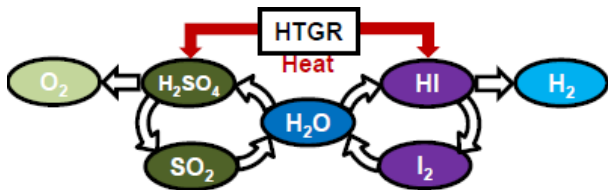


• Heat utilization rate is about 80%

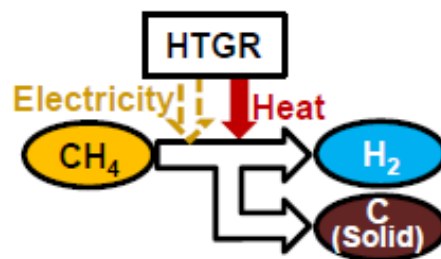


24-02841E\_PUB2096\_body\_v2.indd

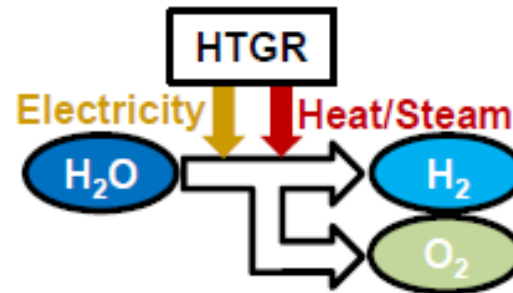
Thermochemical sulphur-iodine (S-I) process



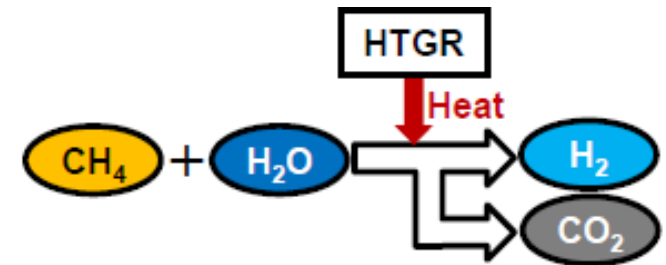
Methane pyrolysis



High temperature steam electrolysis

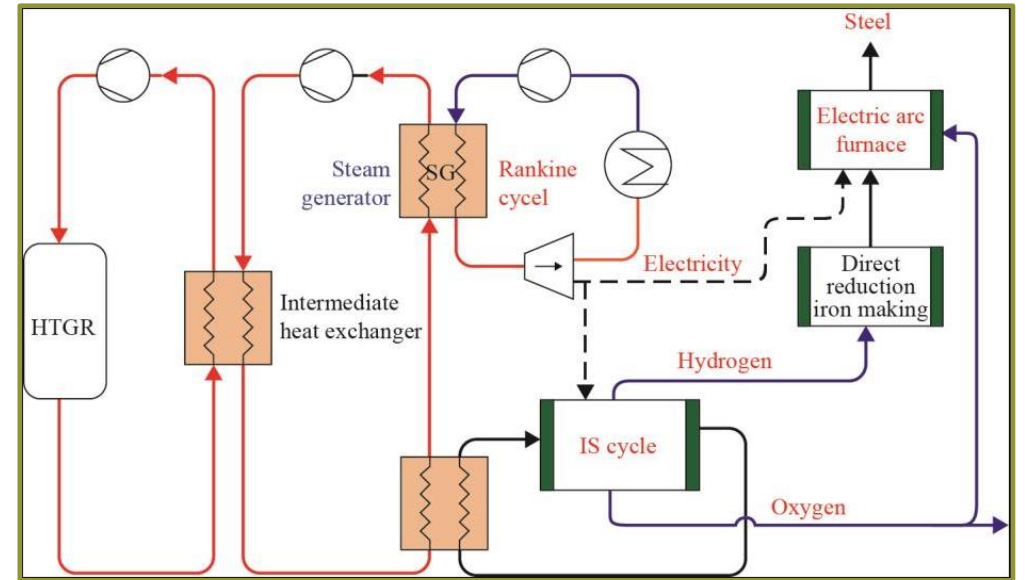


Steam methane reforming



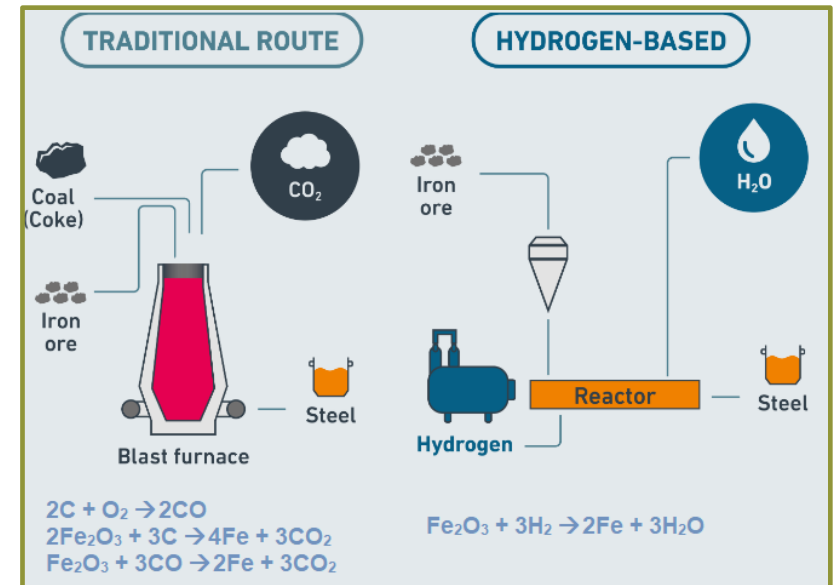
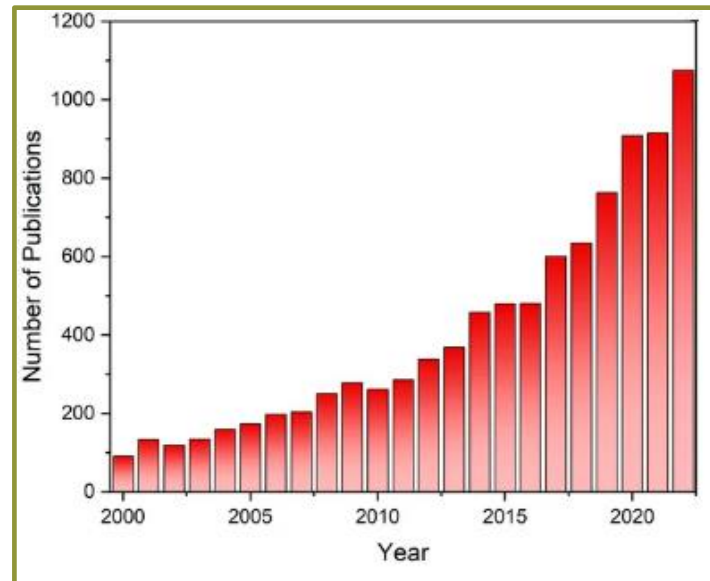
# Application prospects of nuclear hydrogen production with HTGRs

- Some of the applications include the direct reduction of iron, ammonia synthesis, coal liquefaction and refining.
- ~ 2 billion t steel produced annually, and demand is projected to rise by more than 1/3 by 2050



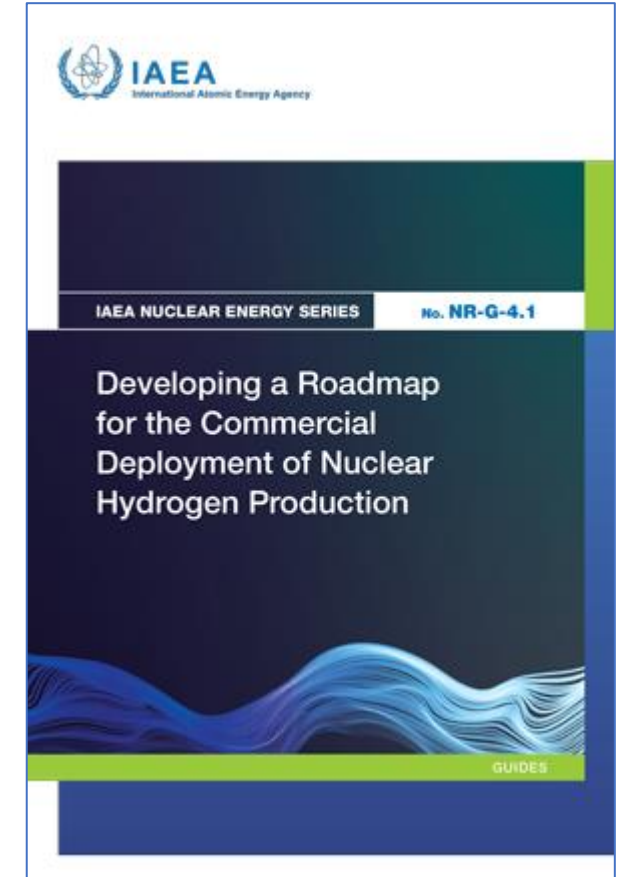
**Number of publications per year on the hydrogen-based reduction of iron oxides**

(Source: *Sustainability* 2023, 15(17), 13047)



# Developing a roadmap for commercial deployment of nuclear hydrogen

- Provide a useful management tool for evaluating, planning and strategizing the development of nuclear hydrogen projects by interested Member States
- Increased understanding of the technologies readiness level for different hydrogen production systems where nuclear energy is capitalized
- Identify the main challenges to be addressed in deployment of hydrogen production using nuclear energy and ways to address these challenges
- Milestones and actions needed by various stakeholders to accelerate deployment of hydrogen production using nuclear energy



[Developing a Roadmap for the Commercial Deployment of Nuclear Hydrogen Production | IAEA](#)

# Developing a roadmap for commercial deployment of nuclear hydrogen

6 main chapters, 196 p

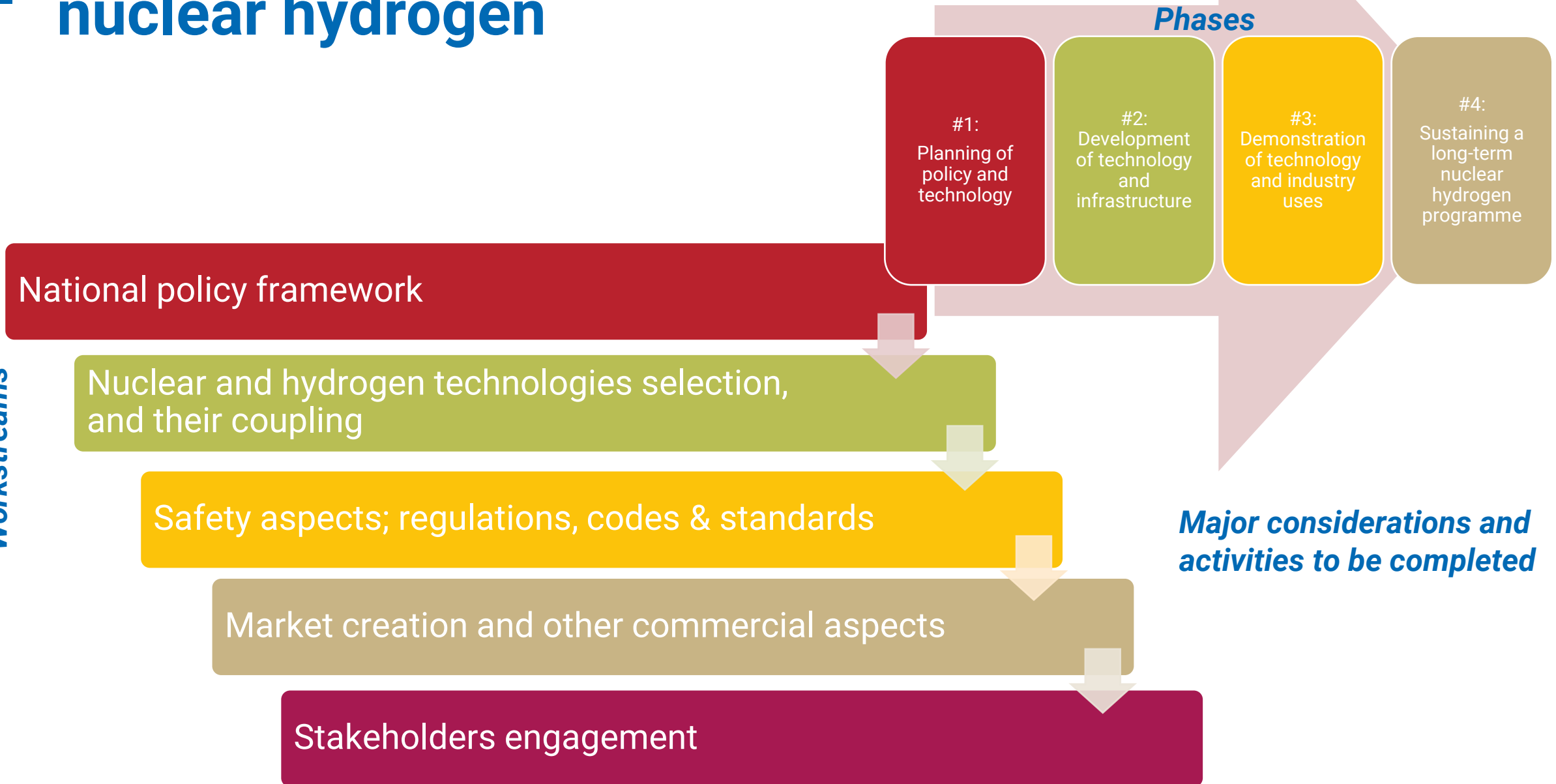


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# Developing a roadmap for commercial deployment of nuclear hydrogen

Workstreams



# HTGR Technology Development Knowledge Base

## HTGR Public - Home

### Welcome to the HTGR Technology Development Knowledge Base


The HTGR Technology Development Knowledge Base, is intended to increase collaboration and experience sharing in the field of innovation and technology for high temperature gas cooled reactors (HTGRs), as well as to retain the critical knowledge transferred from Forschungszentrum Jülich, Germany to the IAEA.

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- Platform to propose new topics, events, and collaborative works such as IAEA topical publications.


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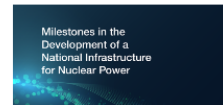
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### Latest News on SMRs and HTGRs



**IAEA Milestones Guidance Update to Include Considerations for SM**

Home > HTGR > HTGR Documentation (transferred from FZJ Germany to the

+ New Send by email Promote

### FZJ Documentation (FZJ, Thesis, Lectures)

Name	Modified
01-FZJ	July 17
09 - Habil PhD Dipl Master Bach L...	August 12
index.xlsx	August 12

### FZJ Documentation - International Projects

Name	Modified
CEC	August 12
EU Projects	August 13
HTMP - High Temperature Materi...	August 13
OECD DRAGON Project	August 13
index - International projects.xlsx	August 14

### FZJ Documentation - THTR

Name	Modified
01-THTR_CEC	September 6

### FZJ Documentation - AVR

Name	Modified	Modified By
03 - AVR	March 25	CONSTANTIN, Alina

## Members Areas (with access based on IAEA nucleus account and approval from the coordinators of the 'HTGR Knowledge Base')

# FZJ Archive – indexed in INIS

(~40 yr of experience documented in reports, articles, technical notes, etc. in ~20,000 files, 300 GB)

- Access INIS – International Nuclear Information System
- Search for “Juelich Preservation Project”

The screenshot shows the INIS search interface. At the top, the IAEA logo and the search query "Juelich preservation project" are visible. The search results are sorted by "Best match" and show 5,122 results. The results are displayed in a list format with filters for file type, resource type, and access status. The first three results are highlighted:


- 2017 (v1) Miscellaneous Open**  
829 **Irradiation Test Facility in NRG Petten for HTR-PM Spherical Fuel Elements**  
Zhao, Hongsheng  
Summary: -- Superb irradiation facilities HFR Petten -- Limited PIE facilities -- X-ray tomography of TRISO spheres have provided insights into the manufacture of high-quality spherical fuel elements  
Uploaded on January 19, 2025 | Published in: 48 p., 2017.  
Proceedings of: IAEA Fellowship Training Course on HTGR Fuel
- 2012 (v1) Miscellaneous Restricted**  
3,991 **Pilot-scale production of UO<sub>2</sub> kernels by sol-gel process at INET**  
Hao, Shaochang  
Outline: 1. Introduction 2. Processes, Facilities and Results 3. Concluding Remarks 4. Acknowledgement  
Uploaded on January 19, 2025 | Published in: 19 p., 2012.  
Proceedings of: 6. International Topical Meeting on High Temperature Reactor Technology
- 2012 (v1) Miscellaneous Restricted**  
829 **Experiments to Improve the Modeling of Air Ingress Accidents in Gas-Cooled VHTR**  
Allelein, H.-J.; Schlögl, B.; Jühe, S.; and 3 others  
Overview: - Introduction - Experiments -- INDEX (small scale) -- NACOK (Integral) - HTR Code Package (HCP) / STAR - Outlook  
Uploaded on January 19, 2025 | Published in: 20 p., 2012.  
Proceedings of: 6. International Topical Meeting on High Temperature Reactor Technology

The fourth result is partially visible:

- 2012 (v1) Miscellaneous Restricted**  
302 **Post-irradiation Examination and Fission Product Inventory Analysis of AGR-1 Irradiation Capsules**  
Harp, Jason M.; Demkowicz, Paul A.; Ploger, Scott A.  
Summary: • Fission product inventory has been extensively surveyed in the AGR-1 capsule components and compacts - Spatial distribution of Ag and Cs studied - Locating Cs activity in graphite fuel holder has lead to identification of particles with defective SIC for further study • Fractional releases from compacts: - Very low Cs release from int..  
Uploaded on January 19, 2025 | Published in: 26 p., 2012.







# HTR2024 Proceedings – indexed in INIS

## [The 11th International Topical Meeting on High Temperature Reactor Technology](#)

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Published 2024 | Version 1.0.0 Book Open

### The 11th International Topical Meeting on High Temperature Reactor Technology

Tsinghua University High Temperature Reactor Branch of China Nuclear Society, Institute of Nuclear and New energy Technology, Beijing, China ;  
Shanghai Electric Group Co., Ltd., Shanghai, China ;  
China North Nuclear Fuel Co., Ltd., Baotou City, China ;  
China Nuclear Power Operation Technology Corporation, Ltd., Wuhan, China ;  
China Techenergy Co., Ltd., Beijing, China ;  
Chengdu Taihua Zhongcheng Technology Group Co., Ltd., Baoji, China 


The HTR conference series focuses on High Temperature Gas-cooled Reactors (HTGR) technology including heat applications.

Inaugurated in 2002 by the European High Temperature Reactor Technology Network (HTR-TN) at Petten, the Netherlands, successive meetings were held in China (Beijing, 2004), South Africa (Johannesburg, 2006), the USA (Washington DC, 2008), Czech Republic (Prague, 2010), Japan (Tokyo, 2012), China (Weihai, 2014), the USA (Las Vegas, 2016), Poland (Warsaw, 2018) and Indonesia (Yogyakarta, 2021).

HTR 2024 is to be held in Beijing, China during 14-18 October, 2024. HTR 2024 aims to accelerate research and development on HTGR and heat application technologies, and make practical use of HTGR systems, through discussing and exchanging on the latest results and information on the above technologies and user requests as well as future perspectives and plans.

#### Files

HTR 2024 - International Conference on High Temperature Reactor Technology ( August 2024) - China.pdf >

Page: 4 of 4 Automatic Zoom: 

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Book

**Conference**  
[The 11th International Topical Meeting on High Temperature Reactor Technology \(HTR 2024\)](#) , Beijing, China, 14-18 October, 2024

**Languages**  
English

#### Associated items

- A concept of fast steady-state therma...
- GEMINI 4.0, a project for developing ...
- Criticality Safety for HTGR Fuel Plan...
- Advancing Nuclear Energy in Indonesia...
- Analysis and Evaluation of Mining Are...
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# HTGR Technology Development Knowledge Base



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### [Repository of HTGR facilities](#)

(This repository was last updated in 2020 and includes not only active facilities but also closed, decommissioned, in care and maintenance facilities).

The up-to-date information on active HTGR facilities is included in the [Catalogue of experimental facilities under NEXSHARE](#) and will be regularly updated.

**Experimental facilities database – 174 entries**  
includes also closed, decommissioned, in care and maintenance facilities

# International Network for Experiment and Code Validation Sharing (NEXSHARE)



NEXSHARE

https://nucleus.iaea.org/sites/connect/NEXPublic/SitePages/Home.aspx

- Home
- Experimental Facilities
- Codes
- Steering Committee & FG
- Resources & Tools
- Collaboration Projects
- Participating Organizations
- Events
- Members Only

- Past Events Materials
- Members Contact
- Published Deliverables
- Initiate Collaboration
- Add/update facility

The International **Network for Experiment and Code Validation Sharing (NEXSHARE)** is forum for global cooperation and resource sharing on experiments and code validation for Small Modular Reactors (SMRs). Participating organizations includes experimental facilities, technology holders, International Organizations and Technical Support Organizations (TSOs).

IAEA

NEXSHARE Network

Experimental Testing and Validation for Design and Safety Analysis

EXPLORE THE EXPERIMENTAL FACILITIES >



Resource & Tools



Collaboration Projects

# HTGR experimental facilities in NEXSHARE



Participating Experimental Facilities

Country Name  Reactor family  Fluid  Application

**23 HTGR facilities included (as of April 2026)**

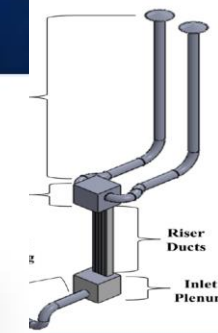
Building connections with research institutes and industry partners

Number of items: 23

See all

Facility	Full Name	Country	Organization	Reactor	Fluid	Application	Max Power(kW)	Max Pressure(MPa)	Max Temperature(C)	Flow Rate(kg/s)
Bell Jar	CNL Vacumm Bell Jar Apparatus	Canada	CNL	WCR, HTGR	N/A	Components/Feature, Materials, Severe Accident	50	0.1	1,400	
HANC	Helium-Air Natural Convection	Canada	CNL	HTGR	helium and air	Containment, Thermohydraulics - separate effects	130	0.2	700	
BTF	Burst Test Facility	Canada	Kinectrics Inc.	WCR, MSR, LMF...	Water/inert gas/Hydrogen/Heliu m (environment can change as required)	Components/Feature, Materials, Severe Accident	138		950	10
STSGSTHTF	Spiral Tube Steam Generator Supercritical Thermal Hydraulic Test Facility	China	CNNC	HTGR	water	Thermohydraulics - separate effects	500	32	600	0.7
YBS	YBS Critical Experimental Assembly	China	CNNC	WCR, HTGR	N/A	Fuel, Neutronics	0.01			
TF-PBF3D	Pebble Bed Flow 3D	China	Tsinohua	HTGR	Air	Thermohydraulics - separate	10	0.1	20	

University of Wisconsin Air-Cooled RCCS Test Facility



GENERAL INFORMATION	
Acronym:	UW-ARCCS
Facility name:	Air-Cooled RCCS Test Facility
Organization:	University of Wisconsin
Country:	United States
Operational since:	
Reactor family:	<input type="checkbox"/> Water Cooled <input type="checkbox"/> Molten Salt <input checked="" type="checkbox"/> High Temperature Gas Cooled <input type="checkbox"/> Liquid Metal Cooled Fast Spectrum Other:

### TECHNICAL DESCRIPTION

Conditions modelled:  Normal operation  Accident conditions  
 please specify:

Phenomena - separate effects  Thermohydraulics - integral effects  Neutronics  
 Components/Feature  Materials  
 Severe Accident  Hazards (seismic)

Facility modelled: Experimental study investigation for a scaled Air-cooled RCCS was designed and built for the Modular High Temperature Reactor cooling system that demonstrates a reactor pressure vessel design. The reactor pressure vessel was used to pressurize a quarter-scale reduced height or slice of the full-scale reactor. The maximum power and surface heat flux of the radiant heaters is 40 kW with a peak heat flux of 25 kW per meter squared. The facility was run under different heat loading cases and operated successfully. Instabilities were observed in some experiments in which one of the two exhaust ducts experienced a flow reversal for some period of time. The data and analysis presented show that the RCCS has the potential to be a decay heat removal system during an accident scenario.

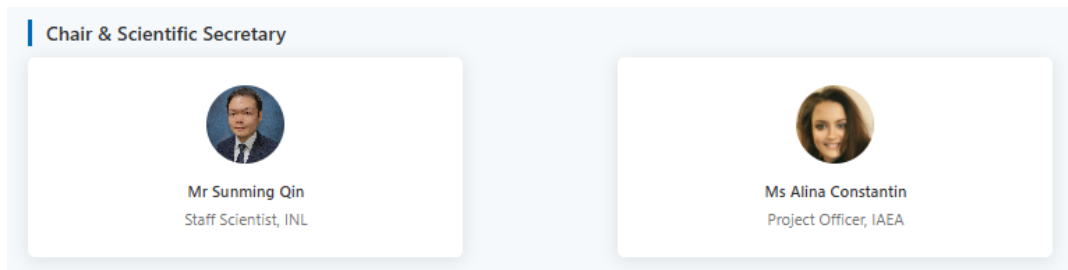
Parameters:

	Min	Max
Power [kW]:	0 [kW]	40 [kW]
Heat flux [kW/m <sup>2</sup> ]:	N/A	N/A
Pressure [MPa]:	0.1 [MPa]	0.1 [MPa]
Temperature [°C]:	20 [°C]	900 [°C]
Flow Rate [kg/s]:	Min Flow Rate [kg/s]	Max Flow Rate [kg/s]

## Collecting experimental facility information for NEXSHARE

- NEUP funded HTGR-TH experimental projects as pilot
- Support from TWG-GCR members and HTGR Focus Group members

# Activities under the NEXSHARE HTGR Focus Group



- **Review of Ongoing International Initiatives:**

A thorough review of existing HTGR initiatives within International Organizations to enhance information sharing and avoid duplication. A working paper including these efforts is available at [nucleus.iaea.org/sites/nexshare/Focus Group/HTGR FG/HTGR\\_FG\\_working\\_paper.pdf](https://nucleus.iaea.org/sites/nexshare/Focus%20Group/HTGR%20FG/HTGR_FG_working_paper.pdf) and a more detailed background document at [HTGR FG - Background Document on International Organization Activities.pdf](https://nucleus.iaea.org/sites/nexshare/Focus%20Group/HTGR%20FG/HTGR_FG_working_paper.pdf)

- **Expand Experimental Facility Database for HTGRs:**

Continuous effort to expand and enrich the database of experimental facilities relevant to HTGR R&D.

- **Review the Existing HTGR Simulation Codes:**

An open list was created with input from the members of the HTGR FG.

- **Organize topical webinars:**

Upcoming webinar in Q2 on Code development and validation efforts for HTR technology (28 May 2026)

30 October 2025, 14:00 – 15:30 (CET)

## *NHSI Industry Track*

### Webinar on:

# **NEXSHARE Network Activities on Experiments and Code Validation for Small Modular Reactors (SMRs) and Related International Organizations Efforts**

### *Agenda*

### *Duration*

**Introduction & NEXSHARE Overview**

0:05

**Presentations from the NEXSHARE Focus Group of WCR, **HTGR**, LMFR and MSR**

0:50

**Panel Session**

0:25

**Q&A**

0:05

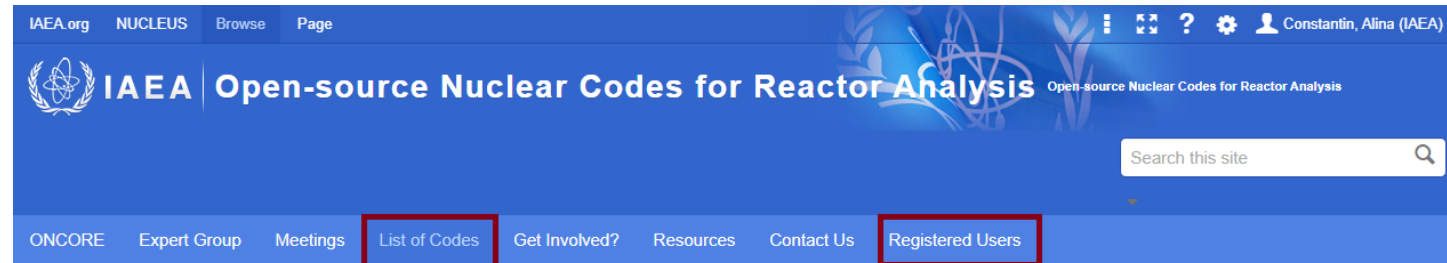
**Conclusions**

0:05

*End of Meeting*

# HTR Codes (HCP, STACY, VSOP)

- HTR codes are accessible through the Open-Source Nuclear Codes for Reactor Analysis (ONCORE)
  - The ONCORE initiative is an IAEA-facilitate international collaboration framework for the development and application of open-source multi-physics simulation tools to support research, education and training for the analysis of advanced nuclear power reactors. Institutions and individuals participating in ONCORE can collaborate in, and benefit from, the development of open-source software in the field of nuclear science and technology.
  - <https://www.iaea.org/topics/nuclear-power-reactors/open-source-nuclear-code-for-reactor-analysis-oncore>
  - <https://nucleus.iaea.org/sites/oncore>



[Home](#)

## List of Codes

**Want to get involved?** You are invited to submit code to be included in this list, please follow the instruction available at [Get Involved?](#)

This preliminary list has been compiled by the ONCORE expert group. It does not include software that is not made available by its developers, nor software whose development has been discontinued. The authors acknowledge the fact that they might have overlooked on several available open-source codes. Would you be aware of other relevant software that could complement this list, please contact us at [\(ONCORE@iaea.org\)](mailto:ONCORE@iaea.org).

### Codes available through ONCORE

Name	Brief description	How to obtain the software	License
VSOP99/11	HTR pebble-type design and safety analysis	Available at: <a href="#">Registered users of ONCORE</a>	MIT
STACY	V/HTR safety analyses for the quantification of fission product release from the fuel.	Available at: <a href="#">Registered users of ONCORE</a>	MIT
HCP	HTGR safety analysis integrated code	Available at: <a href="#">Registered users of ONCORE</a>	MIT

## Source code available in Github

If you need access to source code, please write an email to: [oncore.contact-point@iaea.org](mailto:oncore.contact-point@iaea.org) with detailed explanation of the expected use.

# Workshop on High Temperature Gas Cooled Reactor Technology and Training on the High Temperature Reactor Code Package (3-7 November 2025, hosted by TUM)

- Organized locally by the Technical University of Munich, with IAEA support from TC Programme INT2023
- Agenda included lectures and practical sessions, illustrating the capabilities of the version available in the IAEA ONCORE platform and also the capabilities of the HCP version under development at the Technical University of Munich and of the STACY version developed in Becker Technologies
- Training provided by:
  - ❑ Mr Andre Xhonneux (former developer of HCP code in FZJ, Germany) - HCP
  - ❑ Mr Chunyu Liu (Technical University of Munich, Germany) - HCP
  - ❑ Ms Meryll Colombet (Becker Technologies, Germany) - STACY
  - ❑ Mr Wayne Boyes (STLN, South Africa) - VSOP
- Lectures provided by:
  - ❑ Prof. Rafael Macian (Technical University of Munich, Germany)
  - ❑ Prof. Hans-Josef Allelein (Germany)
  - ❑ Mr Jiong Guo (Tsinghua University, China)
  - ❑ Ms Alina Constantin (IAEA)

On the following topics:

- (1) HTGR plant engineering and safety, (2) fuel performance, (3) reactor physics, (4) thermal fluid dynamics,
- (5) Chinese experience on HTGR, (6) overview of global HTGR technology developments, (7) HTGR challenges

- **17 participants** joined the event from **11 Member States**: China, France, Indonesia, Malaysia, Mongolia, Nigeria, Pakistan, Poland, Singapore, South Africa, Ukraine.
- The aim of the training was to give participants a first glance on the capabilities of the HCP, STACY and VSOP codes and to offer them possibility to practice on exercises to get familiar with creating an input file, running the code and then visualization and interpretation of the results obtained.



4 Fellowships were supported during 2025 by IAEA for Indonesian researchers for capacity building in working with HTR codes (in particular STACY & HCP) at TUM

# Nuclear Graphite Knowledge Base (NGKB)

<https://www.iaea.org/resources/databases/iaea-nuclear-graphite-knowledge-base>



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## IAEA Nuclear Graphite Knowledge Base

This knowledge base supports the preservation and sharing of expert knowledge and experience across the international Graphite Community.

The knowledge base contains general information on the subject of nuclear graphite and specialist knowledge that is restricted to members of the international project.

### Resource Information

Quality Level	Not quality checked
Completeness	Selective - Effort is made to create a comprehensive database on Irradiated Nuclear Graphite but it might not cover all elements
Update Frequency	Updated periodically



Open Knowledge Wiki | Restricted Resources | NGKB Meetings

### Home

## Welcome to the IAEA Nuclear Graphite Knowledge Base

#### Mission Statement

The aim of this knowledge base is to support the preservation and sharing of expert knowledge and experience, across the international Graphite Community.

The knowledge base contains two levels of knowledge:

- General information on the subject of Nuclear Graphite
- Specialist knowledge, secured for members of the international project

Please click on the below images to link to the desired section of the knowledge base

#### Databases

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25th International Nuclear Graphite Specialists' Meeting  
29 Sept – 3 Oct 2025 (Vienna)

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Lists

## INGSM

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Name	Modified
INGSM-1 Material	April 23, 2013
INGSM-10 Material	April 23, 2013
INGSM-11 Material	April 23, 2013
INGSM-12 Material	April 23, 2013
INGSM-13 Material	April 23, 2013
INGSM-14 Material	December 24, 2013
INGSM-15 Material	February 3, 2015
INGSM-16 Material	November 17, 2015
INGSM-17 Material	September 16, 2016
INGSM-18 Material	November 20, 2017
INGSM-19 Material	September 11, 2018
INGSM-2 Material	April 23, 2013
INGSM-20 Material	November 13, 2019
INGSM-21 Material	November 27, 2023
INGSM-22 Material	November 21, 2023
INGSM-23 Material	November 26, 2023
INGSM-24 Material	June 16
INGSM-25 Material	October 7
INGSM-3 Material	December 24, 2013
INGSM-4 Material	April 23, 2013
INGSM-5 Material	April 23, 2013
INGSM-6 Material	April 23, 2013
INGSM-7 Material	April 23, 2013
INGSM-8 Material	April 23, 2013
INGSM-9 Material	April 23, 2013
INGSM_Index	October 8

### TOPICS

- Irradiation damage, in-pile experiments
- Oxidation
- Graphite waste, and graphite assembly and component disassembly and decommissioning
- Use of graphite in molten salt reactors
- Thermo-physical, thermo-chemical, and mechanical properties of graphite
- Microstructure and characterization
- Standards, licensing and graphite qualification
- Physisorption and chemisorption of tritium and other gases in graphite
- Innovations and experience in manufacturing and purification, component and assembly fabrication and installation, and graphite supply chain
- Functional/performance requirements for graphite components



Next year:

26th International Nuclear Graphite Specialists Meeting



# IAEA Publications on HTGR :

## Reactor design and development, thermo-hydraulics, reactor physics, reactor performance, related experiments

**Decay heat removal and heat transfer under normal and accident conditions in gas cooled reactors, [IAEA-TECDOC-757](#) (1994)**

**Design and Development of Gas Cooled Reactors with Closed Cycle Gas Turbines, [IAEA-TECDOC-899](#) (1996)**

- Includes the status and technology development activities for HTGRs with closed cycle gas turbines (HTGR-GT), and identifies pathways of opportunity for international cooperation in the development of common systems and components

**High Temperature Gas Cooled Reactor Technology Development, [IAEA-TECDOC-988](#) (1998)**

- Includes the following topics: HTGR national development and commercialization programmes (as of 1996); safety and management, development of the pebble bed modular reactor in South Africa; status of HTGR test reactor programmes; HTGR plant component and system design development.

**Critical Experiments and Reactor Physics Calculations for Low Enriched High Temperature Gas Cooled Reactors, [IAEA-TECDOC-1249](#) (2001)**

- Covers the work conducted under the IAEA CRP on the Validation of Safety Related Physics Calculations for Low-Enriched HTGRs, providing information on the PROTEUS facility and the results of the critical experiments and validation activities conducted.

**Current Status and Future Development of Modular High Temperature Gas Cooled Reactor Technology, [IAEA-TECDOC-1198](#) (2001)**

- Presents the status and plans for utilization of the modular HTGR as an energy source for industrial applications and the generation of electricity.

**Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to Initial Testing of the HTRR and HTR-10, [IAEA-TECDOC-1382](#) (2003)**

- Reports the computational results made by various international institutes and code-to-code and code-to-experiment comparisons made. The experimental data are mainly related to the initial testing of the Japanese HTRR and the Chinese HTR-10.

**Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to the PBMR-400, PBMM, GT-MHR, HTR-10 and the ASTRA Critical Facility, [IAEA-TECDOC-1694](#) (2013)**

- Presents the findings of an IAEA coordinated research project focusing on validation of the safety and operational aspects of high temperature gas cooled reactors under projected and actual operating conditions.

# IAEA Publications on HTGR : Fuel and materials fabrication and performance

**Response of Fuel, Fuel Elements and Gas-Cooled Reactor Cores under Accidental Air or Water Ingress Conditions, [IAEA-TECDOC-784](#) (1995)**

**Fuel Performance and Fission Product Behaviour in Gas-Cooled Reactors, [IAEA-TECDOC-978](#) (1997)**

- Documents the activities of the IAEA CRP on Validation of Predictive Methods for Fuel and Fission Product Behaviour, with respect to the technical areas of fuel performance during normal and accident oxidizing and non-oxidizing conditions, transport of gaseous and metallic fission products during normal and accident conditions, performance of advanced fuels and GCR fuel design and fabrication programmes

**Status and Prospects for Gas Cooled Reactor Fuels, [IAEA TECDOC \(CD-ROM\) No. 1614](#) (2009)**

- Includes detailed review papers on conventional and advanced fuel designs, fabrication technology, quality assurance and quality control, fuel irradiation qualification, fuel performance, fuel modelling and overall fuel cycle issues.

**High Temperature Gas Cooled Reactor Fuels and Materials, [IAEA TECDOC \(CD-ROM\) No. 1645](#) (2010)**

- Documents the knowledge and experience in the development of HTGRs gained over 50 years and serves as a basis for development of fuels

**Advances in High Temperature Gas Cooled Reactor Fuel Technology, [IAEA TECDOC \(CD-ROM\) No. 1674](#) (2013)**

- Reports on the results of a IAEA CRP on advances in HTGR fuel technology and describes the R&D findings on coated particle developments. Includes two specific benchmark exercises with the application of HTGR fuel performance and fission product release codes, which helped compare the quality and validity of the computer models against experimental data.

**Performance Analysis Review of Thorium TRISO Coated Particles During Manufacture, Irradiation and Accident Condition Heating Tests, [IAEA-TECDOC-1761](#) (2015)**

- Gathers the outcome of an IAEA CRP on near term and promising long-term options for deployment of thorium based nuclear energy. It is based on the compilation and analysis of available results on TRISO coated particle fuel performance in manufacturing during irradiation and accident condition heating tests.

**Coated Particle Fuels for High Temperature Gas-Cooled, Small Modular Reactors, [IAEA-TECDOC-2090](#)**

- Provides an up-to-date compilation of information on technological progress in design, manufacturing, experimentation, modelling, and analysis technologies for HTGR coated particle fuels, and to identify the major challenges and prospects for use of these coated particle fuels in high temperature gas-cooled SMRs.

# IAEA Publications on HTGR :

## Reactor safety, safety requirements

**Gas-cooled Reactor Safety and Accident Analysis (Proceedings of a Specialists' Meeting, Oak Ridge, 13-15 May 1985), [IAEA-TECDOC-358](#) (1985)**

**Heat Transport and Afterheat Removal for Gas Cooled Reactors under Accident Conditions, [IAEA-TECDOC-1163](#) (2001)**

- Documents the activities of the IAEA CRP on Heat Transport and Afterheat Removal for Gas-Cooled Reactors Under Accident Conditions, with respect to the technical areas of code-to-code and code-to-experiment validation of code predictions for normal operation and loss of cooling accidents with and without simultaneous depressurisation.

**Considerations in the Development of Safety Requirements for Innovative Reactors: Application to Modular High Temperature Gas Cooled Reactors, [IAEA-TECDOC-1366](#) (2003)**

- Method for safety assessment based on principle of defence in depth

**Accident Analysis for Nuclear Power Plants with Modular High Temperature Gas Cooled Reactors, [Safety Reports Series No. 54](#) (2008)**

- Specific guidance on conducting accident analyses of nuclear power plants with modular HTGRs, considering specific design, operational and safety features.

**Applicability of Design Safety Requirements to Small Modular Reactor Technologies Intended for Near Term Deployment, [IAEA-TECDOC-1936](#) (2020)**

- Focuses on applying the design requirements in SSR-2/1 (Rev. 1) to LWR and HTGR type SMRs.

**Approach and Methodology for the Development of Regulatory Safety Requirements for the Design of Advanced Nuclear Power Reactors, [IAEA-TECDOC-2010](#) (2022)**

- Identifies key design features of SMRs to be considered important for the process of development or updating the regulatory safety requirements.

# IAEA Publications on HTGR : Applications of HTGRs

**Gas-cooled Reactors and Their Applications (Proceedings of a Technical Committee Meeting, Juelich, 20-23 October 1986), [IAEA-TECDOC-436](#) (1987)**

**Advances in Nuclear Power Process Heat Applications, [IAEA-TECDOC-1682](#) (2012)**

- Compiles the findings of R&D activities related to practical nuclear process heat applications
- Includes an overview of HTGR coupling schemes for different process heat applications, such as hydrogen production and desalination.

**Energy Neutral Mineral Processing with High Temperature Reactors: Resource Identification, Uranium Recovery and Thermal Processes, [IAEA-TECDOC-2023](#) (2023)**

- Reports on the findings of an IAEA Coordinated Research Project which investigated the use of SMRs for energy neutral mineral processing.

# IAEA Publications on HTGR : Graphite management

**The Status of Graphite Development for Gas-cooled Reactors (Proceedings of a Specialists Meeting, Tokai-mura, Japan, 9-12 September 1991) (1993)**

**Improving the Understanding of Irradiation Creep Behaviour in Nuclear Graphite – *upcoming***

# IAEA Publications on HTGR : Reactor decommissioning, spent fuel storage and management

**Technologies for Gas Cooled Reactor Decommissioning, Fuel Storage and Waste Disposal, [IAEA-TECDOC-1043](#) (1998)**

- Reviews the status of decommissioning and associated spent fuel storage and component waste disposal programmes and issues related to GCR plants including facilities sharing common technological aspects such as other types of reactors which have graphite moderators; and especially to identify pathways which may take advantage of the opportunity for international cooperation on developments addressing these activities.

**Waste from Innovative Types of Reactors and Fuel Cycles, [IAEA Nuclear Energy Series No. NW-T-1.7](#) (2019)**

- Sets the stage for considering the waste generation of advanced fuel fabrication, reactor operation and decommissioning, reprocessing of spent fuel and waste pathways early in the development of new reactors and their associated fuel cycles. It describes waste flows in broad chemical and physical terms and identifies possible processing, recycling and disposition pathways.

**Considerations for the Back End of the Fuel Cycle of Small Modular Reactors, [IAEA-TECDOC-2040](#) (2023)**

- Identifies the opportunities and challenges faced at all stages needed for managing the spent fuels from different SMR technologies.



**IAEA**

# Thank you for your attention!

**For inquiries, please contact:**

**[a.constantin@iaea.org](mailto:a.constantin@iaea.org)**

**IAEA Division of Nuclear Power, Nuclear Power Technology Development Section**