IAEA activities on HTGR technology development

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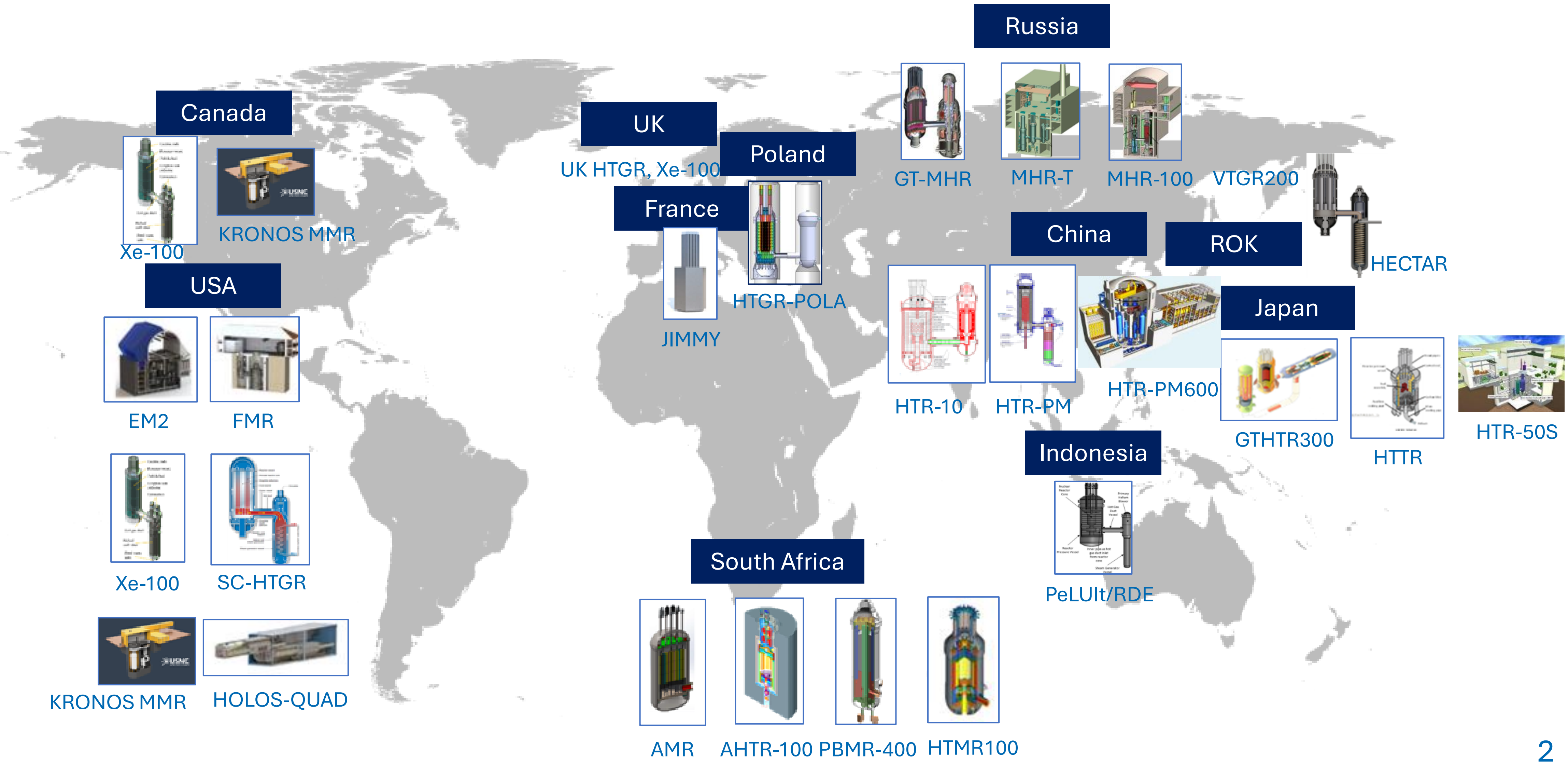
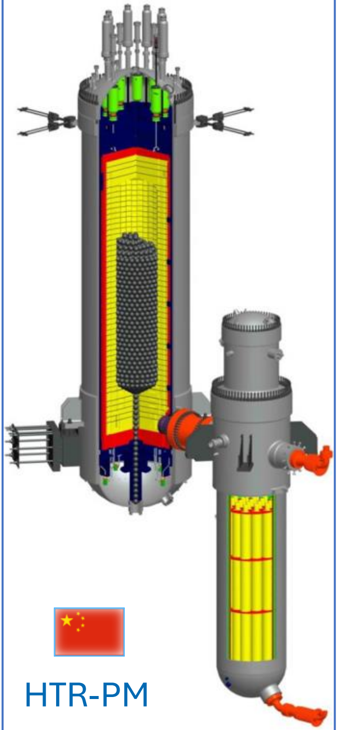
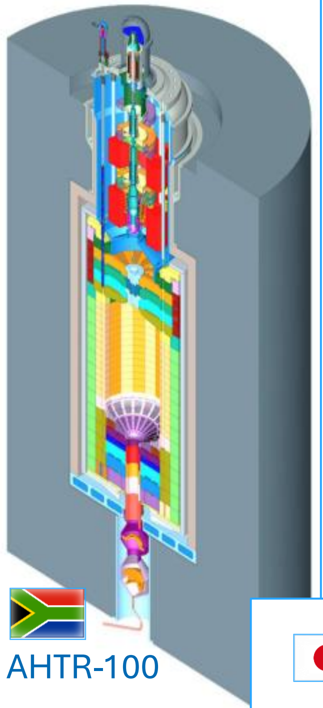
Global Activities on HTGR-SMR development



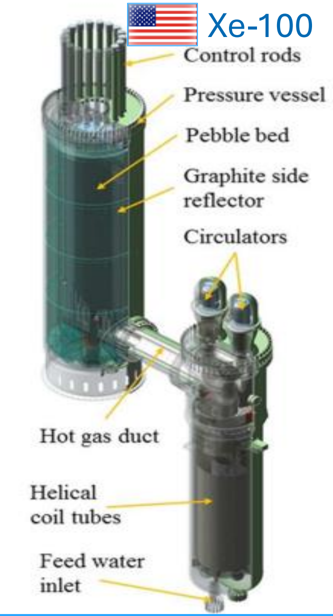
Multiple designs from Member States



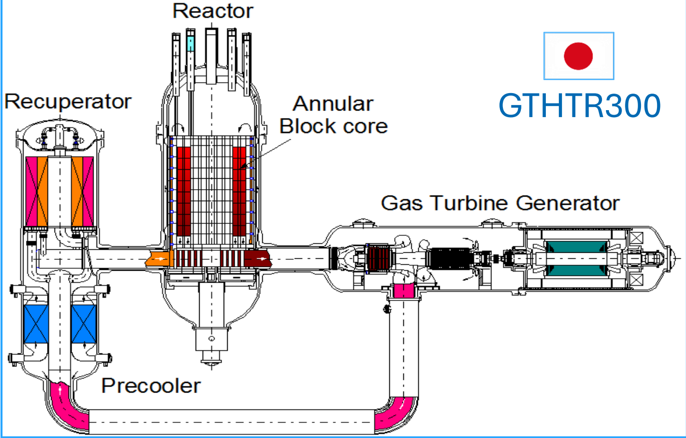
HTR-PM



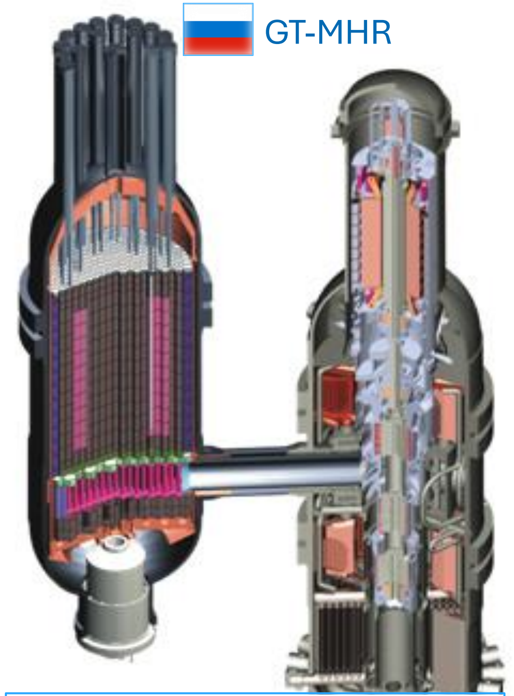
AHTR-100



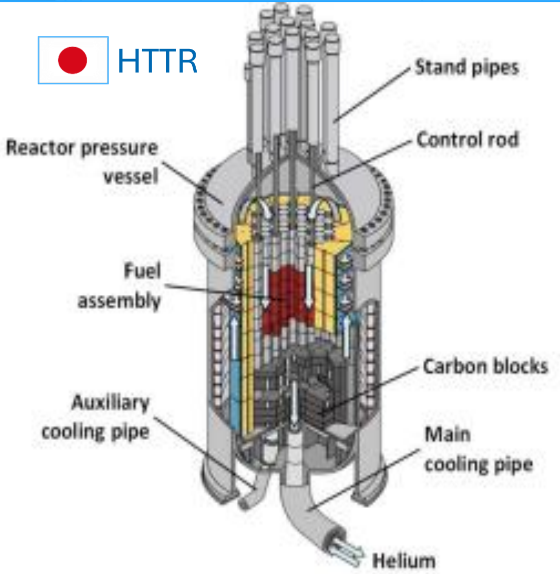
Xe-100



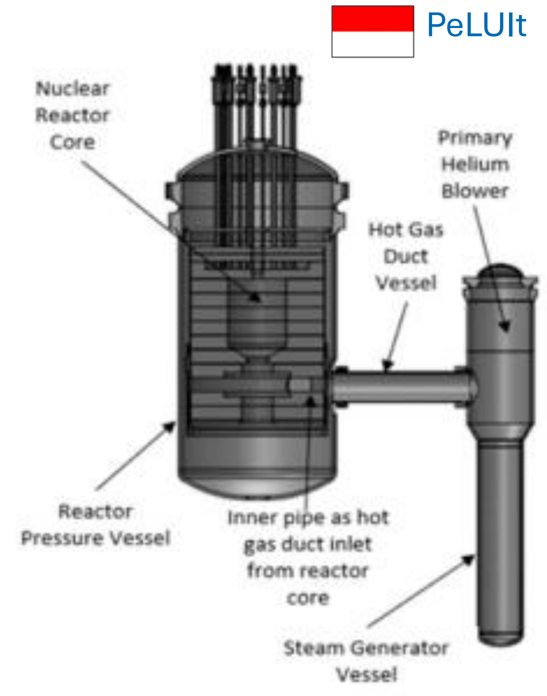
GTHTR300



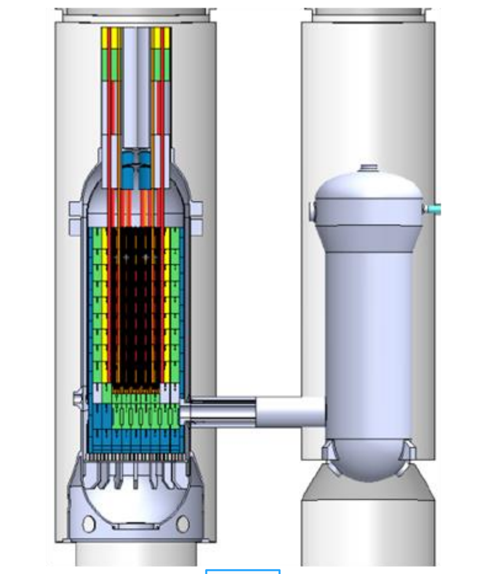
GT-MHR



HTTR



PeLUIt



HTGR-POLA

Number of SMR designs under development (2022)

- 38 Water-cooled
- 45 Non-water cooled
  - HTGR 21
  - MSR 13
  - LFR 8
  - SFR 3

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
- **15 Member States** with designated member (**2021-2024**)
- **2 Observers**: European Commission, OECD/NEA

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

No	Country	Name and Affiliation
1	Canada	**Mr Ali SIDDIQUI** Canada Nuclear Laboratories (CNL)
2	China *The Chair of TWG-GCR (2021-2024)*	**Mr Yujie DONG** Tsinghua University, Institute of Nuclear and New Energy Technology (INET)
3	Finland	**Mr Ville TULKKI** VTT Technical Research Centre of Finland
4	France	**Mr Christoph DÖDERLEIN** Commissariat à l'énergie atomique et aux énergies alternatives (CEA)
5	Germany	**Mr Hans-Josef ALLELEIN** RWTH Aachen
6	Indonesia	**Mr Topan SETIADIPURA** National Research and Innovation Agency (BRIN)
7	Japan	**Mr Tetsuo NISHIHARA** Japan Atomic Energy Agency (JAEA)
8	Republic of Korea	**Mr Chan Soo KIM** Korea Atomic Energy Research Institute (KAERI)

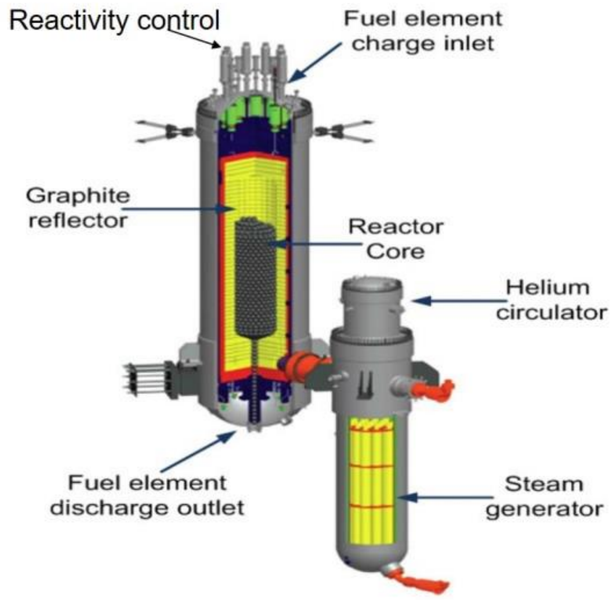
No	Country	Affiliation
9	Poland	**Ms Agnieszka BOETTCHER** National Centre for Nuclear Research (NCBJ)
10	Russian Federation	**Mr Peter FOMICHENKO** National Research Centre Kurchatov Institute
11	South Africa	**Ms Vishana NAICKER** North-West University
12	Switzerland	**Mr Manuel POUCHON** Paul Scherrer Institute (PSI)
13	Ukraine	**Mr Mykola ODEYCHUK** Kharkov Institute for Physics and Technology
14	United Kingdom	**Mr Timothy ABRAM** University of Manchester
15	United States of America	**Mr Gerhard STRYDOM** Idaho National Laboratory (INL)

Suggested strategic topics

- **Develop safety standards applicable to modular HTGRs**
- **Facilitate and support R&D on HTGR for non-electric applications**
  - Economic competitiveness of HTGRs for cogeneration
  - Continue and intensify effort to implement a HTGR based cogeneration, through facilitation of information exchange, 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
- **Disseminate modelling tools and facilitate E&T for HTGRs**
  - Continue and intensify the HTGR safety analysis code knowledge/capability training and sharing
- **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
- **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
- **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**

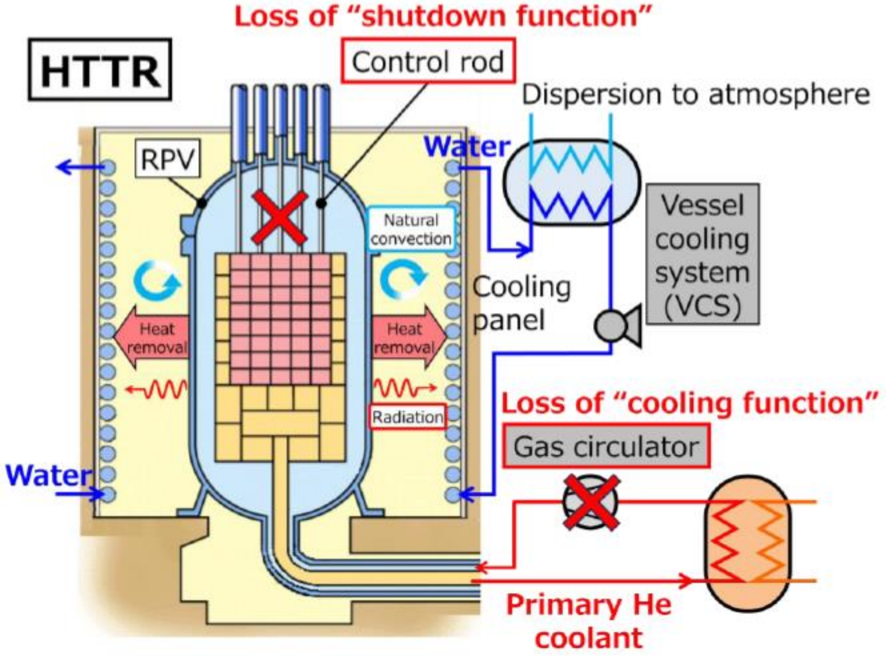
China

- 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.
- 2023: 2 LOFC safety tests were conducted
- **April 2024: China's HTR-PM started heating project operation**
- September 2024: HTR-PM entered its first major maintenance period
- 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.
- Another version is HTR-PM600S for cogeneration having HTR-PM modules with some minor changes.
- August 2025: the safety design training on HTR-PM600 for regulatory personnel was conducted
- August 2024: the **Jiangsu Xuwei nuclear heat project was approved to build 2 Hualong 1 units and 1 HTR-PM600S unit; review for the construction licence of HTR-PM600S started**
- Further development of HTR-PM for elevated temperatures is on-going (900-950C) looking at extending qualification of fuel performance and key component development.
- For H2 production, for I-S and hybrid sulphur processes, basic research and testing in laboratory were completed

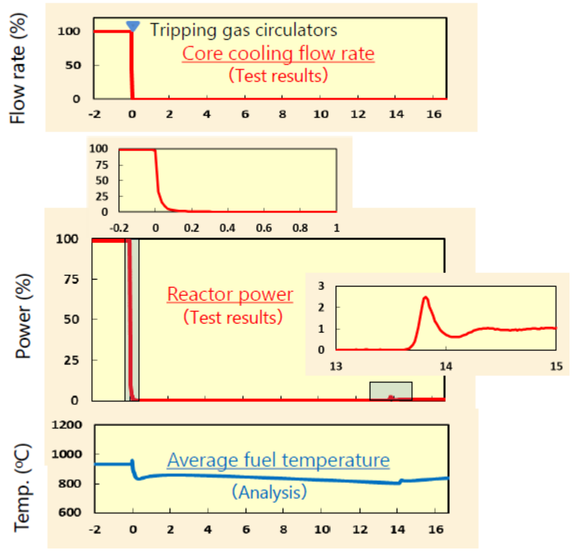


Japan

- 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

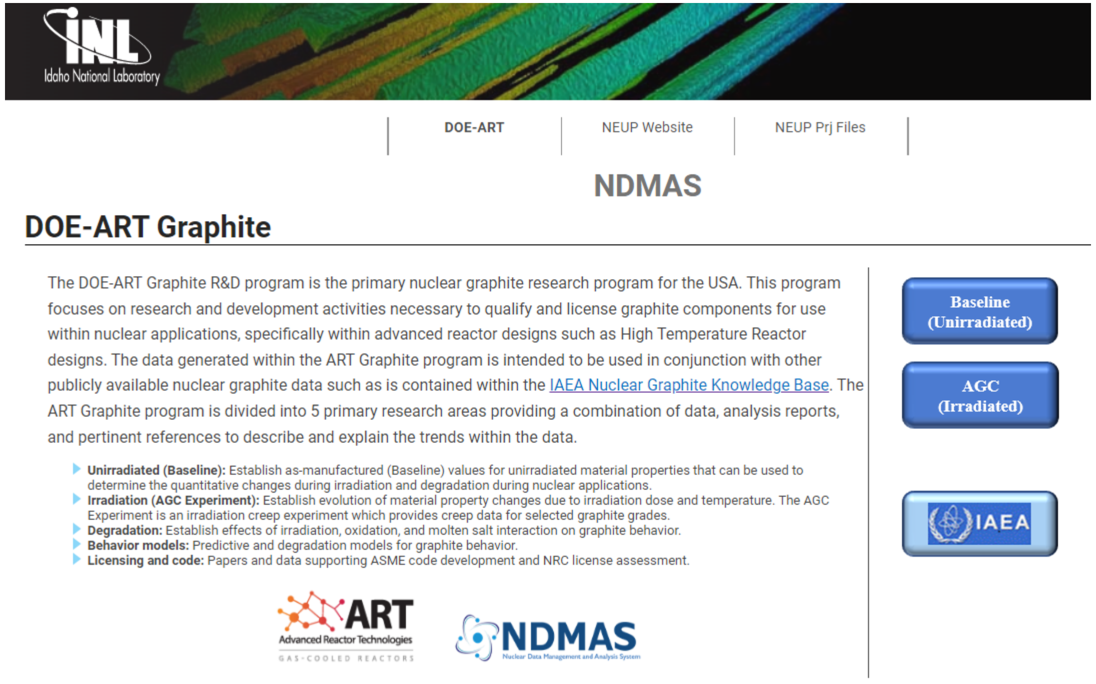


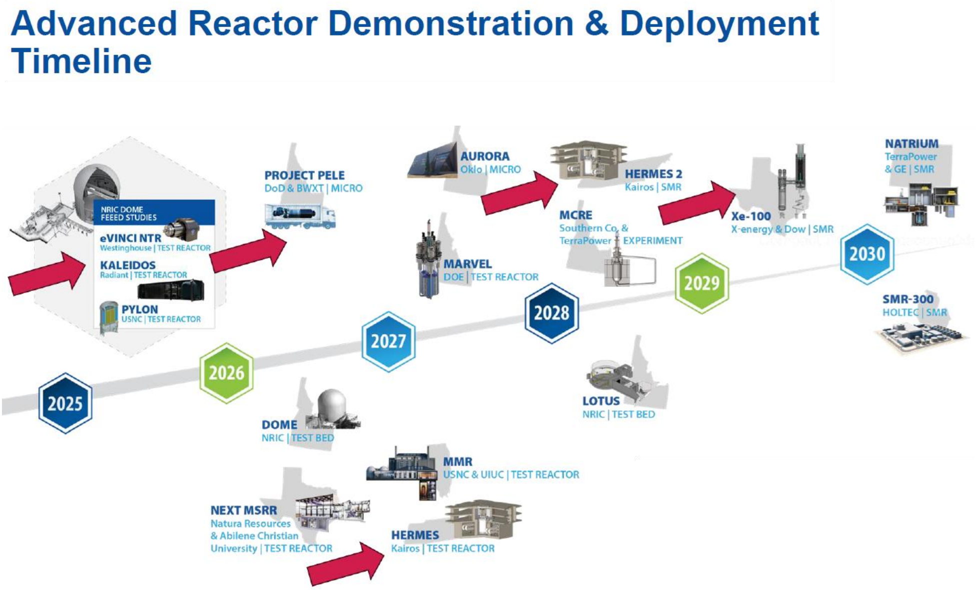
Loss of forced cooling test (All HGC stopped at 100% (30MWt)) March 2024



United States

- Over the past 21 yrs DOE allocated 625M USD to HTGR R&D
- No “national” HTGR is being developed since NGNP (2005-2012)
- DOE 50/50 cost-share demonstration awards to commercial HTR designers – for XE Xe-100: $1,230M over 7 years to build a 2-unit Xe-100 demonstration plant
- XE will site first 4 units at Dow’s Chemical facility in Seadrift, Texas by 2029
- Utility Energy Northwest and XE plan to deploy up to 12 units in central Washington State, with first unit online by 2030
- 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 database [NDMAS - DOE-ART Graphite]
- The activities on metals are focused to achieve ASME codification of alloys and design methods for high-temperature use in pressure vessels, heat exchangers, and other primary circuit components
- US supports experiments (HTTF, NSTF) and international collaboration for V&V of HTGR modeling and simulation tools
- Amazon announced investments in X-energy, with a goal of deploying up to 5 GW of its Xe-100 in US by 2039 (~62 units) for datacenter supply
- XE and Canadian power producer TransAlta Corporation will study feasibility of deploying Xe-100 at a repurposed fossil fuel power plant in Alberta
- Radiant, USNC and Westinghouse microreactors were awarded $3.9M of DOE funding for front-end engineering and experimental design (FEED) in the DOME facility





Other highlights from TWG-GCR members

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.

Indonesia

- target is to acquire 250 MWe from nuclear power in 2033;
- Since 2015 BATAN has been planning to build a 10 MWt pebble bed HTGR; in 2016 the transfer of RDE conceptual design from NUKEM and OKBM Rosatom was completed;
- PeLUIt-40 (an upscale from the 10 MWt reactor to 40MWt) development for de-dieselization, low-carbon hydrogen production, desalination.

Poland

- **HTGR-POLA** 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. The reactor will be a prismatic type HTGR (30-40 MWt), with an outlet temperature of 750C.

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
- KAERI and five domestic companies (Posco E&C, Daewoo E&C, Smart Power, SK Ecoplant and Lotte Chemical) in a public-partnership project started to develop HTGR for industrial process heat (HECTAR, 90 MWt), aiming to have the HTGR basic design and business plan completed by 2027.

Russian Federation

- HTGR technology development in Russia started in the 70s for electricity and heat generation for ammonia production. Several HTGR deigns 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

South Africa

- North West University is conducted a study on the feasibility to restart the PBMR project (Oct 2024-Nov 2025).
- STLN designs: 100 MWt and 30 MWt pebble bed HTGR, and HTR fuel design and manufacturing using U and U/Th

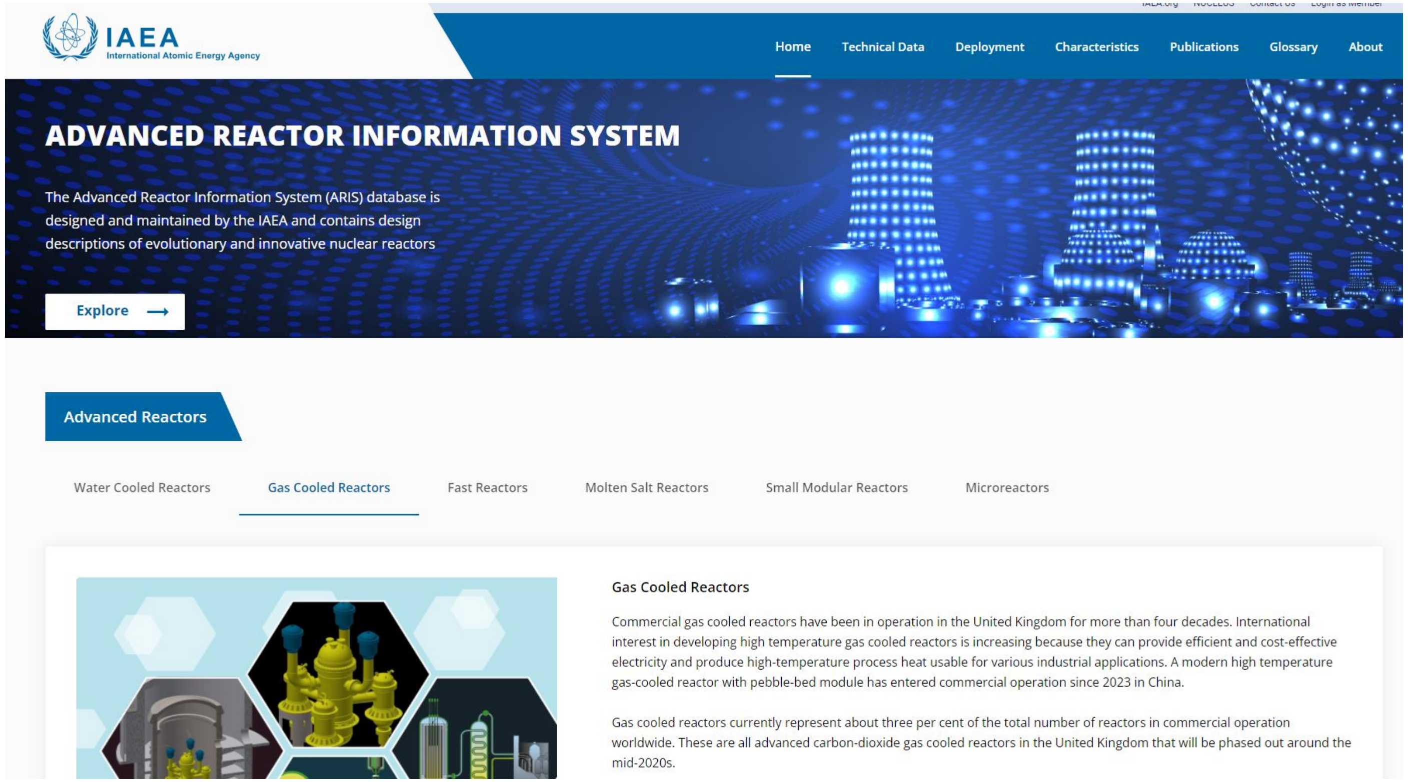
United Kingdom

- UK government funding is mainly directed towards advanced reactor systems, especially HTGRs
- JAEA-NNL collaboration on a pilot-scale TRISO fuel manufacturing capability; Urenco to build a HALEU enrichment facility by 2031

IAEA Advanced Reactor Information System (ARIS)

[Advanced Reactor Information System | Aris (iaea.org)](https://aris.iaea.org)

- Web-accessible database that provides Members States with balanced, comprehensive and up-to-date information about advanced nuclear plant designs and concepts
    - WCRs, GCRs, FRs, MSRs, SMRs, Microreactors
- A tool for Member States at various stages of nuclear power development, offering standardized, impartial data on reactor designs, including evolutionary and innovative concepts, to support informed reactor technology assessments



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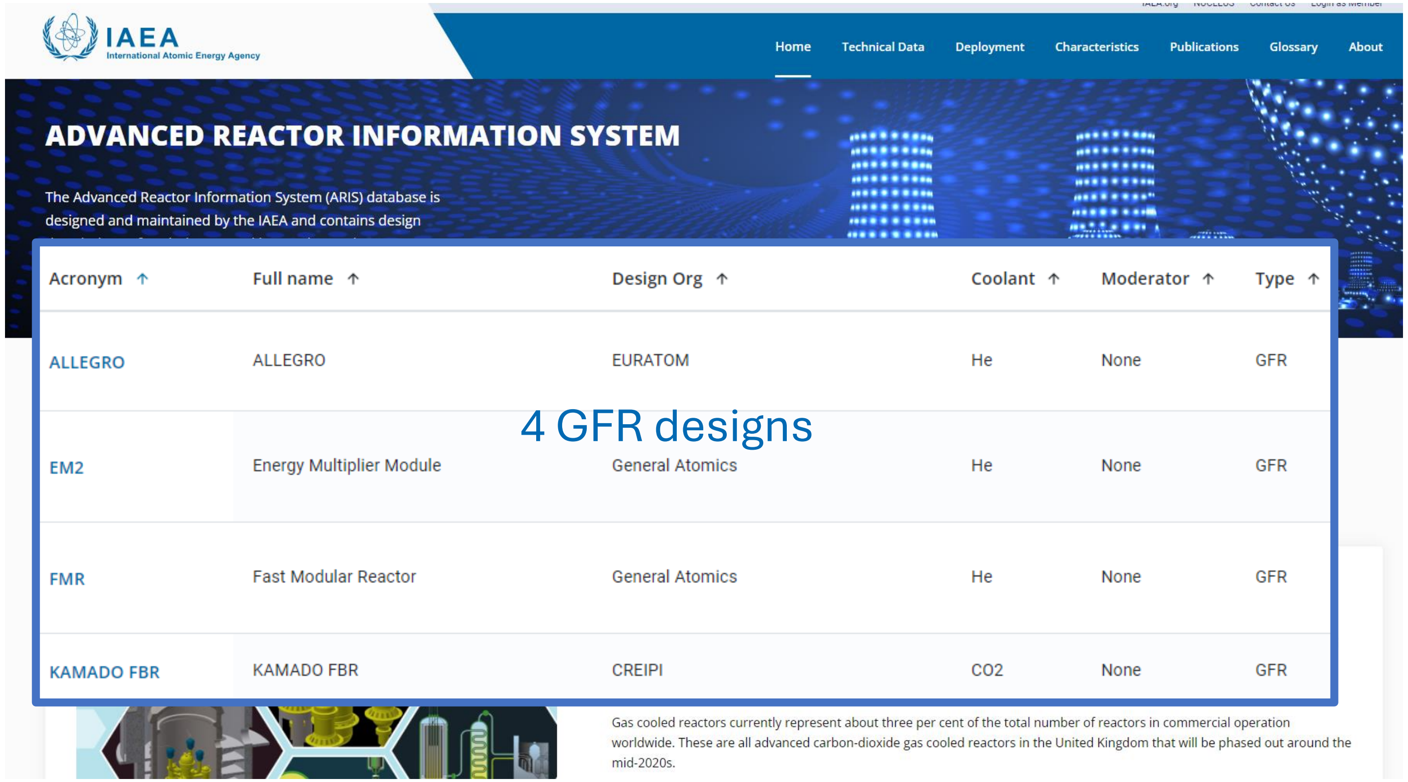
21 GCR designs

Acronym	Full name	Design Org	Coolant	Moderator	Type
AMR	Advanced Micro Reactor	STL Nuclear (Pty) Ltd	He	Graphite	GCR
GT-MHR	Gas Turbine Modular Helium Reactor	SC "Afrikantov OKBM"	He	Graphite	GCR
GTHTR300	Gas Turbine High Temperature Reactor	JAEA, MHI, Toshiba/IHI, Fuji Electric, KHI, NFI	He	Graphite	GCR
HOLOS-MONO	HOLOS-MONO	HolosGen LLC	He	Graphite	GCR
HOLOS-QUAD	HOLOS-QUAD	HolosGen LLC	He	Graphite	GCR
HTGR-POLA	High Temperature Gas-cooled Reactor POLish Atomic	National Centre for Nuclear Research (NCBJ)	He	Graphite	GCR
HTMR100	High Temperature Modular Reactor	STL Nuclear (Pty) Ltd	He	Graphite	GCR
HTR-10	HTR-10	INET, Tsinghua University	He	Graphite	GCR
HTR-PM	High Temperature GCR - Pebble-Bed Module	INET, Tsinghua University	He	Graphite	GCR
HTR50S	HTR50S	JAEA	He	Graphite	GCR
HTTR	High Temperature Engineering Test Reactor	JAEA	He	Graphite	GCR
JIMMY	JIMMY Generator	Jimmy Energy	He	Graphite	GCR
MHR-100	MHR-100	JSC "Afrikantov OKBM"	He	Graphite	GCR
MHR-T	MHR-T Reactor	JSC "Afrikantov OKBM"	He	Graphite	GCR
MMR	Micro Modular Reactor	Ultra Safe Nuclear Corporation	He	Graphite	GCR
PBMR	Pebble Bed Modular Reactor	Pebble Bed Modular Reactor (Pty) Limited	He	Graphite	GCR
PeLUIt-40	PeLUIt-40	National Research and Innovation Agency (BRIN) & Bandung Institute of Technology (ITB)	He	Graphite	GCR
Prismatic HTR	Prismatic Modular High Temperature GCR	General Atomics	He	Graphite	GCR
Pylon D1	Ultra Safe Nuclear Corporation	Ultra Safe Nuclear Corporation	He	Zirconium Hydride	GCR
SC-HTGR	Steam Cycle High Temperature Gas-cooled Reactor	Framatome	He	Graphite	GCR
Xe-100	Xe-100	X-energy	He	Graphite	GCR



4 GFR designs

Acronym	Full name	Design Org	Coolant	Moderator	Type
ALLEGRO	ALLEGRO	EURATOM	He	None	GFR
EM2	Energy Multiplier Module	General Atomics	He	None	GFR
FMR	Fast Modular Reactor	General Atomics	He	None	GFR
KAMADO FBR	KAMADO FBR	CREIPI	CO2	None	GFR

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
CHINA											
HTR-PM (Tsinghua University)	2 × 250 / 210	**In operation**	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
HTR-10 (Tsinghua University)	10 / 2.5	**In operation**	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
FRANCE											
JIMMY (JIMMY ENERGY SAS)	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
INDONESIA											
PeLUIt/RDE (BRIN)	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
JAPAN											
GTHTR300 (JAEA Consortium)	<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
HTTR (JAEA)	30 MWt	**In operation**	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.)

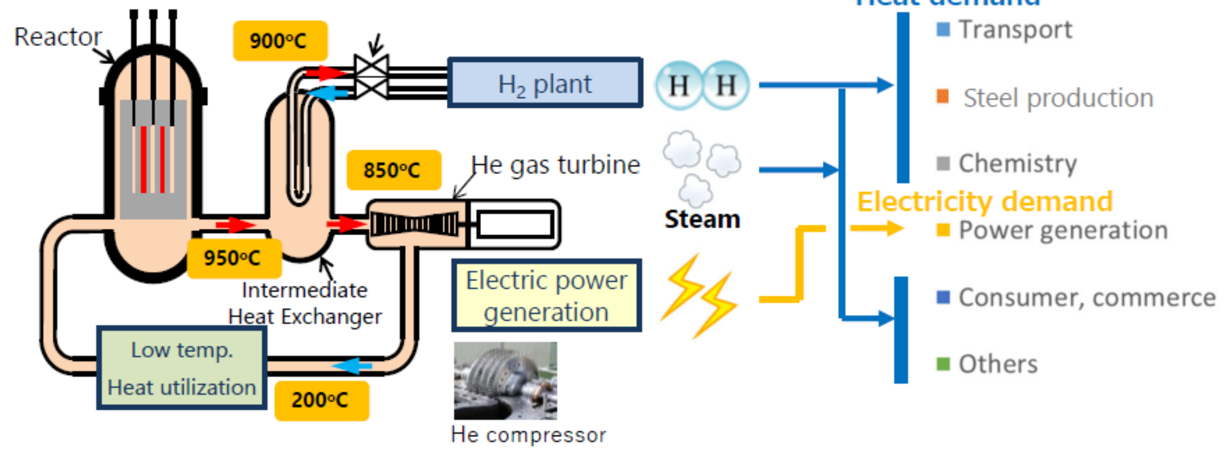
	Thermal/ electrical capacity, MWt/MWe	Design status	Core type	Core inlet/outlet coolant temp (°C)	NSSS Operating Pressure (primary/secondary), (MPa)	Fuel enrichment	Core Discharge Burnup (GWd/t)	Design life (years)	Refuelling cycle (months)	Fuel cycle requirements/ Approach	Distinguishing features
RUSSIAN FEDERATION											
GT-MHR (JSC "Afrikantov OKBM")	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
MHR-T Reactor (JSC "Afrikantov OKBM")	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
MHR-100 (JSC "Afrikantov OKBM)	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
SOUTH AFRICA											
AHTR-100 (Eskom Holdings SOC Ltd.)	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;
PBMR®-400 (PBMR SOC Ltd.)	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
HTMR100 (STL Nuclear (Pty) Ltd.)	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.)

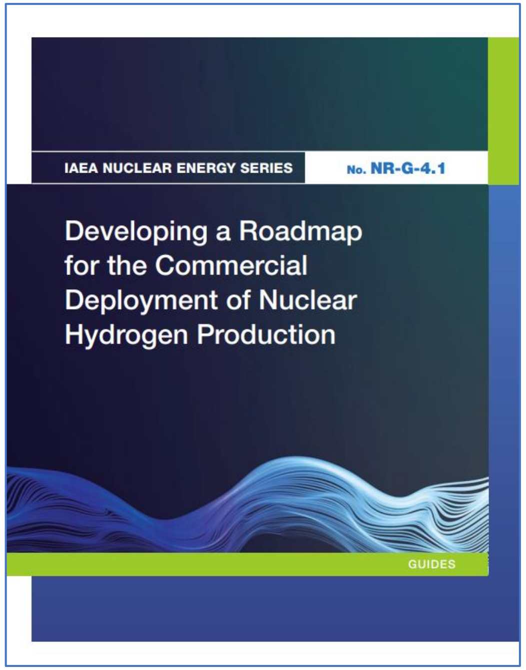
	Thermal/ electrical capacity, MWt/MWe	Design status	Core type	Core inlet/outlet coolant temp (°C)	NSSS Operating Pressure (primary/ secondary), (MPa)	Fuel enrichment	Core Discharge Burnup (GWd/t)	Design life (years)	Refuelling cycle (months)	Fuel cycle requirements/ Approach	Distinguishing features
UNITED STATES											
EM2 (General Atomics)	500 / 265	Conceptual design		550 / 850	13.3	7.7%	130	60	360	Open fuel cycle	Fast reactor, convert and burn
FMR (General Atomics)	100 / 50	Conceptual design		509 / 800	7	19.75%	100	60	96	Open fuel cycle	Silicon carbide composite cladding
Xe-100 (X-energy, LLC)	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
SC-HTGR (Framatome Inc.)	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
CANADA, UNITED KINGDOM AND UNITED STATES											
STARCORE (StarCore Nuclear)	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
POLAND											
HTGR-POLA	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

Hydrogen production using HTGR - Options

HTGR cogeneration system



• Heat utilization rate is about 80%



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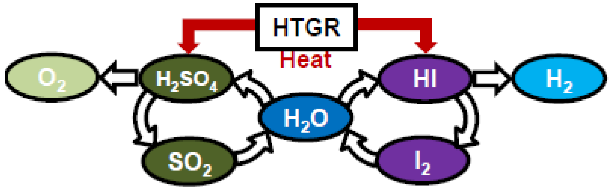
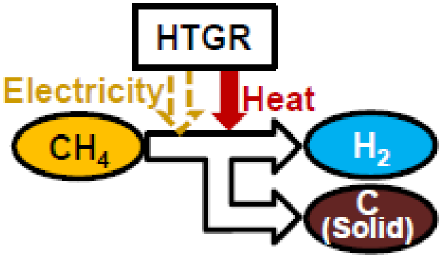
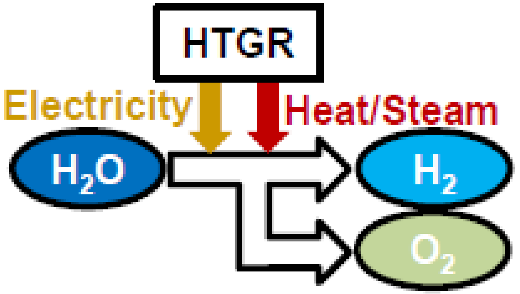
Thermochemical sulphur-iodine (S-I) process



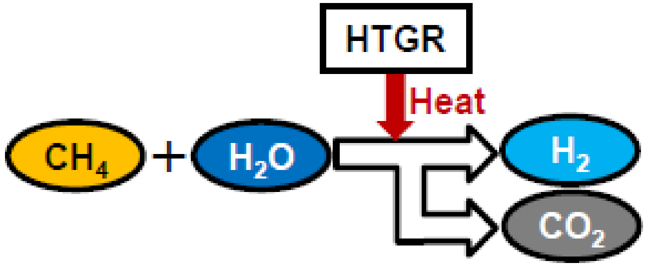
Methane pyrolysis



High temperature steam electrolysis



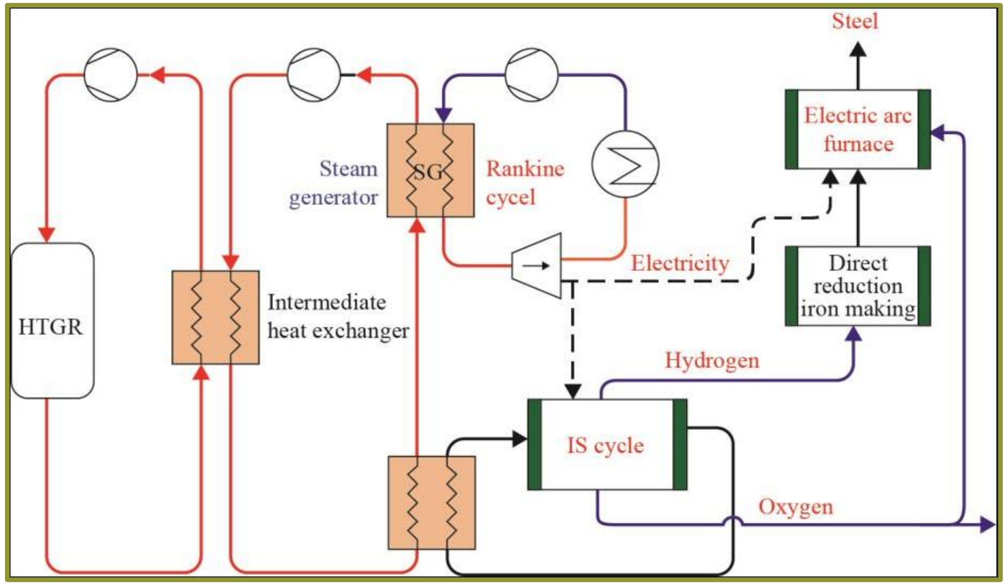
Steam methane reforming



Application prospects of nuclear hydrogen production with HTGRs

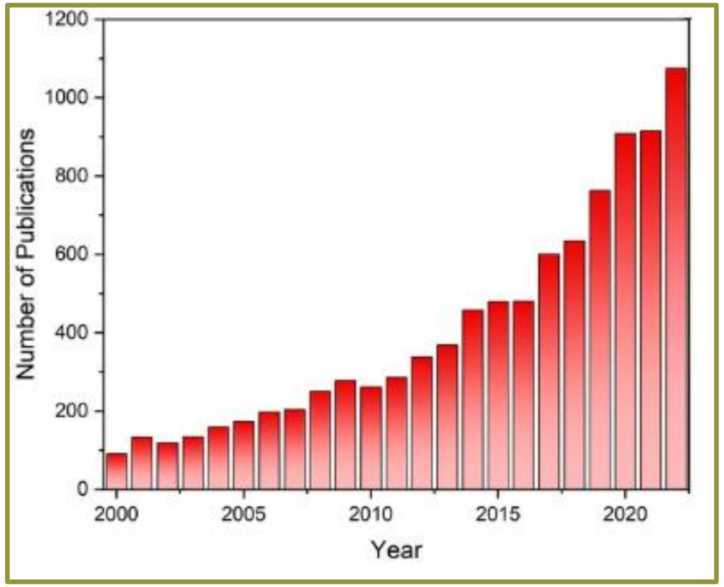
- Some of the applications include the direct reduction of iron, ammonia synthesis, coal liquefaction and refining.

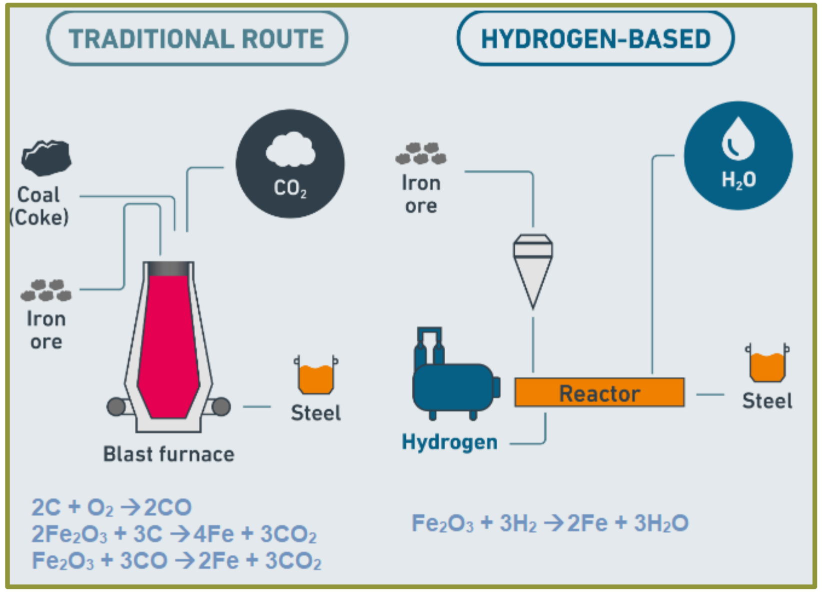
○ ~ 2 billion t steel produced annually, and demand is projected to rise by more than 1/3 by 2050



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

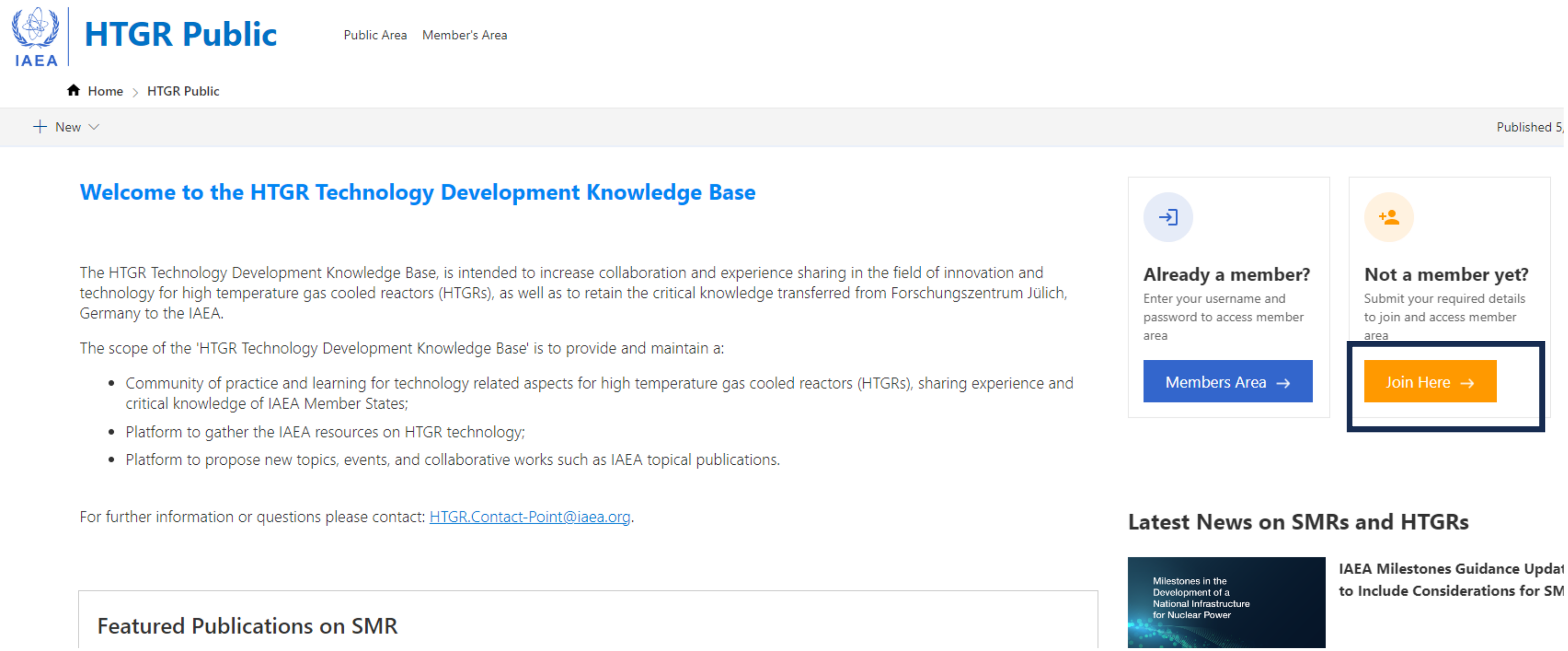
(Source: *Sustainability* **2023**, *15*(17), 13047)



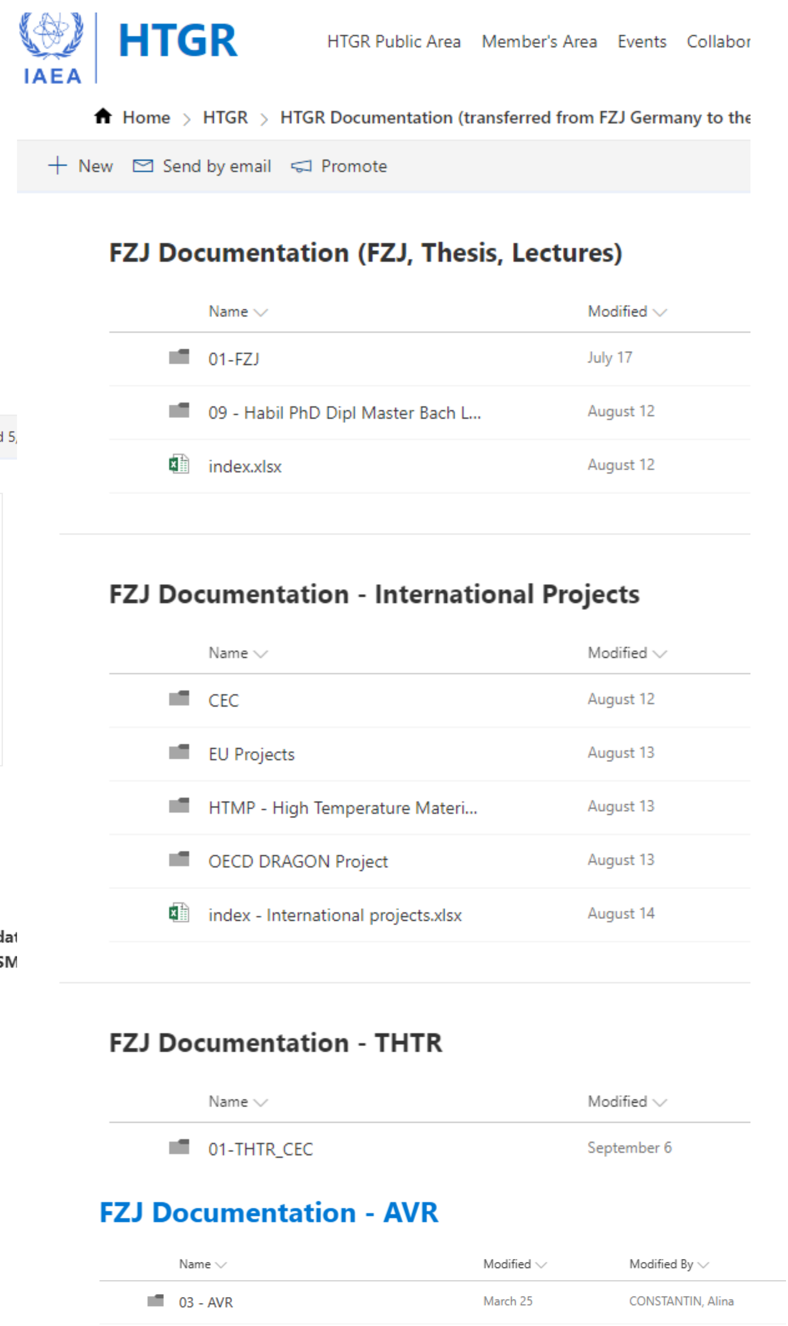


HTGR Technology Development Knowledge Base

HTGR Public - Home



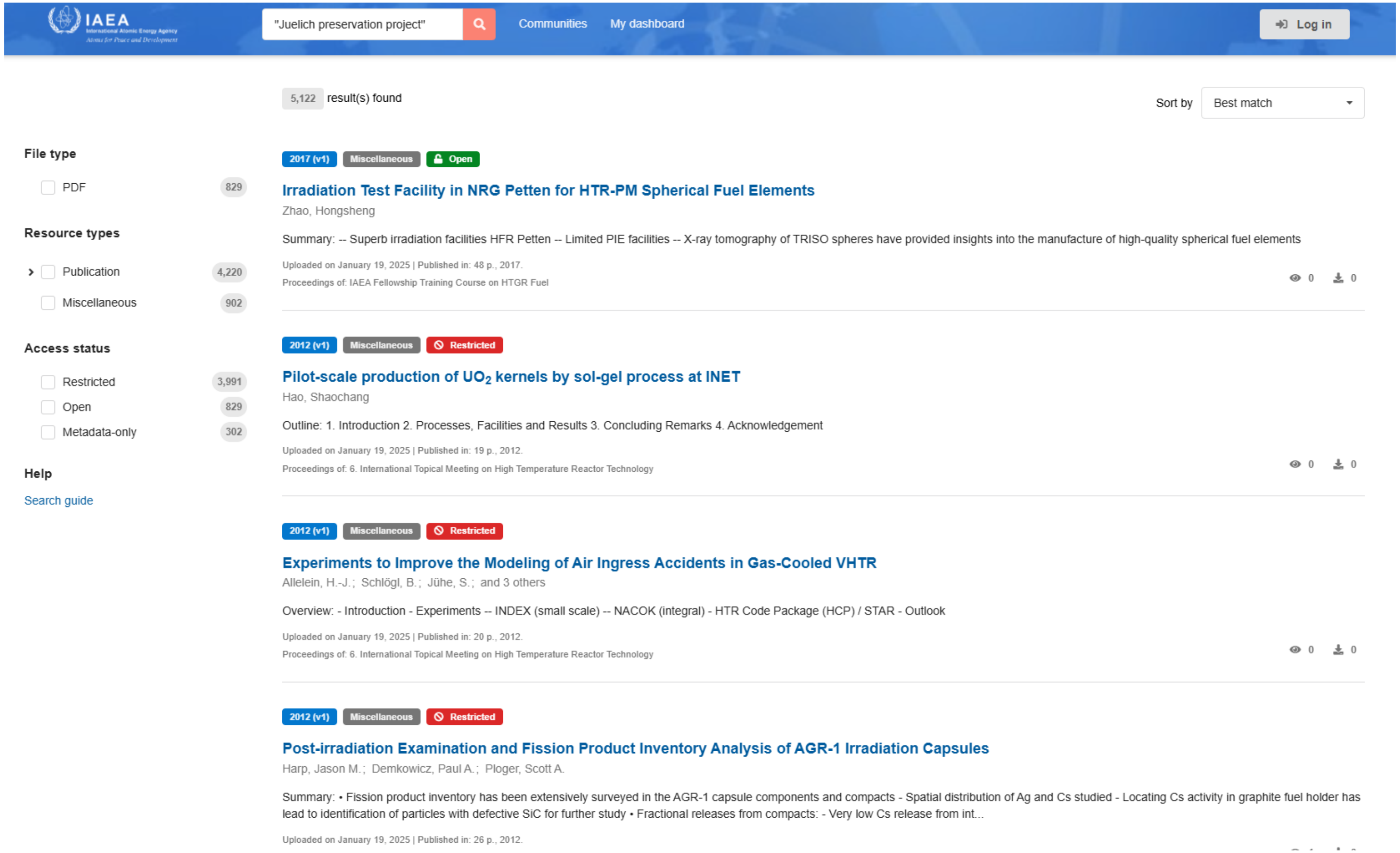
Members Areas (with access based on IAEA nucleus account and approval from the coordinators of the 'HTGR Knowledge Base')



FZJ Archive – indexing 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
- 4500 files from FZJ archive already indexed (as of Dec 2024)
- Search for “Juelich Preservation Project”



HTGR Technology Development Knowledge Base

Resources

HTR codes documentation | HTR2024 | Publications | Experimental Facilities

HTGR Code Documentation

IAEA HCP Workshop in Jakarta 2017

Name	Modified
Presentations	March 20
Tutorials	March 20

HTR2024

The Proceedings of the Conference are also available through IAEA INIS at The 11th International Topical Meeting on High Temperature Reactor Technology.

HTR2024 Papers

See all

Name	Modified	Modified By
100 - Particle deposition characte...	December 16, 2024	CONSTANTIN, Alina
102 - Influences of Boundary Con...	December 16, 2024	CONSTANTIN, Alina
103 - A Comparative Analysis on ...	December 16, 2024	CONSTANTIN, Alina
104 - CP Thermodynamics and Ca...	December 16, 2024	CONSTANTIN, Alina
106 - Simulation Modelling of Sec...	December 16, 2024	CONSTANTIN, Alina
108 - Project plan of HTTR heat a...	December 16, 2024	CONSTANTIN, Alina
111 - Commissioning of HTR-PM ...	December 16, 2024	CONSTANTIN, Alina

HTR2024 Proceedings – indexed in INIS

The 11th International Topical Meeting on High Temperature Reactor Technology

IAEA International Atomic Energy Agency Atoms for Peace and Development

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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

Contact

Inquire about this record

Details

Resource type
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...
- Simplifying Reactor Protection System...
- Three-dimensional Radiation View Fact...
- Advanced Process Design for HTR TRISO...
- Effects of Urea and Cerium Addition i...
- ...

Find all associated items

Papers Search results

Versions

Version 1.0.0 2024

Citation

HTGR Technology Development Knowledge Base

Resources

HTR codes documentation | HTR2024 | Publications | Experimental Facilities

IAEA Publications

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)

•High Temperature Gas Cooled Reactor Technology Development, IAEA-TECDOC-988 (1998)

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

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

•Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to Initial Testing of the HTTR and HTR-10, IAEA-TECDOC-1382 (2003)

•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)

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)

•Status and Prospects for Gas Cooled Reactor Fuels, IAEA TECDOC (CD-ROM) No. 1614 (2009)

•High Temperature Gas Cooled Reactor Fuels and Materials, IAEA TECDOC (CD-ROM) No. 1645 (2010)

•Advances in High Temperature Gas Cooled Reactor Fuel Technology, IAEA TECDOC (CD-ROM) No. 1674 (2013)

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

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)

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

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

HTGR Technology Development Knowledge Base

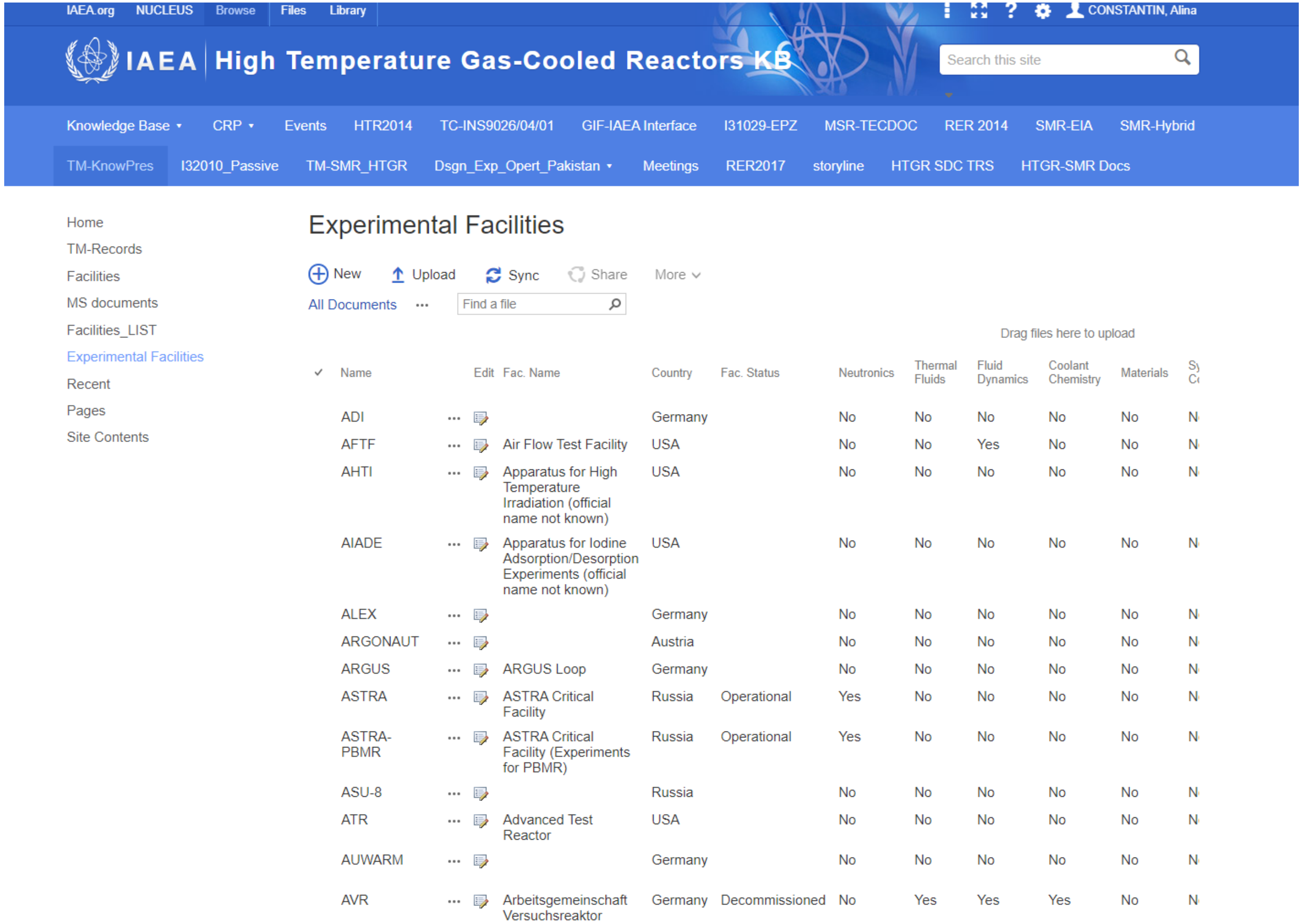
Resources

HTR codes documentation | HTR2024 | Publications | Experimental Facilities

Experimental Facilities

Name	Fac. Name	Country	Fac. Status	Neutronics	Thermal Fluids	Fluid Dynamics	Coolant Chemistry	Materials	Sys. & Comp.
ADI.docx		Germany		No	No	No	No	No	No
AFTF.docx	Air Flow Test Facility	USA		No	No	Yes	No	No	No
AHTI.docx	Apparatus for High Temperature Irradiation (official name not known)	USA		No	No	No	No	No	No
AIADE.docx	Apparatus for Iodine Adsorption/Desorption Experiments (official name not known)	USA		No	No	No	No	No	No
ALEX.docx		Germany		No	No	No	No	No	No
ARGONAUT.docx		Austria		No	No	No	No	No	No
ARGUS.docx	ARGUS Loop	Germany		No	No	No	No	No	No
ASTRA.docx	ASTRA Critical Facility	Russia	Operational	Yes	No	No	No	No	No
ASTRA-PBMR.docx	ASTRA Critical Facility (Experiments for PBMR)	Russia	Operational	Yes	No	No	No	No	No
ASU-8.docx		Russia		No	No	No	No	No	No
ATR.docx	Advanced Test Reactor	USA		No	No	No	No	No	No

Experimental facilities database – 174 entries



Name	Edit	Fac. Name	Country	Fac. Status	Neutronics	Thermal Fluids	Fluid Dynamics	Coolant Chemistry	Materials	Sy Co
ADI			Germany		No	No	No	No	No	N
AFTF		Air Flow Test Facility	USA		No	No	Yes	No	No	N
AHTI		Apparatus for High Temperature Irradiation (official name not known)	USA		No	No	No	No	No	N
AIADE		Apparatus for Iodine Adsorption/Desorption Experiments (official name not known)	USA		No	No	No	No	No	N
ALEX			Germany		No	No	No	No	No	N
ARGONAUT			Austria		No	No	No	No	No	N
ARGUS		ARGUS Loop	Germany		No	No	No	No	No	N
ASTRA		ASTRA Critical Facility	Russia	Operational	Yes	No	No	No	No	N
ASTRA-PBMR		ASTRA Critical Facility (Experiments for PBMR)	Russia	Operational	Yes	No	No	No	No	N
ASU-8			Russia		No	No	No	No	No	N
ATR		Advanced Test Reactor	USA		No	No	No	No	No	N
AUWARM			Germany		No	No	No	No	No	N
AVR		Arbeitsgemeinschaft Versuchsreaktor	Germany	Decommissioned	No	Yes	Yes	Yes	No	N

Austria
Belgium
China
France
Germany
Japan
Netherlands
Russian Federation
South Africa
UK
USA

Includes also closed, decommissioned, in care and maintenance facilities

Experimental Facilities - All Documents

Template for the compilation of the catalogue
“EXPERIMENTAL FACILITIES SUPPORTING HIGH TEMPERATURE REACTORS”

Template for the compilation of the catalogue
“EXPERIMENTAL FACILITIES SUPPORTING HIGH TEMPERATURE REACTORS”

*Field is mandatory

GENERAL INFORMATION

*NAME OF THE FACILITY	ASTRA Critical Facility
*ACRONYM	ASTRA
*MAIN PURPOSE	Studies of neutronic characteristics of different HTGR core configurations; acquisition of experimental data for validation of calculational models and codes
*MEMBER STATE (country):	Russia
*OPERATOR / OWNER	National Research Centre “Kurchatov Institute”
LOCATION (address):	Moscow, Russia
OPEN TO COOPERATION?	Yes
*CONTACT PERSON(S) (name, address, institute, function, telephone, email):	FOMICHENKO Peter, Deputy head of complex, Kurchatov sq., 1, 123182, Moscow, Russia +7(499)196-74-79 Fomichenko_PA@nrcki.ru

***STATUS OF THE FACILITY**	Operational
Start of operation (period):	1980's
Exact dates facility operated:	1980-2010

***MAIN RESEARCH FIELDS**

true	Neutronics / Zero Power Facility
false	Thermal-Fluids
false	Fluid Dynamics
false	Coolant Chemistry
false	Materials
false	Systems and Components (including design decisions)
false	Instrumentation & ISI&R
false	Fuel Manufacturing
false	Fuel Irradiation and Testing (PIE)
true	For V&V and Licensing Purposes (including code V&V and component testing)
false	Design Basis Accidents (DBA) and Design Extended Conditions (DEC)
false	Fission Product Transport
false	Fuel and Fission Products (including spent fuel experiments)
false	Cross Cutting (several advanced reactor systems (GIF systems))
	Other: [Other Research]

TECHNICAL DESCRIPTION

Description of the facility

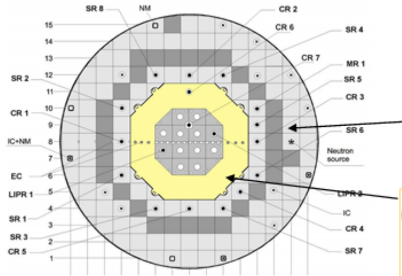
The ASTRA critical facility is intended for experimental justification of neutronic characteristics of various HTGR core configurations, as well as acquisition of experimental data for validation of calculational models and codes. The ASTRA critical assemblies use medium-enriched uranium dioxide fuel (about 21% of U-235) in the form of fuel particles with multilayer coating distributed in the graphite matrix of spherical fuel elements. The set of spherical elements available at the facility includes absorber and graphite spheres, and all of them have the outer diameter of 60 mm. The critical facility has a cylindrical steel vessel with a bottom. Graphite blocks of the lower and side reflectors are installed inside the vessel. The critical assembly for experiments at room temperature includes the side graphite reflector in the form of a cylinder of 380 cm outer diameter and 460 cm high, with a cavity inside, where an inner graphite reflector can be installed. The space formed between the inner and side reflectors is filled with fuel spheres, thus producing the annular core. The reflectors are made of graphite blocks of square cross-section, and each graphite block has a central hole where a graphite plug, a control rod, neutron flux detectors, a leave-in-place poison rod etc. can be inserted. The inner and top reflectors can also be made of graphite spheres.

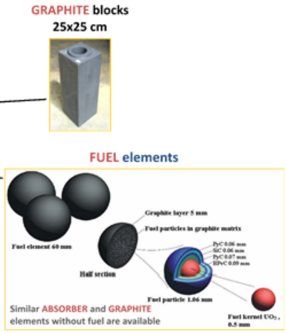
Acceptance of radioactive material

Yes

Click here to enter text.

Scheme/diagram





CR1..CR7 – compensating control rods; EC – experimental channels; MR1 – manual control rod; LIPR1, LIPR2 – leave-in-place poison rods; SR1..SR8 – safety rods; IC – ionization chambers; NM – neutron counters

3D drawing/photo

GT-MHR core mockup

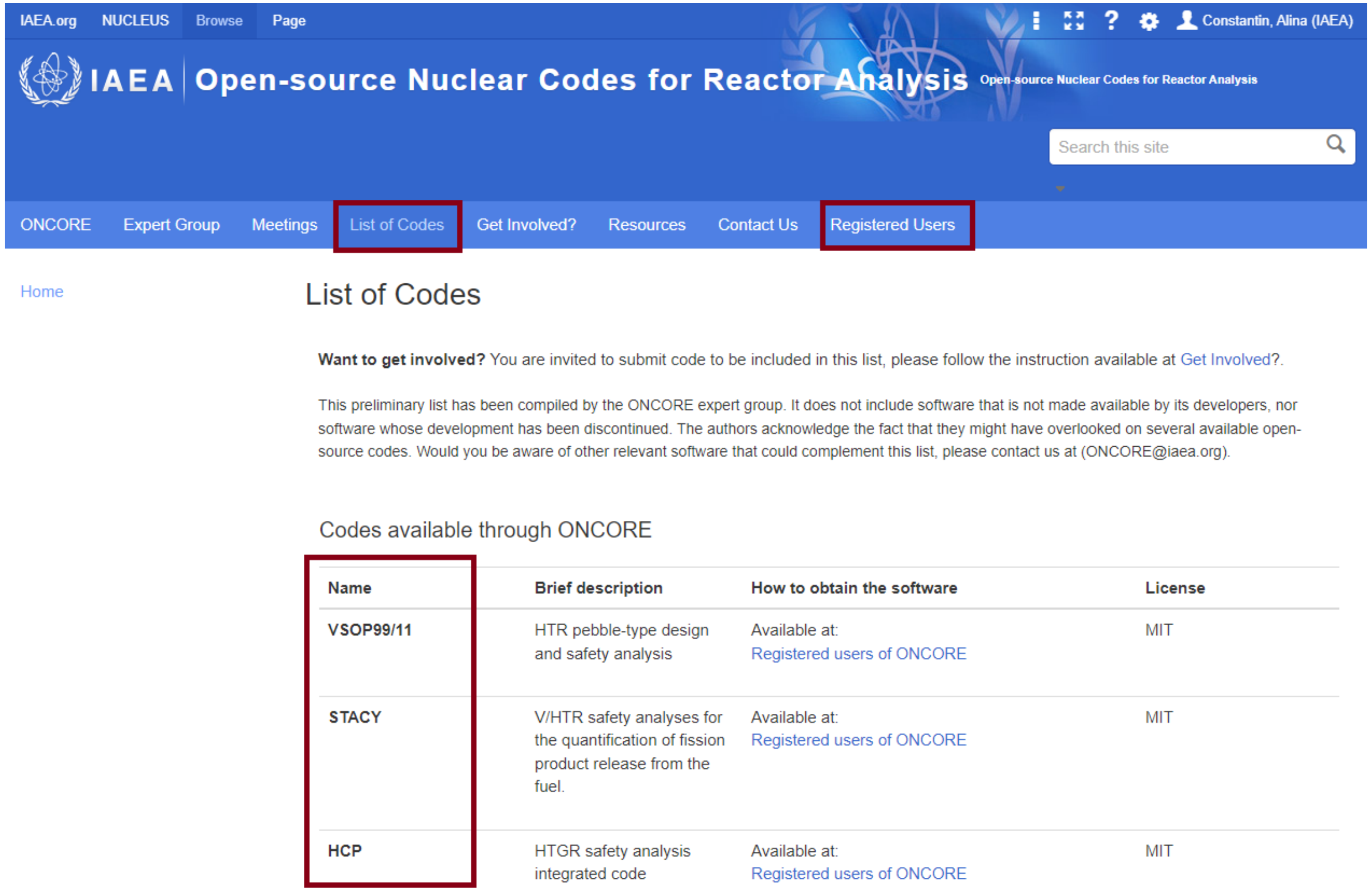
PBMR core mockup

Parameters table

Coolant	Ambient air for room temperature experiments Inert gas in the heated pebble bed (planned after modernization)
Temperature	Room temperature Plans to introduce heaters / heat up the facility has been prepared
Pressure	Atmospheric
Radioactive substances	Click here to enter text.
Nuclear material	Pebble type coated particle fuel
Medium (Coolant) inventory	Click here to enter text.
Power/ Heater power	100 W neutron power 25 kW electrical heater power (planned after modernization)
Test sections	
TS #1	*Characteristic dimensions* 320 cm height, 8200 l volume pebble bed core, up to 32000 fuel elements, 18000 graphite elements, 2000 absorber elements, side graphite reflector in the form of a cylinder of 380 cm outer diameter and 460 cm high *Static/dynamic experiment* Both; measurements performed: – critical parameters; – spatial power density distributions (reaction rates measurements); – reactivity worth of control rod mockups, their interference factors and calibration curves; – neutron kinetic parameters; neutron gas temperature *Temperature range in the test section (Delta T)* Room temperature Up to 400-600 °C (planned after modernization) *Operating pressure and design pressure* Atmospheric *Flow range (mass, velocity, etc.)* Zero for room temperature experiments Natural convection of inert gas (planned after modernization)

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


IAEA.org NUCLEUS Browse Page
Constantin, Alina (IAEA)
IAEA Open-source Nuclear Codes for Reactor Analysis
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ONCORE Expert Group Meetings List of Codes Get Involved? Resources Contact Us Registered Users
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).

Codes available through ONCORE

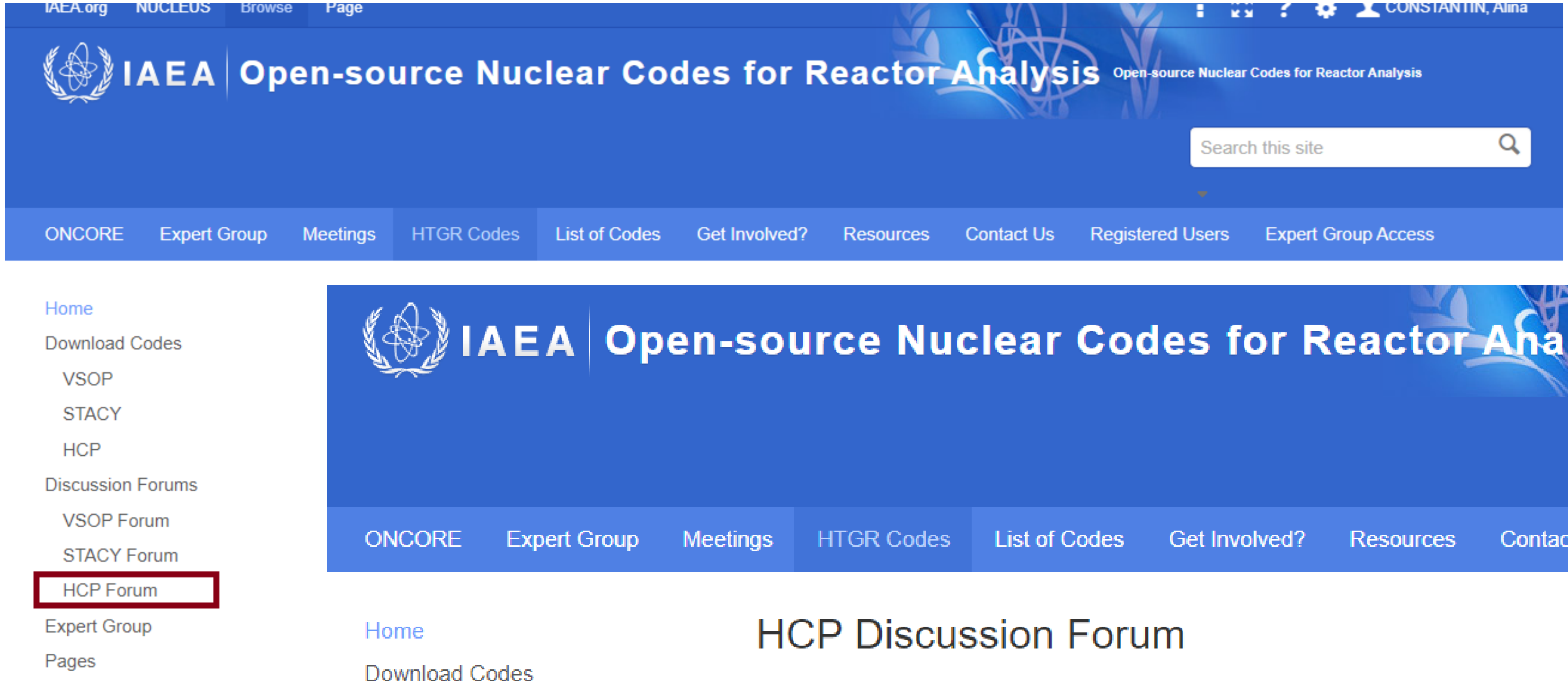
Name	Brief description	How to obtain the software	License
VSOP99/11	HTR pebble-type design and safety analysis	Available at: Registered users of ONCORE	MIT
STACY	V/HTR safety analyses for the quantification of fission product release from the fuel.	Available at: Registered users of ONCORE	MIT
HCP	HTGR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

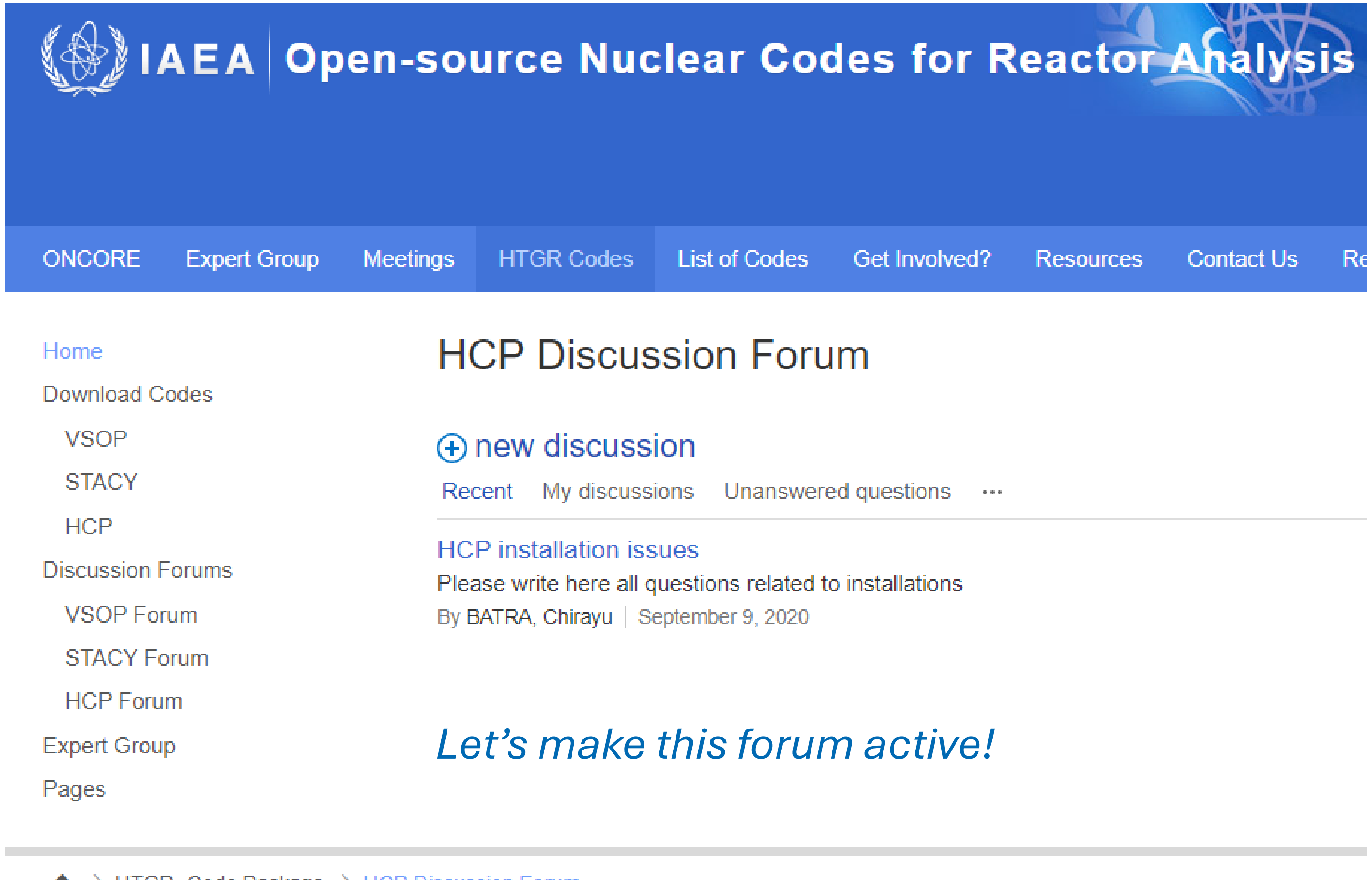
Source code available in Github

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

2025: Workshop on High Temperature Gas Cooled Reactor Technology and Training on the High Temperature Reactor Code Package (2-7 November 2025, hosted by TUM)

HCP Forum





Let's make this forum active!

IAEA Expert mission on the High Temperature Gas-cooled Reactor Package (HCP) and STACY code (4-8 November 2024, Bandung, Indonesia)

- Organized locally by the National Research and Innovation Agency (BRIN), at the premises of Bandung Institute of Technology, Bandung (ITB), with IAEA support from TC Programme INS2019
- 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)
  - ❑ Mr Chunyu Liu (Technical University of Munich)
  - ❑ Meryll Colomber (Becker Technologies, Germany)
- 9 Indonesia BRIN staff and 5 online South African participants joined the event
- It is expected that the new version developed by TUM to be transferred to IAEA during 2025 and make available to Member States through the IAEA ONCORE platform
- The need of having a comprehensive documentation of the HCP code and its modules (including training material, developer manual and theoretical manual) was highlighted by the participants; however, currently the developers have limited resources and the focus is to complete the expected enhanced versions of both HCP and STACY
- Participants (both users and developers) agreed on the need to have a dedicated HCP Forum to exchange data, models, work on further developments, report bugs/issues, create test cases

Nuclear Graphite Knowledge Base (NGKB)

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

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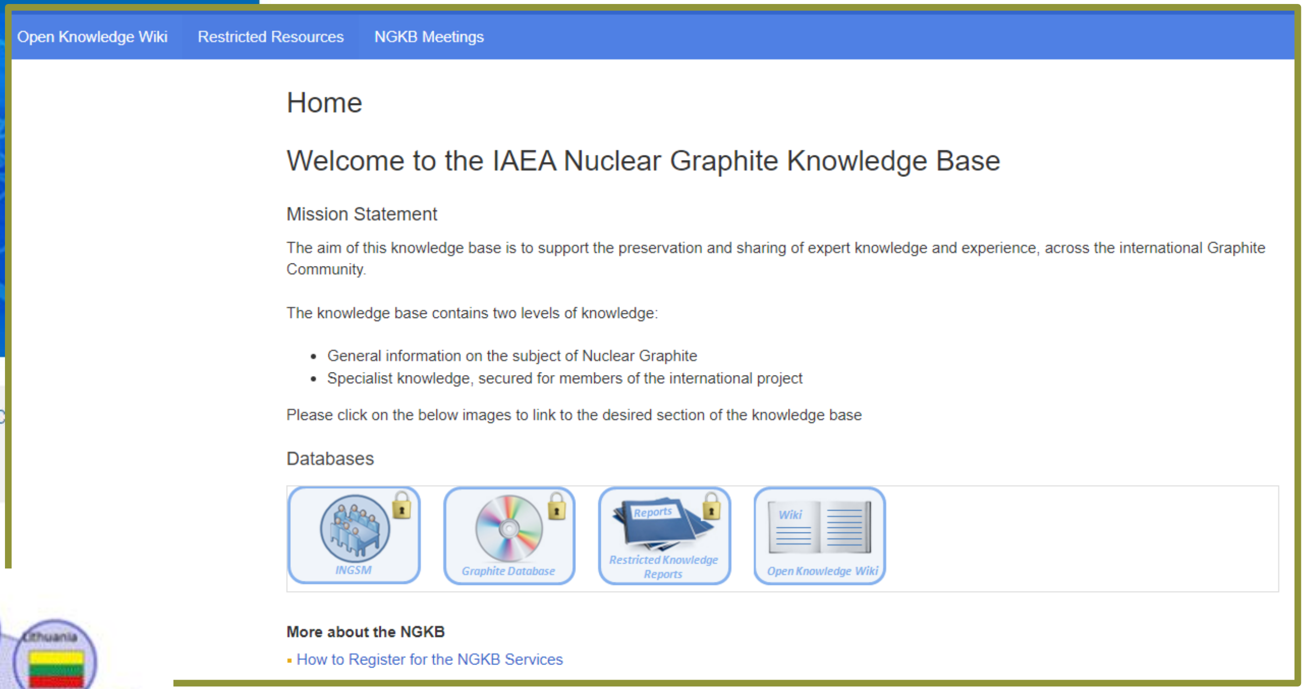
IAEA Nuclear Graphite Knowledge Base

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 Irradia 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

INGSM Graphite Database Restricted Knowledge Reports Open Knowledge Wiki

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- How to Register for the NGKB Services

- Open knowledge wiki
- Restricted resources
- NGKB meetings

International Nuclear Graphite Specialists Meeting (INGSM)

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INGSM-10 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-11 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-12 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-13 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-14 Material	···	December 24, 2013	GLADYSHEV, Maxim	
INGSM-15 Material	···	February 3, 2015	YASAR, Tufan	
INGSM-16 Material	···	November 17, 2015	SANTOSO, Sigit	
INGSM-17 Material	···	September 16, 2016	KIM, Wooseub	
INGSM-18 Material	···	November 20, 2017	KIM, Yongjin	
INGSM-19 Material	···	September 11, 2018	REITSMA, Frederik	
INGSM-2 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-20 Material	···	November 13, 2019	LUQUE GUTIERREZ, Juan Cristhian	
INGSM-21 Material	···	November 27, 2023	Frederik Reitsma	
INGSM-22 Material	···	November 21, 2023	SUBKI, M. Hadid	
INGSM-23 Material	···	November 26, 2023	SUBKI, M. Hadid	
INGSM-3 Material	···	December 24, 2013	GLADYSHEV, Maxim	
INGSM-4 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-5 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-6 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-7 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-8 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
INGSM-9 Material	···	April 23, 2013	ELGHOBARY, Ahmed	
Copy of INGSM_Index	···	September 11, 2018	REITSMA, Frederik	51 KB
INGSM_Index	···	August 9	CONSTANTIN, Alina	847 KB

24th International Nuclear Graphite Specialists Meeting (INGSM)

ingsm 2024 Berkeley UNIVERSITY OF CALIFORNIA

24th International Nuclear Graphite Specialists Meeting

- 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

25th International Nuclear Graphite Specialists Meeting (INGSM)

29 Sept – 3 Oct 2025 (Vienna)

25th International Nuclear Graphite Specialists' Meeting

Abstracts must be received by 30th JUNE 2025

International Network for Experiment and Code Validation Sharing (NEXSHARE)

IAEA **NEXSHARE**   NEXSHARE Public   NEXSHARE Member's Area

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

Home > NEXSHARE

Send by email

Welcome to the IAEA NEXSHARE

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).

The objectives of **NEXSHARE** are to:

- Reduce costs and increase schedule certainty associated with experiments and code validation needed for design and licensing of SMRs.
- Provide greater reach and enhanced utilization of existing experimental facilities.
- Share information on existing experimental programs and identify potentials for resource sharing and synergies.
- Provide greater confidence on data and codes used in safety cases by establishing globally recognized common approaches and best practices in testing and code validation.

For further information or questions please contact NEXSHARE.Contact-Point@iaea.org

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Members Area →

Not a member yet?

Submit your required details to join and access member area

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NEA NUCLEAR ENERGY AGENCY

GEN IV International Forum Expertise | Collaboration | Excellence

***This Network is currently under construction. It is open to members for proposals and recommendations on its format, content and functionalities*

International Network for Experiment and Code Validation Sharing (NEXSHARE)

IAEA NEXSHARE

NEXSHARE Public NEXSHARE Member's Area

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

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About NEXSHARE | Participating Vendors | Participating Experimental Facilities | Resources

Collaboration Projects | Events | Contact Us | Steering Committee

International Organizations and Technical Support Organizations (TSOs).

The objectives of **NEXSHARE** are to:

- Reduce costs and increase schedule certainty associated with experiments and code validation needed for design and licensing of SMRs.
- Provide greater reach and enhanced utilization of existing experimental facilities.
- Share information on existing experimental programs and identify potentials for resource sharing and synergies.
- Provide greater confidence on data and codes used in safety cases by establishing globally recognized common approaches and best practices in testing and code validation.

For further information or questions please contact NEXSHARE.Contact-Point@iaea.org

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Facility	Full Name	Country Name ↑	Organization	Reactor family	Fluid	Application	Operating conditions	Max Power (kW)
Bell Jar	CNL Vacumm Bell Jar Apparatus	Canada	CNL	High Temperature Gas Cooled, Wa...	N/A	Severe Accident, Components/Feature, Materials	Accident conditions	50
HANC	Helium-Air Natural Convection	Canada	CNL	High Temperature Gas Cooled	helium and air	Containment, Thermalhydraulics - separate effects	Accident conditions	130
BTF	Burst Test Facility	Canada	Kinectrics Inc.	High Temperature Gas Cooled, Wa...	Water/inert gas/Hydrogen/Helium (environment can change as required)	Components/Feature, Materials, Severe Accident	Accident conditions, Normal operation	
HTTR	High Temperature Engineering Test Reactor	Japan	Japan Atomic Energy Agency	High Temperature Gas Cooled	Helium gas	Neutronics	Normal operation	30,000
HITHEF	High-Temperature Helium Experimental Facility for HTGRs	United States of America	University of Michigan	High Temperature Gas Cooled	Helium	Components/Feature, Thermalhydraulics - separate effects	Normal operation	52
GOTeF	Graphite Oxidation Test Facility	United States of America	University of Michigan	High Temperature Gas Cooled	Helium	Materials	Accident conditions, Normal operation	
MAGNET/He-CTF	Microreactor AGile Non-nuclear Experimental Test bed/Helium Component Test Facility	United States of America	Idaho National Laboratory	High Temperature Gas Cooled	Helium, Nitrogen	Components/Feature, Thermalhydraulics - integral effects, Thermalhydraulics - separate effects	Normal operation	250
PBRT	Pebble Bed Reactor Test facility	United States of America	Purdue University	High Temperature Gas Cooled	Helium	Thermalhydraulics - integral effects, Thermalhydraulics - separate effects	Accident conditions, Normal operation	
EXHALE	Experimental Helium Analysis and Laboratory for Extreme Conditions	United States of America	Texas A&M University	High Temperature Gas Cooled	Helium	Severe Accident, Thermalhydraulics - separate effects	Accident conditions, Normal operation	1.2

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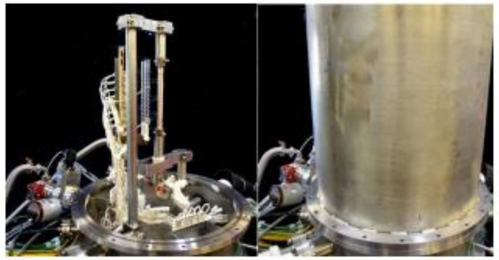
BT			ature, Materials, Severe	Accident conditions, Normal operation	
HT				Normal operation	30,000
HI		s	ature, Thermalhydraulics s	Normal operation	52
GC		s		Accident conditions, Normal operation	
MA CTF	Experimental Test bed/Helium Component Test Facility	s	ature, Thermalhydraulics , Thermalhydraulics -	Normal operation	250
PBRT	Pebble Bed Reactor Test facility	United States	cs - integral effects, cs - separate effects	Accident conditions, Normal operation	
EXHALE	Experimental Helium Analysis and Laboratory for Extreme Conditions	United States	Thermalhydraulics -	Accident conditions, Normal operation	1.2

- Operator
- Applications
- Detailed technical description
- Past and planned experiments

CNL Vacuum Bell Jar Apparatus

Canadian Nuclear Laboratories
Laboratoires Nucléaires Canadiens



GENERAL INFORMATION	
Acronym:	Bell Jar
Facility name:	CNL Vacuum Bell Jar Apparatus
Organization:	Canadian Nuclear Laboratories
Country:	Canada
Operational since:	2008
Reactor family:	☒ Water Cooled ☒ High Temperature Gas Cooled ☐ Molten Salt ☐ Liquid Metal Cooled Fast Spectrum Other:
Website/Contact email:	www.cnl.ca/ chuk.azih@cnl.ca

TECHNICAL DESCRIPTION

Operating conditions modelled: ☐ Normal operation ☒ Accident conditions
please specify:

Application:

☐ Thermalhydraulics - separate effects ☐ Thermalhydraulics - integral effects ☐ Neutronics
☐ Fuel ☒ Components/Feature ☒ Materials
☐ Containment ☒ Severe Accident ☐ Hazards (seismic)

Other:

Phenomena modelled:

Radiation heat transfer

Key facility parameters:

Fluid: Vacuum, oxidizing gas atmosphere (e.g., helium-oxygen mixture, CO_2, oxygen mixture).
Dimensions/volume: 1.4×10^6 mm^2 surface area.

	Min	Max
Facility power [kW]:	0	4.8 - 50*
Pressure [MPa]:	1.33×10^{-8}	0.1
Temperature [°C]:	0	1100 - 1400*

* Indicates upgrades that have been recently implemented but not tested.

Key characteristics for integral effect test facilities:

- Measures total hemispherical emissivity of metals up to 1100°C
- Measures total hemispherical emissivity of graphite up to 1400°C
- Measure total hemispherical emissivity of silicon carbide up to 1400°C
- Measures emissivity under vacuum conditions
- Measures emissivity of coated and oxidized specimens

Last update: 22 January 2024

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Upcoming HTGR Events (rest of 2025)

- International Nuclear Graphite Specialists Meeting, INGSM-25 (organized in cooperation with the IAEA) (29 September – 3 October 2025)
- Workshop on High Temperature Gas Cooled Reactor Technology and Training on the High Temperature Reactor Code Package (3-7 November 2025, Technical University of Munich, Germany) – selection of participants completed
- Meeting of the Technical Working Group on Gas Cooled Reactors (1-4 December 2025)

Thank you for your attention.

For inquiries, please contact:
Small Modular Reactor Technology Development Team
IAEA Division of Nuclear Power, Nuclear Power Technology Development Section
E-mail: SMR@iaea.org, a.constantin@iaea.org