

Overview of Experiments and Code Validation Efforts for High Temperature Gas-cooled Reactors Development within International Organizations

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Introduction

High Temperature Gas-Cooled Reactors (HTGRs) have had a long and varied history which dates back to the very early days of the development of nuclear energy in the 40s. The design of the HTGR was developed to fulfil the two basic needs of the energy market consumers: their electricity and process heat demand.

In 2001, Generation IV international forum (GIF) endorsed 6 nuclear systems among which the Very High Temperature Reactor (VHTR) that can be licensed, constructed and operated by 2030 and which will deliver affordable energy products while satisfactorily addressing the issues of nuclear safety, waste and proliferation. It recognized the VHTR exceptionally suitable not only for electricity generation but also for hydrogen production and industrial heat applications.

HTGR technology is being actively researched and developed in various countries. HTGR is attracting interest from newcomer countries, such as Indonesia, Jordan and Poland. Projects are underway to design and construct demonstration reactors and advance the commercial deployment. Following a successful 168-hour demonstration operation, the HTR-PM in Shidao Bay, China entered on 6 December 2023 in commercial operation, delivering 210 MWe through a single steam turbine connected to the 2 reactors. Using high temperature steam extracted from the HTR-PM steam system, a district heating project was started in March 2024. Japan is advancing its work on the High Temperature Test Reactor and in March 2024, JAEA announced it successfully completed the world's first safety demonstration test for a prismatic HTGR to demonstrate core melt does not occur even in the loss of forced cooling. In October this year, Amazon announced it will invest in X-energy developing its Xe-100 design, with the goal of deploying up to 5 GW of its small modular reactors in the USA by 2039. Dow Chemicals also announced last year their interest to have a four-unit Xe-100 facility at one of its sites in the US.

HTGR technology has made significant strides in recent years, particularly with the successful deployment of the HTR-PM in China, demonstrating its potential for safe, efficient, and high-temperature operation. However, despite these advancements, there are still some gaps and challenges to be tackled. One particular challenge is the need for advanced experimental facilities capable of simulating the unique operating conditions of HTGRs, such as high temperatures and helium coolant environments. These facilities are essential for validating the performance and safety of HTGR components and systems. Additionally, there is a critical need for the development and validation of sophisticated codes and modelling tools that can accurately predict reactor behaviour under various operating scenarios. These tools have to address complex phenomena such as neutron transport, thermal-hydraulics, and fuel behaviour. Ensuring the reliability and accuracy of these models is crucial for the safe and efficient design and operation of HTGRs. In addition, the integration of these tools into regulatory frameworks and their acceptance by international regulatory bodies remains a significant hurdle.

To better share knowledge on these challenges and align efforts, this paper summarizes the current status of the related activities within several International Organizations, including the IAEA, GIF, the Nuclear Energy Agency (NEA) and the European Commission (EC) Research and Technological Development (RTD) Euratom Research.

Past and Current Relevant IAEA Activities

Information Sharing on Experimental Facilities and Code Validation

To reduce costs and increase schedule certainty associated with experiments and code validation needed for the design and licensing of SMRs and advanced reactors, the IAEA has established the Network for Experiment and code validation sharing (NEXSHARE) since 2024. NEXSHARE is a technical forum of global cooperation and resource sharing for experiments and code validation among design organizations, experimental facilities, international organizations, Technical Support Organizations (TSOs) and code developers. The Network provides a database of experimental facilities applicable to SMRs and advanced reactors, including HTGRs. Under the cooperation with International Organizations and Members States, this database is continuously expanded and updated. Integration with other existing IAEA experimental databases and extension to simulation codes are also planned.

The IAEA is committed to organizing regular activities that further support and strengthen NEXSHARE. The first workshop in support of NEXSHARE was jointly organized by the IAEA, the NEA and GIF. More than 70 participants from 29 countries and 3 international organizations attended this event. This workshop marked the official launch of NEXSHARE and featured presentations on topics such as experimental testing and code validation programmes, including similarities, gaps and potential for collaborations, as well as common approaches and best practices in testing and code validation.

In 2025, under the frame of NEXSHARE, four reactor technology Focus Groups (FGs) have been established, including Water Cooled Reactors, High Temperature Gas Cooled Reactors, Liquid Metal Fast Reactors and Molten Salt Reactors. The NEXSHARE FGs are short-term groups to work on experiments and code validation on a specific focus area. For the HTR FG, they plan to map the existing HTR experimental facilities and populate the database, issue working papers and organize webinars on topics of interest to the industry. All outcomes and activities will be shared through the NEXSHARE platform. A previous effort conducted by IAEA gathers a comprehensive collection of HTGR facilities, including the ones decommissioned, closed or in care and maintenance. This collection of information is available at the IAEA HTGR Technology Development Knowledge Base¹.

HTR Computer Codes for Safety Analyses

IAEA ONCORE – HTR codes

The IAEA ONCORE platform (Open-source Nuclear Codes for Reactor Analysis²) supports the development and application of open-source software for research, education, and training in nuclear science and engineering. Within this initiative, several codes relevant to HTRs, namely VSOP99, STACY and HCP, are made available and promoted for modeling and simulation tasks:

- VSOP99/11 has been used extensively for HTR pebble-type design and safety analysis. It is mainly used for core design, fuel cycle and reactor operation simulation on a long-time scale. This allows for studying safety aspects under normal operating conditions, e.g. establish a core status as input for an accident analysis. The version distributed under ONCORE represents the last release and has been updated to use new cross section libraries and has a detailed user manual and a wealth of background information over many years. It is based on well-known legacy codes such as DIFF3D, GAM, ZUT, THERMIX etc.
- The STACY is used in the frame of V/HTR safety analyses for the quantification of fission product release from the fuel both during normal operation and under the conditions of various accident scenarios. Historically, it was a standalone tool based on legacy codes like FRESCO and PANAMA. A new version, to address limitations and integrate it into the HTR Code Package (HCP) for full-core simulations, is currently being developed by Becker Technologies, Germany.
- Over several decades, Forschungszentrum Jülich and Aachen University have developed a suite of specialized computer codes to simulate various aspects of HTR cores. To preserve this expertise, address current limitations, and adapt the tools for modern computing environments, these individual programs were unified into a

¹ <https://nucleus.iaea.org/sites/connect/htgrpublic/SitePages/Home.aspx>

² <https://nucleus.iaea.org/sites/oncore/SitePages/Home.aspx>

comprehensive and consistent software suite known as the HTR Code Package (HCP). This is an integrated system that combines the physics models in a tightly coupled framework, enabling precise spatial and temporal simulations; its architecture is illustrated in Fig.1. Integrating all modules into a unified code package involves managing a vast volume of data that was previously dispersed across various legacy codes. Development of HCP is continuing at the Technical University of Munich, Germany, adding practical features such as control rod models and further integrating modules (such as STACY) and performing further verification and validation.

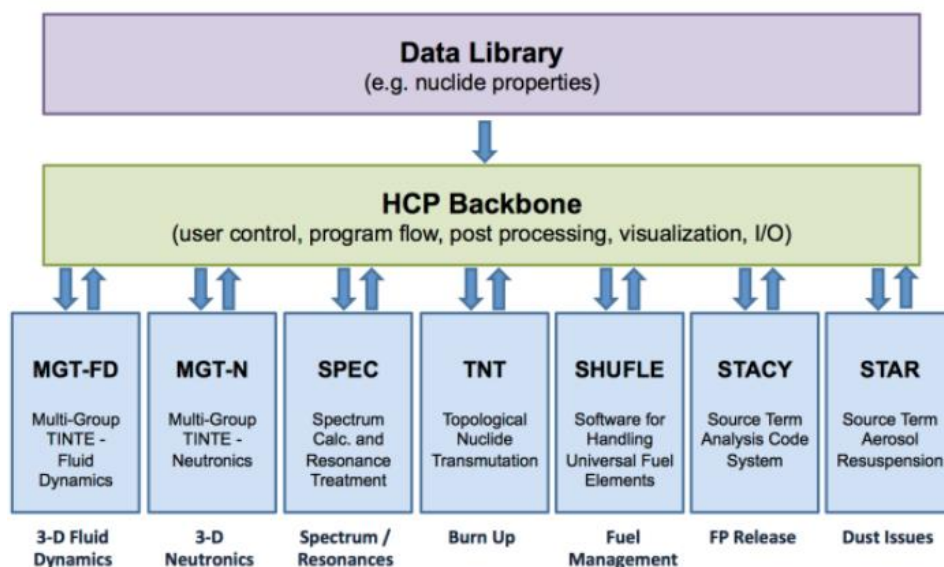


Fig. 1. Architecture of the HTR Code Package (HCP).

The versions currently available in the IAEA ONCORE can be requested at oncore.contact-point@iaea.org. IAEA has been organizing expert missions and dedicated workshops and trainings for capacity building aiming to equip the participants with the fundamental skills to perform design optimization and safety analyses of HTGRs using these codes. In 2025, an Interregional Training Course on Advances in High Temperature Gas-cooled Reactor Type Small Modular Reactors and their Analytical Tools, focus on HCP and STACY, was organized by IAEA and hosted by the Technical University of Munich, on 3-7 November. The new developments in the HCP conducted by the Technical University of Munich are showcased in a dedicated session with the occasion of the [Second Joint ICTP-IAEA Workshop on Open-Source Nuclear Codes for Reactor Analysis](#) (22-26 September 2025).

Mapping of HTR codes

The categorization of computer codes for safety analysis is considered to be helpful, therefore, the IAEA NEXSHARE HTR Focus Group aims at developing a list of HTR computer codes, including information on the scope and developer/owner. The list will cover codes for the following key areas: neutronics codes, thermohydraulic codes, multiphysics codes and code coupling methods. An initial list

was created and provided in Fig. 2. The outcome of this work is expected to be included as a deliverable in the NEXSHARE platform.

Purpose	Codes	Remark
Framework	OSCAR-5, MOOSE	
Cross Section Generation	SCALE, SERPENT, Shift, HCP Package, NJOY, TRANSX, OpenMC, XPZ	Slowing-down in graphite, heterogeneity, leakage, control rods
Criticality Calculations, Rod Worth, Reactivity feedback.	RMC, PARTISN, HCP Package, SERPENT, Shift, OpenMC.	
Equilibrium core Running-in (PBRs)	VSOP, HCP, PANGU, Pronghorn-Griffin	
Time-dependent Reactor Dynamics	PARCS-AGREE, NEM-THERMIX, Pronghorn-Griffin, NESTLE, PANGU, MGT, TINTE, GAMMA, FLOWNEX Nuclear, HCP Package, VSOP (limited), SPECTRA	Multiphysics tools for transient and steady state simulations (Load-follow, steam ingress, PLOFC/DLFOC) 3D Neutronics or point kinetics coupled with thermal hydraulic.
Computational Fluid Dynamic	ANSYS, Star-CCM+, NEK, COMSOL, HCP Package	High fidelity conjugate heat transfer using finite element analysis
Core-wide Thermal Fluidics System Analysis	RELAP5-3D, AGREE, GASNET, SAM, Pronghorn, Gothic, Flownex, ARIANT, Modelica, GAMMA, MELCOR, CATHARE, TEMPIN, HCP Package, DAYU, TALOS, SAM, SPECTRA	1-D thermal hydraulic tools for bypass flow, flow mixing calculations etc.
Thermomechanical Analysis	ANSYS, ABAQUS, COMSOL, GRIZZLY, HCP Package	2-D and 3-D solid mechanics with time history.
Seismic	ANSYS, MASTODON, Project Chrono.	
Fuel Performance	PARFUME, COPA, TIMCOAT, BISON MELCOR, STACY, HCP Package	Fuel performance data and models may indicate that one need not take credit for retention in the building
Source Term Calculations	NUIT, FRAT, FIST, MELCOR, HCP Package (STAR)	

Fig. 2. Preliminary list of HTR codes as provided by HTR Focus Group members.

Transfer of the HTR documentation archive from Forschungszentrum Jülich (FZJ) to IAEA

Since 2014, efforts have been made to ensure that critical research knowledge on HTRs at the Forschungszentrum Jülich in Germany conducted during 1962 to 2002 timeframe can be useful for the global research community on HTR. In 2024, the transfer of documentation agreed with IAEA was completed. It includes an extensive collection of documents emerged in 40 yr of research and operation of HTR technology, consisting of reports, articles, research thesis, documents from the industry. The non restricted information is available in the IAEA platform called HTGR Technology Development Knowledge Base³. Registration to the platform can be done by request, using the IAEA nucleus account.

All the files in the FZJ documentation archive are indexed in the International Nuclear Information System (INIS). INIS was established in 1970 and hosts one of the world's largest collections of published information on the peaceful uses of nuclear science and technology. The INIS repository contains bibliographic references and full-text documents of conventional and non-conventional literature, including scientific and technical reports, conference proceedings, patents and theses. The FZJ archive recorded in INIS can be accessed by searching for “Juelich Preservation Project” in the INIS website⁴.

IAEA Technical Working Group on Gas Cooled Reactors (TWG-GCR)

The TWG-GCR advises the IAEA DDG-NE on specific topics of relevance to the IAEA programmatic activities in the field, since 1978. It shares information and knowledge on national and international programmes. It 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. The cycle for 2021-2024 included representatives from 15 Member States: Canada, China, Finland, France, Germany, Indonesia, Japan, Republic of Korea, Poland, Russian Federation, South Africa, Switzerland, Ukraine, United Kingdom, United States of America and 2 observers: the European Commission and the NEA.

The strategic topics related to HTR technology suggested by the TWG-GCR members to be considered by IAEA in its activities included: (1) Develop safety standards applicable to modular HTGRs, (2) Facilitate and support R&D on HTGR for non-electric applications, (3) Disseminate modelling tools and facilitate education and training activities for HTGRs, (4) Foster information sharing and collaboration (reactor technology, fuel cycle, waste management), (5) Facilitate experiments, code development and data sharing for validation of

³ <https://nucleus.iaea.org/sites/connect/htgrpublic/SitePages/Home.aspx>

⁴ <https://inis.iaea.org/>

thermo-hydraulic, neutronics, materials and safety codes, and uncertainties assessment, (6) Establishing a systematic approach for developing the safety analysis report for HTGR, (7) Support young generation through dedicated events and programmes, (8) Identify and improve material and component classification and reduce as much as possible the nuclear grade materials for economic reasons, (9) Address challenges of transportable HTGRs, transport after operation, safeguards and security.

IAEA Activities on Fuel for HTGRs

A fundamental aspect of high-temperature reactors design is understanding coated particle fuel behaviour. Many countries have conducted extensive research on TRISO fuel, focusing on design, fabrication, characterization, irradiation performance, accident simulation testing, and predictive modelling for both normal and off-normal conditions.

In 2025 IAEA has published state of the art report IAEA-TECDOC-2090 on “Coated Particle Fuels for High Temperature Gas-Cooled, Small Modular Reactors – Progress in Design, Manufacturing, Experimentation, Modelling and Analysis Technologies”⁵.

Many countries have developed their own TRISO fuel performance analysis codes, which simulate neutron flux exposure and high-temperature accident scenarios. Refinement of these models relies on comparisons with in-pile and out-of-pile experimental data that replicate HTGR irradiation conditions. The IAEA has historically contributed to benchmarking predictive methods for TRISO fuel and fission product behaviour, starting with CRP-2 (1997), which was later expanded to cover fuel fabrication, quality assurance, irradiation performance, safety testing, and spent fuel. CRP-6 (2012) introduced benchmarking activities to compare fuel performance codes under normal operation and transients. However, significant discrepancies were observed in fission product release predictions across different codes. There remain knowledge gaps in material properties relevant to fuel performance modelling, which could pose challenges for the widespread deployment of this technology. Addressing these gaps through international collaboration and coordinated research will be essential for advancing HTR fuel technology.

In response, the IAEA has launched the CRP T12034 (2024-2029) on Fuel Modelling Exercises for Coated Particle Fuel for Advanced Reactors Including Small Modular Reactors | IAEA aimed at support interested IAEA Member States (MSs) in their efforts to enhance advanced modelling/simulation of TRISO fuel performance, improving the fundamental understanding of key TRISO fuel properties and radiation effects on the TRISO fuel properties in high-

⁵ INTERNATIONAL ATOMIC ENERGY AGENCY, Coated Particle Fuels for High Temperature Gas-Cooled, Small Modular Reactors – Progress in Design, Manufacturing, Experimentation, Modelling and Analysis Technologies, TECDOC 2090, IAEA, Vienna

radiation environment, to deploy GEN-IV reactors, SMRs and MMRs, that use TRISO-coated particle fuels.

To achieve this, the CRP T12034 focuses on the following objectives:

- To improve knowledge and fill the gaps of key TRISO fuel properties for codes development (e.g., diffusivities, creep, thermal conductivity, mechanical properties, SiC strength) with new data (e.g., obtained from new experiments and/or generated by multiscale modelling) to provide input to these codes
- To exchange information on operating and transient envelopes for TRISO-fuelled reactors among participating MSs for potential new measurements of fuel properties and new calculations
- To share a list of FPs considered critical for licensing in MSs
- To develop a database based on existing experimental data and new datasets collection to validate codes
- To perform code benchmark exercises, including uncertainty propagation.

CRP T12034 is expected to significantly enhance confidence in fuel performance codes supporting TRISO-fuelled reactors. It will provide a new database containing experimental data (both existing and newly collected), along with improved models and correlations for material properties such as thermo-mechanical properties (e.g., SiC strength) and irradiation properties (e.g., diffusion coefficients of fission products in materials like the SiC layer) for implementation into fuel performance codes.

Currently several MSs participate the CRP: China, France, South Africa, Germany, Netherlands, Ghana, Morocco, Indonesia, Bangladesh, but the CRP is still open for research proposals. Several Fuel Performance Codes participate in benchmark: SPFAC (China), COMSOL (Ghana), TRIAC (Indonesia), ATLAS (Framatome), STACY (China, South Africa, Germany), BISON (South Africa). The First RCM was held in May 2025, and second is planned for end 2026.

IAEA Activities on the Safety of HTGRs

In 2023, the IAEA published an assessment of the applicability of its safety standards to non-water-cooled reactors and small modular reactors (SMRs), in its IAEA Safety Report Series No. 123, *Applicability of IAEA Safety Standards to Non Water-Cooled Reactors and Small Modular Reactors*⁶, with the aim of identifying potential applicability issues and gaps. The publication includes considerations related to the interfaces between safety, security and safeguards. Building on this work, an IAEA technical document on ‘Considerations on the Safety of High Temperature Gas Reactors’ is currently under development. The TECDOC is intended to address the applicability issues and gaps in design safety and safety assessment

⁶ INTERNATIONAL ATOMIC ENERGY AGENCY, [Applicability of IAEA Safety Standards to Non-Water Cooled Reactors and Small Modular Reactors](#), Safety Reports Series No. 123 (2023).

standards identified in the report IAEA Safety Report Series No. 123. In particular, it focuses on the fulfilment of the fundamental safety functions, with emphasis on the concept of functional containment, the implementation of defence in depth and the notion of design extension conditions, including severe accidents. The document will also provide a list of computer codes for safety analysis. In parallel, a safety guide (DS537: Safety Demonstration of Innovative Technology in Nuclear Power Plants) and several technical documents related to design safety and safety assessment cross-cutting topics are also under development, covering both existing and advanced technologies, including HTGRs. In particular, Safety Guide DS537 addresses the use of computer codes for safety analysis.

Nuclear Energy Agency Activities

The NEA has been supporting the activities of the Generation IV International Forum since 2005. It provides technical secretariat services, archiving records and documents, as well as maintaining the Forum's internal and external websites.

In addition to this role, the NEA has been initiated a series of activities supporting HTGR development. Those are being documented by the NEA and will be added into this document.

Generation-IV International Forum Activities

The Generation IV International Forum (GIF) was created in 2001 as a collaborative international endeavor to coordinate and share research and development funded by the 13 GIF Framework signatories. As one of the six Gen-IV systems, the Very High-Temperature Reactor (VHTR)⁷ System Arrangement current covers active collaborations between research and deployment organizations in Australia (ANSTO), Canada (CNL and AECL), Euratom, France (CEA), Japan (JAEA), Korea (KAERI), Switzerland (PSI), the United Kingdom (NNL), and the United States (INL, ORNL, ANL).⁸ Within this VHTR System Arrangement, four Project Arrangements (PA) were created to manage the R&D outputs shared by these organizations: (1) Hydrogen Production, (2) Fuel and Fuel Cycle (FFC), (3) Materials, and (4) Computational Methods and Validation Benchmarks (CMVB). It is important to note that as a collaborative forum, GIF depends on the contributions from the member signatories, i.e., GIF itself does not fund or conduct any experimental or modeling activities.

The reference HTGR design considered by GIF utilizes the TRi-structural ISOtropic (TRISO) particles based fuel, helium coolant, as well as a tall and slender core layout and lower power density that allow removal of decay heat through passive heat transfer mechanisms. The FFC

⁷ HTGRs with very high outlet temperatures (VHTRs) above 1000°C was the initial focus in GIF VHTR, when process heat and specially hydrogen production was seen as a primary end-use case for these reactors. However, no VHTRs currently exist or are under development; and the more general HTGR term is preferred in the text that follows (referring to designs that limits helium outlet temperatures to 950°C or less).

⁸ <https://www.gen-4.org/>

PA focuses on TRISO fuel developments such as uranium oxycarbide kernels or ZrC coatings to improve burnup capability, reduce fission product release, and increase resistance to core heat-up accidents. As a specific example of how experimental data is utilized for fuel performance code validation within the GIF framework, a joint benchmark was developed in 2015 on “TRISO Fuel Performance Models under Accident Conditions”. This three-year benchmark effort leveraged TRISO fuel fabrication, irradiation, and safety testing experiments that were conducted by INL, JAEA and KAERI.⁹ The objectives of the benchmark were to: (1) build on the preceding IAEA Coordinated Research Project (CRP) benchmark¹⁰ in TRISO fuel performance modeling (2) model fission product release under accident conditions, (3) compare results obtained by the fuel performance modeling codes of the benchmark participants, and (4) compare these code predictions to experimental data. The safety tests selected included the first and second experiments completed at INL as part of the Advanced Gas Reactor program (AGR-1 and AGR-2) and the High Flux Reactor (HFR)-EU1bis experiment in Petten, Netherlands.

Out of the six Gen-IV nuclear system candidates, the HTGR is most focused on supporting electricity and heat cogeneration, and its high outlet temperature is ideal for hydrogen production and use in the chemical, oil, and iron industries. Within the Hydrogen Production PA, the recent focus has been on the hybrid copper-chlorine thermo-chemical and the hybrid sulfur cycles. The R&D efforts address material development, feasibility, optimization, efficiency and economic evaluation for industrial-scale hydrogen production. There are currently no code validation tasks using experimental data defined within this PA.

For structural materials utilized in HTGR designs, the VHTR Materials PA mainly focuses on high-temperature alloys, graphite and ceramic composite materials development, industry standards and codes development (e.g., what would be required to qualify a new material under the ASME¹¹ code), and develops and maintains the Gen-IV VHTR Materials “Handbook”. The Gen-IV Materials Handbook is a database hosted by INL that contains a variety of data for structural materials such as metals, graphite, ceramics, and composites. These data involve various activities including materials selection, component design, stress analysis, and code development. As of October 2024, the Handbook has collected 551 Research and Development (R&D) reports from 11 Signatories.¹² The Materials PA members are currently producing the baseline (non-irradiated) and irradiated data sets for these structural materials and sharing the results generated within their national programs, and a few examples exists of summary

⁹ P. Demkowicz, et al., “Generation IV Benchmarking of TRISO Fuel Performance Models Under Accident Conditions Final Report”, INL/EXT-20-60147, Idaho National Laboratory (2020).

¹⁰ “Advances in High Temperature Gas Cooled Reactor Fuel Technology”, IAEA-TECDOC-CD-1674, International Atomic Energy Agency (2012).

¹¹ American Society of Mechanical Engineers

¹² <https://www.gen-4.org/resources/gif-tools-and-databases/generation-iv-materials-handbook>

publications on modeling methodologies (e.g., for creep-fatigue behavior of Alloy 617)¹³, or overall modeling approaches to materials behaviour.¹⁴ However, no comparative validation benchmarks were performed up to now that are specifically intended for validation of structural material codes such as the commercial finite element analysis code COMSOL.¹⁵

The validation of new computational methods and codes in the fields of HTGR thermal hydraulics, thermal mechanics, core physics and chemical transport is crucial for ensuring the reliability and accuracy of reactor performance assessments under normal, upset and accident conditions. This validation process involves benchmark, code-to-code comparisons, and the use of diverse data sources, such as operating experience from existing HTGRs, for example the HTTR (Japan)¹⁶, HTR-10¹⁷ and HTR-PM (both China)¹⁸, and results from separate and integral effect tests at facilities such as the HTR-PROTEUS experiments (Switzerland)¹⁹, the Natural Convection Shutdown Heat Removal Test Facility (NSTF)²⁰ and the High Temperature Test Facility (HTTF)²¹ (both USA). The objective of the CMVB Project is to conduct collaborative research on and development of HTGR Computational Methods Validation and Benchmarks, and currently involves organizations from South Korea, China, Canada, the U.S., the UK, Europe, and Japan.

Examples of current CMVB activities include:

Reactor physics and source term:

- In 2021 and 2022, a series of first-critical experiments under various gas mixtures and temperatures were performed at the HTR-PM.²² This data is extremely useful to the HTGR community, since the HTR-PM reactor are the world's first modular HTGR connected to the commercial grid for power production. This data will be shared with the CMVB members as a GIF technical report.
- The data generated between 1992-1996 for the critical experiments completed at the PSI PROTEUS facility were collected as part of an IAEA CRP on "Validation of Safety

¹³ G. Douzian, et al, "Assessment of Modelling Methodologies for Prediction of High-Temperature Creep-Fatigue Behaviour of Alloy 617", International Journal of Pressure Vessels and Piping, Vol 187, 2020, <https://doi.org/10.1016/j.ijpvp.2020.104150>.

¹⁴ M. Samaras, et al., "Materials Modeling - A Key for the Design of Advanced High Temperature Reactor Components", proceedings of ICAPP 2007, France (2007).

¹⁵ Comsol website: <https://www.comsol.com/>

¹⁶ S. Saito, et al. "Design of High Temperature Engineering Test Reactor (HTTR)", JAERI 1332, Japan Atomic Energy Research Institute (1994).

¹⁷ Z. Wu, D. Lin, D. Zhong, "The design features of the HTR-10", Nucl. Eng. Des., 218 (2002).

¹⁸ Z. Zhang, et al, "The Shandong Shidao Bay 200MWe high-temperature-gas-cooled reactor pebble-bed module (HTR-PM) demonstration power plant: an engineering and technological innovation", Engineering, 2 (2016).

¹⁹ R. Brogli, et al., "LEU-HTR critical experiment program for the proteus facility in Switzerland", Energy, Volume 16 (1991).

²⁰ D. Lisowski, et al., "Water NSTF Design, Instrumentation, and Test Planning", ANL-ART-98, Argonne National laboratory (2017).

²¹ I. Gutowska, B.G. Woods and J. Halsted, "Developing PCC and DCC integral effects test experiments at the High Temperature Test Facility", Front. Energy Res. 11:1088070 (2023).

²² D. She, et al., "Prediction calculations for the first criticality of the HTR-PM using the PANGU code", NUCL SCI TECH (2021) 32:90 (2021).

Related Reactor Physics Calculations for Low-Enriched HTR's".²³ Within the CMVB work scope, a subset of these pebble-bed reactor critical facility experiments are utilized for modern code-to-code and code validation benchmarks.

- The AGR-1 TRISO particle irradiation test was the first Advanced Gas Reactor (AGR) test irradiated in the Advanced Test Reactor (ATR) at the Idaho National Laboratory. The post-irradiated measured fission product data was combined in 2018 with a description of the AGR-1 test train, capsules, compacts, and TRISO particle fuel in sufficient detail to perform a detailed high-fidelity (Monte Carlo) depletion calculation.²⁴ Since very few validation benchmarks exist for depletion of TRISO fuel, this is a valuable contribution to the reactor physics domain of HTGR simulations. Several GIF members are currently developing models for this benchmark.
- Comprehensive studies on irradiated graphite spheres of HTR-10²⁵ will provide valuable information for the source term analysis, waste minimization and radiation protection of HTGRs. The nuclide-specific data collected from irradiated HTR-10 graphite spheres (e.g., H-3, C-14, Co-60, Cs-137, Eu-152 and Eu-154) will be used to compare fission and activation product generation and transport predictions by the latest generating codes developed by the CMVB members.

Thermal fluids and oxidation kinetics:

The path of coolant flow through an HTGR tends to be complex in both space and time. The random packing of the pebble-bed core, bypass flow through reflector blocks, and the complex geometry of other system components make understanding of the flow and heat transfer in these systems difficult. Within the CMVB work scope, several data sets from integral and separate effects experiments are being used to provide validation of system and CFD code models.

- The lower plenum in an HTGR is a particularly complex subsystem designed to enhance the mixing of hot and cold streams of flow from the core and bypass regions to minimize the temperature differential of the fluid that enters the steam generator. The hot gas mixing in the lower plenum of HTR-PM is used to analyze the heat mixing performance and flow resistance property of this mixing structure using experimental data from INET.²⁶
- The helium flow distributions in typical prismatic HTGR coolant channels, as well as bypass flows in the gaps between the blocks, were experimentally assessed at the INL Matched Index of Reflection (MIR)²⁷ and Seoul National University²⁸ facilities. These data sets are being used by the CMVB members to validate detailed CFD flow predictions, as well as margin assessments for system code analysis.

²³ "Critical experiments and reactor physics calculations for low enriched high temperature gas cooled reactors", IAEA-TECDOC-1249, International Atomic Energy Agency (2001).

²⁴ J. Sterbentz, J.J. Cogliatti, "AGR-1 Depletion Benchmark", INL/EXT-18-45840, Idaho National Laboratory (2018).

²⁵ F. Xie, et al., "A comprehensive study on source terms in irradiated graphite spheres of HTR-10", Annals of Nuclear Energy, Vol. 122 (2018).

²⁶ Z. Yangping, et al., "Thermal hydraulic analysis for hot gas mixing structure of HTR-PM", Nuclear Engineering and Design, Vol. 271 (2014).

²⁷ P. Sabharwal, et al., "Matched-Index-of-Refractive Flow Facility for Fundamental and Applied Research", Proceedings of the 2014 ANS Winter Meeting, American Nuclear Society (2014).

²⁸ N.I. Tak, et al., "Validation of Numerical Methods to Calculate Bypass Flow in a Prismatic Gas-Cooled Reactor Core", Nuclear Engineering and Technology, Vol. 45, Issue 6 (2013).

- Another important heat transfer path exists in HTGRs between the reactor vessel outer surface and the reactor cavity volume. Typically, a water, air or hybrid cavity cooling system is utilized to keep the vessel and other structural components within allowable limits, especially during loss of cooling transient. The CMVB work scope includes several activities utilizing experimental data sets generated by scaled or full size Reactor Cavity Cooling Systems (RCCS), including the INET HTR-10 design²⁹, the University of Wisconsin-Madison³⁰, the Natural Convection Shutdown Heat Removal Test Facility experiment (NSTF) air-cooled³¹ and water-cooled³² experiments at Argonne National Laboratory, and the KAERI Natural Cooling Experimental Facility (NACEF).³³
- A subset of the data available from the experiments performed at the Oregon State University at the High Temperature Test Facility (HTTF) were packaged in the ongoing NEA international benchmark on “Thermal hydraulic code validation benchmark for high temperature gas-cooled reactors using HTTF data”³⁴. Since not all GIF members have access to this NEA benchmark, a similar data package will also be distributed within the GIF CMVB collaboration to enable system and CFD code validation.
- The impact of the pebble bed geometry and the way in which moving pebbles and bed density changes influence helium flow and heat distributions in a pebble core, has been studied extensively over the last two decades. Withing the CMVB work scope, data from a dedicated pebble flow dynamics test facility at INET³⁵ will be used to assess the impact of bed configurations on the flow characteristics of a pebble bed.
- An air ingress into to the core after the primary circuit depressurization due to large breaks of the pressure boundary is considered as a design-based accidents for most HTGR systems. In order to study the effects of air flow driven by natural convection as well as to investigate the corrosion of graphite, the NACOK (Naturzug im Core mit Korrosion) facility was built at Jülich Research Center in Germany, and the data from the NACOK tests will be utilized within the CMVB to validate current predictive code capabilities on coupled flow, temperature and oxidation kinetics.³⁶

²⁹ H. Zhao, et al., “Three-dimensional numerical investigation of heat transfer process in the reactor cavity of HTR-10”, Annals of Nuclear Energy, Volume 114 (2018).

³⁰ D. Lisowski, “Thermal Hydraulic Analysis of an Experimental Reactor Cavity Cooling System with Water: Performance and Stability”, PhD Dissertation, University of Wisconsin-Madison (2013).

³¹ D. Lisowski, et al., “Final Project Report on RCCS Testing with the Air-based Natural convection Shutdown heat removal Test Facility (NSTF)”, ANL-ART-47, Argonne National Laboratory (2016).

³² D. Lisowski, et al., “Experimental observations of natural circulation flow in the NSTF”, Nuclear Engineering and Design, Vol. 306 (2016).

³³ J-H Kim, et al., “The Test Results of the NACEF RCCS Test Facility”, Transactions of the Korean Nuclear Society Spring Meeting, Korea (2015).

³⁴ https://www.oecd-neo.org/jcms/pl_71708/thermal-hydraulic-code-validation-benchmark-for-high-temperature-gas-cooled-reactors-using-httf-data-htgr-t/h

³⁵ X. Jia, et al., “Experimental study of flow field characteristics on bed configurations in the pebble bed reactor”, Annals of Nuclear Energy, Volume 102 (2017).

³⁶ Y. Zheng and M.M. Stempniewicz, “Investigation of NACOK air ingress experiment using different system analysis codes panel”, Nuclear Engineering and Design, Vol. 251 (2012).

European Commission Activities

The EC-RDT-funded GEMINI+ and GEMINI 4.0 projects have made significant strides in advancing High Temperature Gas-cooled Reactor (HTGR) technology, including various activities in the areas of experimentation and code validation.

GEMINI+

GEMINI+ (2017-2021) demonstrated the feasibility of HTGRs for nuclear cogeneration (electricity, process heat) and prepared for licensing and deployment in Europe. GEMINI+ leveraged the extensive body of knowledge, experimental data, and modelling tools developed over the past 18 years through numerous European R&D initiatives focused on advancing HTGR technology such as RAPHAEL, PUMA, EUROPAIRS, ARCHER, and NC2I-R. It also drew upon the outcomes of national programs and benefited from international collaborations, including contributions from the Generation IV International Forum (GIF)³⁷. The key activities related to experiments and code validation conducted in GEMINI+ were focused on:

- Feasibility studies for realistic load-following operations in cogeneration mode³⁸
- Assessment of advanced instrumentation and structural materials for HTGRs³⁹
- Evaluation of helium turbomachinery and heat transport systems, critical for validating thermal-hydraulic codes⁴⁰
- Simulation and modelling of hybrid energy systems (HES) integrating HTGRs⁴¹.

The project deliverables available at <https://cordis.europa.eu/project/id/755478/results>

contain the detailed outcomes. Prototypes of helium-compatible heat exchangers and turbomachinery were tested under realistic thermal and flow conditions. The S-ALLEGRO loop was used to validate performance of large-scale components and support thermal-hydraulic model validation.

GEMINI4.0

The Horizon Europe GEMINI4.0 project (2022-2025)⁴² built on GEMINI+ demonstrated the poly-generation capabilities of HTGRs (e.g., heat, hydrogen, electricity) and investigated safety aspects in industrial decarbonization contexts. It included work on validation of system behaviour under various industrial load scenarios using simulation tools to support regulatory

³⁷ <https://gemini-initiative.com/geminiplus/>

³⁸ Mull, T., Herr, W., Feasibility of realistic load following with HTGR in cogeneration mode, GEMINI+ Deliverable D3.14 (2020)

³⁹ KALIVODOVÁ, J., Berka, J., Melichar, T., Test and assessment of advanced structural materials and instrumentation, GEMINI+ Deliverable 3.1 (2021)

⁴⁰ Ohashi, H., Yan, X., Mizuta, N., Review of helium turbomachinery, piping, valves and other technologies for power conversion and heat transport, GEMINI+ Deliverable D3.8 (2018)

⁴¹ GEMINI+ Deliverable D3.13 Concept of Hybrid Energy Systems for a concrete application in the Polish industry context (2019)

⁴² <https://gemini-initiative.com/gemini-4-0/>

engagement, system integration, and public communication, alongside technical validation. GEMINI 4.0 demonstrated that the GEMINI+ system can, beyond CO₂ free process heat, provide a global solution for competitive and safe decarbonisation of industrial activities. It confirms that this new form of poly-generation of various energy products has no negative effect on the safety of the combined plant.

To clear the way towards safety demonstration and subsequent deployment of this solution, the proposed project:

- Consolidate the GEMINI+ system safety demonstration and have its licensing readiness assessed by regulators and TSOs including when used in poly-generation mode,
- Develop the capability of the GEMINI+ system to operate in a cost-effective way in poly-generation mode,
- Plan for the development of a European consistent fuel cycle for this type of reactor with respect to fissile resources and to a safe and acceptable back-end,
- Launch an ambitious communication plan towards political and industry stakeholders, as well as towards the public, aimed at removing obstacles to nuclear solutions for decarbonisation of industry.

The key experiments and code validation activities conducted in GEMINI4.0 were focused on:

- Core design optimization with SERPENT and SPECTRA/MELCOR to reduce the height of the core in order to increase the performance of the reactivity control system and reduce the temperature in accident conditions.
- Assessment of hot gas mixing within layers of reflector and transition graphite blocks below the core, up to the inlet to the mixing plenum with a CFD model. Validation of the study was performed using the HTTF benchmark.
- Testing program to validate the passive safety features of a GEMINI+ demonstrator and of research-demonstrator HTGR-POLA reactor (Polish program). For HTGR-POLA, also a «standard» commissioning test program has been drafted. Instrumentation for tests has been identified together with information to assist in instrument qualification.
- Specify and perform the experiment of B₄C oxidation as well as post-experimental examinations.
- Assess the consequences of a potential accident of hydrogen release on the combined plant. A CFD study using CALIF3S evaluated a hydrogen detonation scenario.

The project deliverables reporting the studies are confidential and access is submitted to the authorization of the authors of the reports.

TREASURE

TREASURE (2025-2028) a Horizon Europe Research and Innovation Action supporting the development of the ALLEGRO demonstrator, the first experimental Gas-cooled Fast Reactor

(GFR) in Europe⁴³. The project brings together ALLEGRO⁴⁴ ⁴⁵ developers and European organizations with expertise in GFR and HTGR research to address key technical challenges. The Project is a follow-up of the SafeG Project and its main objectives are:

- Optimization of the GFR fuel design and fuel cycle. Explore feasibility of reprocessing of UOX fuel used in Light Water Reactors (LWR) for GFR. Develop strategies for GFR fuel qualification and finally propose a conceptual design of the ALLEGRO fuel system. Exploring possibilities to further reduce produced waste and to further enhance proliferation resistance in GFRs.
- ALLEGRO safety concept demonstration, including large-scale experimental verification using S-Allegro Integral Test Facility (ITF) operating in Pilsen, Czech Republic. The aim is to qualify existing Thermal Hydraulic (TH) and CFD tools, to reduce uncertainty and to increase accuracy of the computational tools that will be utilized for ALLEGRO development. A complex experiment will be carried out on S-Allegro ITF including those of DBA and DEC category.
- Finalize the basic and detailed design of Decay Heat Removal (DHR) heat exchanger (HX), manufacturing and testing innovative DHR HX using advanced manufacturing technology (3D printing).
- Optimization of operation flexibility and performance of GFRs, using intermediate heat storage system, cogeneration, and hydrogen production. Develop a secondary system design with integrated heat storage system to increase capabilities for industrial applications of GFR technology and to evaluate connected safety concerns. To evaluate potential high-efficiency hydrogen production methods tailored for GFR systems such as high-temperature electrolysis.
- Further enhancing ALLEGRO safety via use of passive and redundant safety systems. Increase nuclear safety by proposing other safety measures and systems e.g. prolonging run-down of primary system compressors.
- Optimize the core configuration in terms of minimizing the criticality of the progressively degrading core geometry in post-severe accident conditions.
- Increase accuracy of the best estimate thermalhydraulics calculations by proposing realistic heat transfer model of the ALLEGRO core.
- Attracting students and young professionals to work on GFR development by direct involvement of the students in the TREASURE Project. Organization of summer schools and seminars and exchange of information and experience with national regulators.

⁴³ <https://www.projecttreasure.eu/>

⁴⁴ Kvizda B., Bebjak S. Mayer G., Vacha P. Melichar T., Ding C., Advancing Gas-cooled Fast Reactor Technology: Outcomes of the Euratom SafeG Project on ALLEGRO Research and Development, EPJ - Nuclear Sciences & Technologies, Volume 11, 26 August 2025

⁴⁵ <https://www.safeg.eu>

The ambition of the project is to bring the ALLEGRO demonstrator concept closer to realisation by solving remaining uncertainties in the safety concept, including complex experiments and prototype testing in the integral mock-up facility S-Allegro, as well as tackling issues connected to the GFR technology, such as possibilities to use the produced high-temperature heat on an industrial scale while keeping the reactor operation safe and reliable at the same time, or issues connected with nuclear waste production minimization, multiple recycling of the nuclear fuel, and easy and fast decommissioning by design in the future.

Conclusion

Despite significant progress in HTGR research and development, critical gaps remain in experimental data, modelling fidelity, and code validation. Experimental limitations include the scarcity of full-scale prototypic tests and comprehensive irradiation campaigns for graphite and TRISO fuel under representative conditions. On the modelling side, challenges persist in accurately capturing multi-physics coupling, double heterogeneity effects, and uncertainty quantification across diverse operating scenarios. Code validation still relies heavily on code-to-code comparisons due to limited benchmark datasets. Moving forward, priorities should focus on establishing coordinated international experimental programmes, expanding high-quality benchmark data, modernising legacy codes with robust uncertainties quantification frameworks, and fostering open-source, multi-physics platforms integrated with advanced computational tools.

These efforts are essential to reduce uncertainties, accelerate licensing, and enable the safe deployment of next-generation HTGRs and require a coordinated and integrated approach among international organizations. In this context, the IAEA's NEXSHARE initiative, working in collaboration with the NEA, GIF and the EC RDT, has developed a workplan for its HTGR activities. This plan builds on past, ongoing and planned efforts described in this document, aiming to provide a more comprehensive view of available experimental facilities, data and tools that supports code validation of SMRs.

The workplan aligns with existing code validation initiatives to ensure consistent and efficient validation campaigns for SMR design and licensing, while avoiding duplication of efforts among the different Agencies. These efforts are summarized in the NHSI working paper entitled: "*NEXSHARE Efforts Supporting Experiments and Code Validation for High Temperature Gas Cooled Reactors*".

By fostering a shared understanding of validation needs and priorities, this collaborative framework can help accelerate the development of robust, reliable, and licensable SMR technologies.