Very-high-temperature reactor

The system arrangement for Gen-IV international R&D collaboration on the very-high-temperature reactor (VHTR) was signed in 2006 and extended for ten years. Several members have joined recently: Australia (2017) and the United Kingdom (2019), with Canada rejoining in 2021. The current signatories are Australia, Canada, China, Euratom, France, Japan, Korea, Switzerland, the United Kingdom and the United States. The VHTR system research plan (SRP) is intended to cover the needs of the viability and performance phases of the development plan described in the GIF R&D Outlook for *Generation IV Nuclear Energy Systems: 2018 Update* (GIF, 2018). From the six projects outlined in the SRP, three are effective and one is provisional:

- VHTR fuel and fuel cycle, with China, Euratom, France, Japan, Korea, and the United States as members;
- VHTR materials, with all signatories participating;
- VHTR hydrogen production, with China, Canada, Euratom, France, Japan, Korea, and the United States as members;
- VHTR computational methods validation and benchmarks project remains provisional, with China, Euratom, Japan, Korea and the United States expected to be members.

Main characteristics of the system

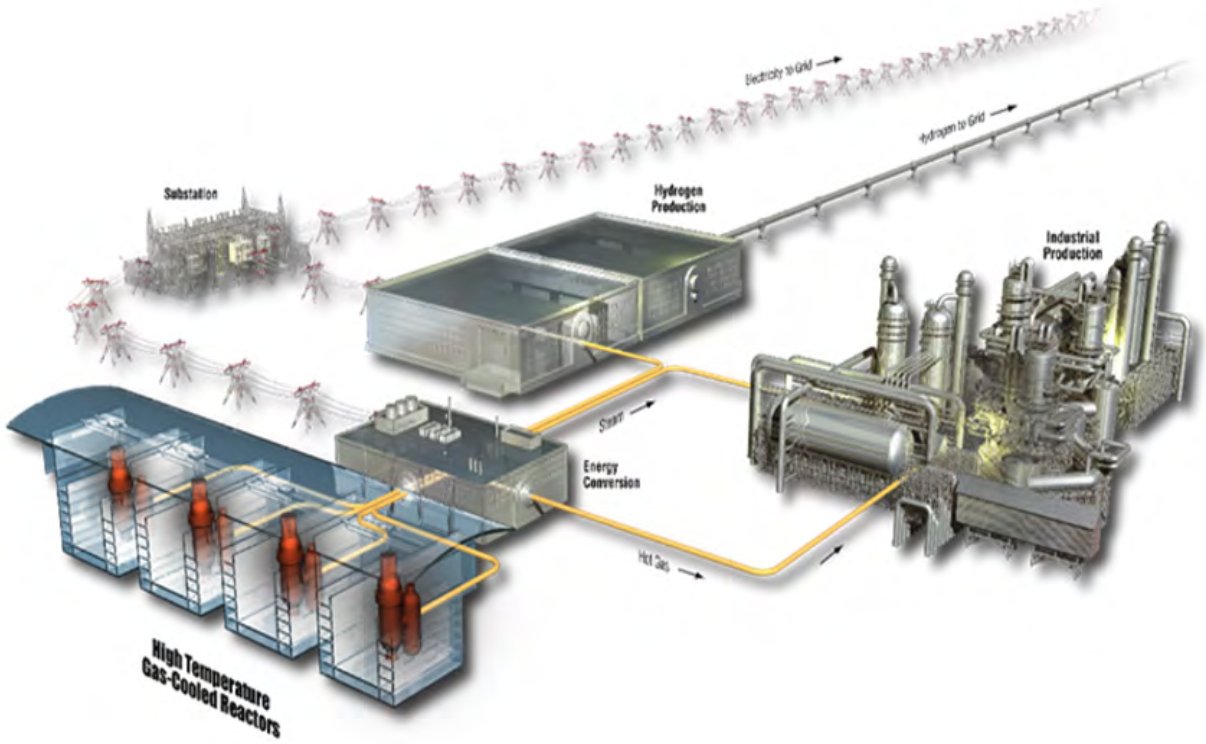
High-temperature gas-cooled reactors (HTRs or HTGRs) are helium-cooled graphite-moderated nuclear fission reactors using fully ceramic tri-structural isotropic (TRISO) coated particle-based fuels. They are characterized by inherent safety features, excellent fission product retention in the fuel, and high-temperature operation suitable for the delivery of industrial process heat, and in particular for hydrogen production. Typical coolant outlet temperatures range between 700°C and 950°C, thus enabling power conversion efficiencies of up to 48%. The VHTR is understood to be a longer-term evolution of the HTR, targeting even higher efficiency and more versatile use by further increasing the helium outlet temperature to 1 000°C or even higher. Above 1 000°C, however, such reactors will require the use of new structural materials, especially for the intermediate heat exchanger.

VHTRs can be built with power outputs that are typical of SMRs. They are primarily dedicated to the cogeneration of electricity and process heat (combined heat and power [CHP]), for example for hydrogen production. The initial driver for VHTR development in GIF was thermo-chemical hydrogen production with the sulfur-iodine cycle requiring a core outlet temperature of approximately 950°C. Further market research across GIF signatories has shown that there is also a very large near-term market for process steam of approximately 400-550°C, achievable with lower temperature HTR designs. R&D in GIF has therefore shifted to cover both lower and higher temperature versions of this reactor type.

The operational temperatures of HTRs and VHTRs can be adapted to specific end-user needs. Reactor thermal power is limited by the requirement to fully self disperse decay heat in accident conditions. The different core pressure drops, which govern the capacity for passive heat removal, translates to < 250 MWth for pebble bed reactors with single zone core and < 625 MWth for hexagonal block type reactors. The reactor power can be flexibly adapted to local requirements, for example the electricity/heat ratio of an industrial site. The power density is low and the thermal inertia of the core is high, thus granting inherent safety in accident conditions. The potential for high fuel burnup (150-200 gigawatt days per metric tonne of heavy metal [GWd/tHM]), high efficiency, high market potential, low operational and maintenance costs, as well as modular construction, all constitute advantages favoring commercial HTR deployment.

The VHTR can be designed with either a pebble bed or a prismatic block core. Despite these differences, however, all VHTR concepts show extensive commonalities, allowing for a joint R&D approach. The standard fuel is based on UO_2 TRISO coated particles (UO_2 kernel, buffer/inner pyrocarbon [iPyC]/SiC/outer pyrocarbon [oPyC] coatings) embedded in a graphite matrix, which is then formed either into pebbles (i.e. tennis ball-sized spheres) or into compacts (i.e. thumb-sized rodlets). This fuel form exhibits a demonstrated long-term temperature tolerance of 1 600°C in accident situations with sufficient safety margin. This safety performance may be further enhanced, for example through the use of a uranium-oxycarbide (UCO) fuel kernel, a ZrC coating instead of SiC, or the replacement of the graphite matrix material with SiC. The fuel cycle will first be a once-through, very high burnup, low-enriched uranium fuel cycle. Solutions to adequately manage the back end of the fuel cycle are under investigation, and potential operation with a closed fuel cycle will be prepared by specific head-end processes to enable the use of existing reprocessing techniques. Power conversion options include indirect Rankine cycles or direct or indirect Brayton cycles. Near-term concepts will be developed using existing materials, whereas more advanced concepts will require the development, qualification and coding of new materials and manufacturing methods.

High-core outlet temperatures enable high efficiencies for power conversion and hydrogen production, as well as high steam qualities (superheated or supercritical). Hydrogen production methods include high-temperature electrolysis and thermo-chemical cycles, such as the sulfur-iodine process, hybrid cycles or steam methane reforming. The transfer of heat to a user facility over several kilometers can be achieved with steam, gases, certain molten salts or liquid metals. The use of nuclear CHP with HTRs has a very large potential for the reduction of fossil fuel use and of noxious emissions, and is the prime motivation for the signatories of the VHTR system. The



Source: Gougar, H., (2014).

Figure VHTR-1. Artist's view of a 4-module VHTR poly-generation plant

increased use of nuclear energy for powering industrial processes and for large-scale bulk hydrogen is a strong motivation for VHTR development and enables the integration of nuclear power with renewable energy sources in hybrid energy systems (see Figure VHTR-1).

R&D objectives

While VHTR development is mainly driven by the achievement of very high temperatures, other important topics are driving the current R&D: demonstration of inherent safety features and high fuel performance (temperature, burnup), validation of new computational methods and code developments, coupling with process heat applications, cogeneration of heat and power, and the resolution of potential conflicts between these challenging R&D goals.

In the near term, lower-temperature demonstration projects (700°C to 950°C) are being pursued to meet the needs of current industries interested in early applications. Future operation at higher temperatures (1 000°C and above) requires the development of HT alloys, the qualification of new graphite types and the development of composite ceramic materials. Lower temperature versions of HTRs will enter the demonstration phase, based on the high-temperature gas-cooled reactor – pebble-bed module (HTR-PM) experience in China, which reached operation in 2021 and is a major HTGR milestone in 2021. Several current HTGR projects in the United States and Canada will also demonstrate small and micro pebble, as well as prismatic, reactor types in the 2025+ timeframe.

Technical highlights – fuel and fuel cycle project

Fuel and fuel cycle investigations are focusing on the performance of TRISO coated particles (the basic fuel concept for the VHTR). R&D aims to increase the understanding of standard design (UO_2 kernels with SiC/PyC coating) and examine the use of UCO kernels and ZrC coatings for enhanced burn-up capability, best fission product confinement and increased resistance to core heat-up accidents (above 1 600°C). This work involves fuel characterization, post-irradiation examination, safety testing, fission product release evaluation, as well as assessment of chemical and thermo-mechanical materials properties in representative service and accident conditions. The R&D also addresses spent-fuel treatment and disposal, including used-graphite management, as well as the deep burn of Pu and MAs in support of a closed cycle.

PIE and safety testing of fuel specimens from the advanced gas-cooled reactor-2 (AGR-2) UCO and UO_2 TRISO fuel irradiation experiment were completed in the United States in 2021. A final report has been issued on the results (Stempien et al., 2021), which represents a seven-year effort that included extensive destructive examination of 14 as-irradiated fuel compacts and particles, and 16 high-temperature post-irradiation safety tests at temperatures of 1 500°C to 1 800°C with subsequent destructive examination of the fuel specimens. Over the course of this work, thousands of individual particles were

gamma counted and examined microscopically with a variety of techniques. The combined results have provided extensive information on the performance of UO_2 and UCO TRISO fuel particles across a broad range of conditions, as well as key data on the transport of fission products.

PIE of the US AGR-5/6/7 UCO TRISO fuel irradiation experiment began in early 2021 and is expected to continue for several years. This experiment consisted of 5 separate capsules containing a total of 570 000 particles in 194 fuel compacts irradiated in the advanced test reactor for 360 effective full power days. As of December 2021, two of the capsules had been partially disassembled, and fuel specimens removed for analysis (Pham et al., 2021).

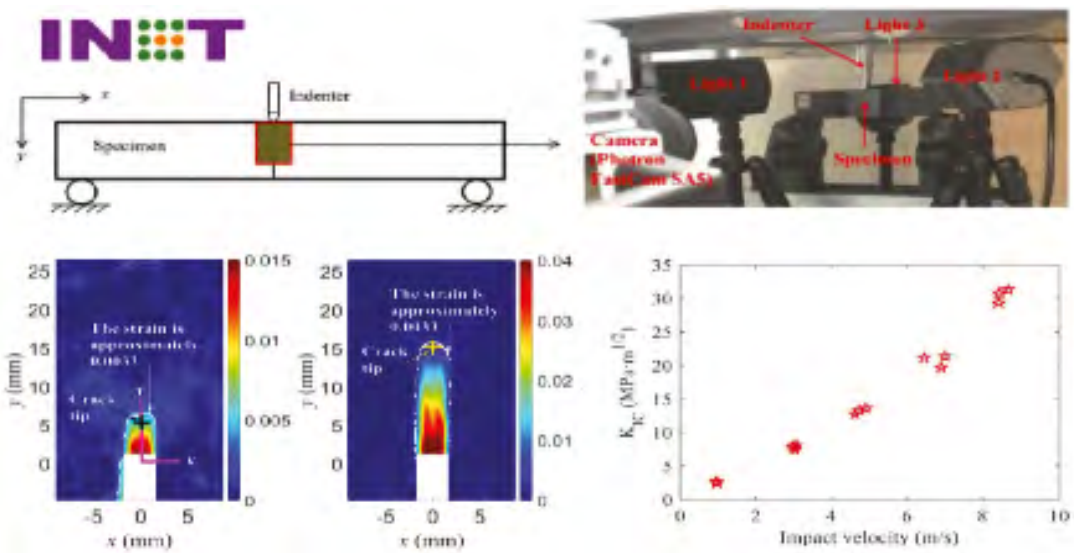
PIE of fuel pebbles discharged from the Chinese HTR-10 reactor is in progress at the newly commissioned hot cells at the Institute of Nuclear and New Energy Technology (INET). It includes burnup measurement and pebble deconsolidation. Hot testing of the new KÜFA furnace at INET, and subsequent accident testing of a fuel pebble from the high-flux reactor (HFR-EU1) irradiation experiment in the JRC Institute for Transuranium Elements, were postponed as a result of hot cell maintenance and complications from COVID-19, but these activities are tentatively planned for 2022.

Dedicated radionuclide source term experiments are an important part of TRISO fuel qualification, as the release and transport of fission products in fuel and reactor core materials impacts safety analyses. PIE of the US AGR-3/4 UCO TRISO fuel irradiation experiment is continuing. This experiment and ongoing PIE are dedicated to evaluating the in-pile transport of key fission products (including silver (Ag), cesium (Cs), and strontium (Sr)) through fuel matrix and core graphite materials. The AGR-3/4 PIE also includes post-irradiation heating tests of fuel specimens, which contain intentionally failed particles, to explore fission product transport at elevated temperatures. An important feature of these heating tests is the re-irradiation of the fuel prior to the heating tests, such that short-lived iodine 131 (a risk-dominant isotope during reactor accidents) is generated, and its release behavior can be assessed during the tests (Stempien et al., 2021).

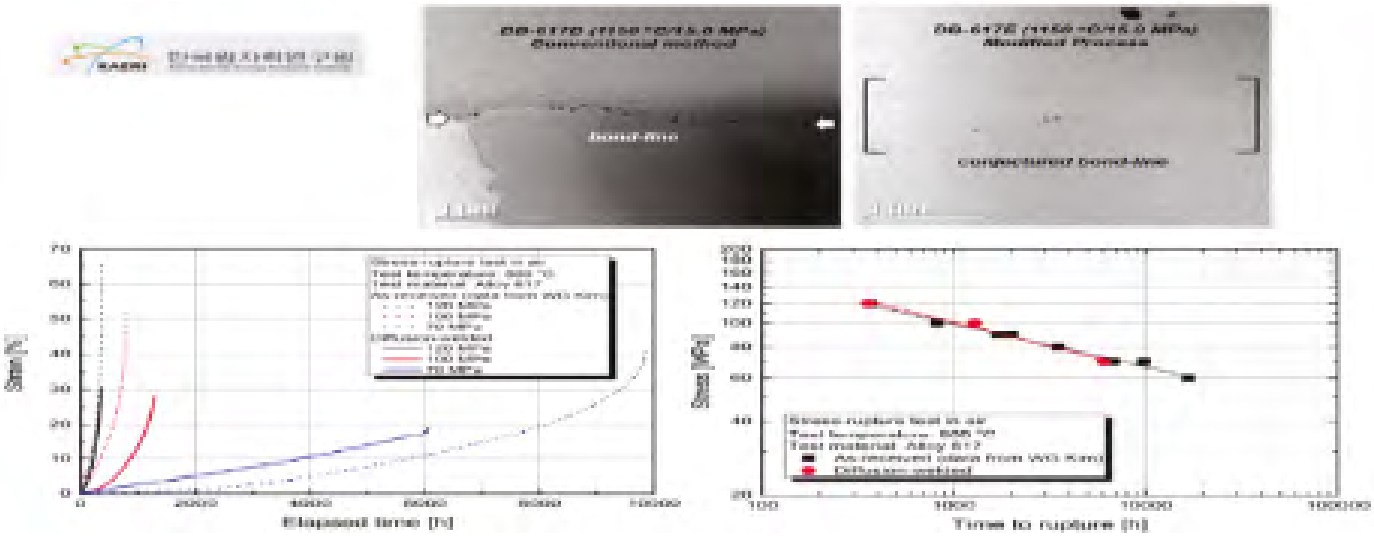
Several PMB members are pursuing the development of advanced TRISO fuel. In China, studies on UCO fuel kernel processing and fabrication of ZrC coatings are in progress. KAERI is currently developing fabrication methods for large (~800-µm-diameter) UO_2 kernels (with potential application in accident tolerant fuels based on TRISO coated particles), as well as refining fluidized bed chemical vapor deposition (FBCVD) coating methods for these larger kernels. In addition, KAERI is developing an advanced coating design incorporating a ZrC-SiC composite structure intended to extend fuel life in a VHTR; this work includes both computational studies (e.g. determining stresses in coating layers during operation) and experimental studies.

In China, approximately 860 000 fuel pebbles for HTR-PM have been fabricated as of September 2021. This includes fabrication of all first core fuel pebbles (4.2% enriched 235-U). Approximately 150 000 fuel pebbles have been loaded into the HTR-PM. Efforts are currently underway to develop a new fuel manufacturing line with larger capacity to support additional reactor modules.

Figure VHTR-2: Dynamic fracture toughness of graphite (KIC) is determined as a function of impact velocity using instrumented drop hammer testing and high-speed digital image correlation



Source: INET.



Source: KAERI.

Figure VHTR-3: Techniques developed to restrict the formation of Ti-rich carbides and/or Al-rich oxides at diffusion-bonding interfaces resulted in comparable times-to-rupture of the weldment as compared to the as-received alloy

In Japan, the restart of the High Temperature Engineering Test Reactor (HTTR) was announced in July 2021.[1] Various tests are planned with the newly restarted reactor, including fuel performance tests.

As part of Euratom activities, a new HTGR-focused research program is underway at the National Centre for Nuclear Research (NCBJ) in Poland. It includes research reactor conceptual design studies, dialogue with the nuclear regulator and pertinent materials studies, including as-fabricated TRISO fuel characterization work. There are also plans in progress for fuel irradiation in the Maria research reactor.

Technical highlights – materials project

Materials development and qualification, design codes and standards, as well as manufacturing methodologies are essential for VHTR system deployment. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in the operating environments. In 2021, research activities continued to focus on near- and medium-term project needs (i.e. graphite and high-temperature metallic alloys) with limited activities on longer term activities related to ceramics and composites.

Additional characterization and analysis of selected baseline data and its inherent scatter of candidate grades of graphite was performed by multiple members. The behavior of mechanical, physical and fracture properties were examined for numerous grades. Graphite irradiations and post-irradiation examinations and analysis continued to provide critical data on property changes, while related work on oxidation examined both short-term air and steam ingress, as well as the effects of their chronic exposure on graphite. The use of boron coatings to minimize the impact of oxidation on graphite core components was conducted. Examination and validation continued of the multi-axial loading response of graphite from dimensional changes and seismic events using large-scale experiments on graphite blocks. An example of cutting-edge research on graphite being conducted within the Project Arrangement on the dynamic testing of graphite fracture properties is illustrated in Figure VHTR-2.

Data to support graphite model development was generated in the areas of microstructural evolution, irradiation damage mechanisms and creep. Support was provided for both the American Society for Testing and Materials (ASTM) and the American Society of Mechanical Engineers (ASME) development of the codes and standards required for use of nuclear graphite, which continue to be updated and improved.

Examination of high-temperature alloys provided valuable information for their use in heat exchanger and steam generator applications. These studies included an evaluation of the existing data base and its extension through aging, creep, creep fatigue and creep crack growth rate testing to 950°C for alloys 800H and 617. Welding studies on 617,

1. For more information, see: www.jaea.go.jp/english/news/press/2021/073003/.

800H, and dissimilar welds of T22 to 800H were performed. Testing to qualify new metallic materials (e.g. alloy 709, high entropy alloys, and ODS alloys) for construction of high-temperature nuclear components was pursued. Examination of enhanced diffusion bonding techniques for construction of compact heat exchangers (CHEs) showed very promising results, and extensive modeling and testing of CHEs are laying the groundwork for their qualification in VHTRs. An example of the improvements in diffusion bonding of Ni-based superalloys needed for printed circuit CHEs is illustrated is illustrated in Figure VHTR-3.

A new thrust to develop and qualify advanced manufacturing methods for nuclear components (e.g. laser fusion, consolidation of metal powders, direct deposition) was extensively investigated by several signatories. Additionally, new approaches to synthesis of novel, high-temperature structural materials were explored. An additional task on advanced manufacturing methods has been included as a new task in the 3rd amendment to the VHTR Materials Project Arrangement.

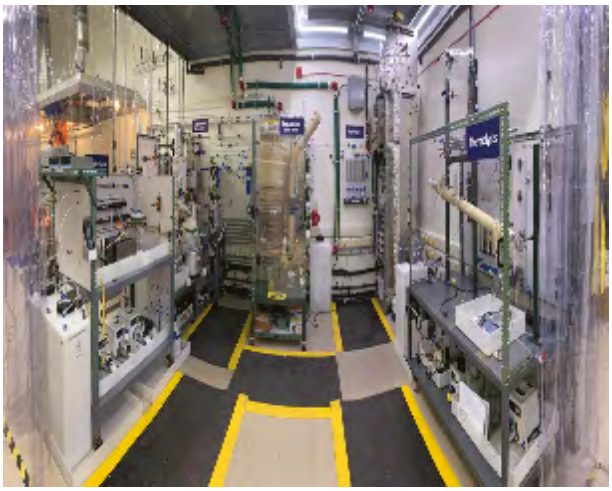
In the near/medium term, metallic alloys are considered as the main option for control rods and internals in VHTR projects, which target outlet temperatures below about 850°C. However, future projects are considering the use of ceramics and ceramic composites where radiation doses, environmental challenges, or temperatures (up to or beyond 1 000°C) will exceed capabilities of metallic materials. This is especially true for control rods, reactor internals, thermal insulation materials and fuel cladding. Work continued to examine the thermomechanical properties of SiC and SiC-SiC composites, including irradiation-creep effects, and oxidation in carbon-carbon (C-C) composites. Studies to evaluate radiation damage and examine the fracture behavior of C-C composites were begun, as were methods for direct 3D printing of SiC and SiC-SiC composites. The results of this work are being actively incorporated into developing testing standards and design codes for composite materials, and into examining irradiation effects on ceramic composites for these types of applications.

Technical highlights – hydrogen production project

For hydrogen production, two main processes for splitting water were originally considered: the sulfur/iodine thermo-chemical cycle and the high-temperature steam electrolysis (HTSE) process. Evaluation of additional cycles has resulted in focused interest on two additional cycles: 1) the hybrid copper-chlorine thermo-chemical cycle; and 2) the hybrid sulfur cycle. R&D efforts in this PMB address material development, feasibility, optimization, efficiency and economic evaluation for industrial scale hydrogen production. Performance and optimization of the processes are being assessed through integrated test loops, from laboratory scale through pilot and demonstration scale, and include component development.

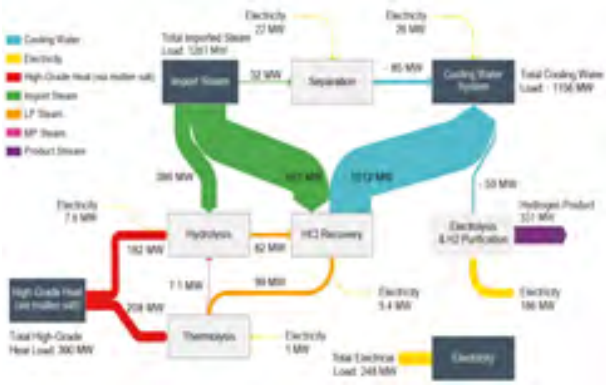
Over the last year, the Canadian effort has focused on setting up in laboratory a 50 NL/h hydrogen production system based on the copper-chlorine cycle. Figure VHTR-4 below shows the four steps of the process set up for integrated continuous operation.

Figure VHTR-4: Lab-scale 50 L/h copper-chlorine hydrogen production system for continuous operation



Source: CNL.

Figure VHTR-5: Sankey diagram of the thermal and electrical demands of a 200 tonne/d copper-chlorine cycle



Source: CNL.

Source: INET.

Figure VHTR-6: Prototype of the sulfuric acid decomposer

Source: INET.

Figure VHTR-7: Installation of the helium loop

In parallel, a major effort has been made to determine the thermal and electrical demands of a 200 tonne/d hydrogen plant based on the copper-chlorine cycle coupled to concentrated solar power or an SMR as an energy source towards deriving a levelized cost of hydrogen (LCOH). Figure VHTR 5 shows the initial results in the form of a Sankey diagram. This effort required selection of lab-scale equipment to model the complete cycle for detailed flowsheet analysis and optimization. With continuing laboratory developmental input to this analysis, the overall LCOH analysis is expected to conclude in early 2022.

Nuclear hydrogen production technologies, for the very-high temperature gas-cooled reactor in China, have been focusing on the iodine-sulfur process and hybrid sulfur process. Currently, R&D are at the stage of development of components. The main progress in the past year is as follows: 1) the prototype of the sulfuric acid decomposer was developed (see Figure VHTR-6), the leak tightness under 4 MPa and room temperature was tested with helium, and the leakage is less than 1%/48h. The lifetime test of the catalyst for the SA decomposition reaction has been conducted for more than 1 000 hours, which is considered to satisfy the demand for future pilot-scale demonstration; 2) the development of the prototype of the hydroiodic (HI) acid decomposer was completed. To test the performance of the HI decomposer, a 600°C helium loop was manufactured and erected (see Figure VHTR-6). A series of experiments were conducted to test integrity, operational performance, heat transfer features, etc. The facility will be integrated into the ongoing helium loop to carry out the HI decomposition reaction; 3) fabrication of most of the main components of the ongoing high-temperature helium loop (>900°C, 100 kW) was completed, including the helium blower, electric heater, condenser, high-temperature valve and connectors, and the installation of the helium loop is underway (Figure VHTR-7); and 4) R&D on the sulfur dioxide depolarized electrolysis (SDE) technology is continuing. The performance of the SDE cell was continuously improved, and a SDE cell with an active area of 1 863 cm^2 was developed. A multi-cell stack with hydrogen production of 1 Nm^3 has also been produced.

France has made the decision to accelerate the industrialization of the high-temperature electrolysis system to produce clean hydrogen from nuclear energy and renewable energy. Thanks to the R&D efforts of the CEA, developments have passed the first generation of cell and stack, and the manufacturing step at industrial scale is in progress with a new public-private partnership to develop a pilot line and produce high-power modules of stacks. The CEA is now developing a second generation, higher performance and durability cell by combining numerical and experimental approaches at different scales from raw material to the single cell. Through modeling and characterization of the microstructure using the European Synchrotron Radiation Facility, it has been possible to predict the performance of the cell by incorporating a mass transportation model. The CEA also performed long-term tests (over 10 000 h) and proposes a mechanism of degradation of the cell by migration of Zr. Indeed, a loss of Zr4++ from the electrolyte might be due to the formation of zirconates that could facilitate the inter-diffusion of gadolinium (Gd), reducing the local ionic conductivity and thus significantly contributing to the largest increase

in the ohmic resistance observed. Figure VHTR-8 shows the observed phenomenon. The next critical step focuses on the cell fabrication to increase reproducibility on stack and to reach a degradation rate of <1%/1 000h @ 1.3V and 0.85 A/cm^2 on a larger cell area (200 cm^2).

The feasibility studies on sulfuric acid decomposition with high-temperature heat absorbed by particles has been carried out by a consortium of EU laboratories in the framework of the European PEGASUS project.In the context of demonstrating the final design for a 2 kW sulfuric acid decomposer/sulfur trioxide (SO_3) splitting reactor/heat exchanger, allothermally heated to the necessary temperature through a high-temperature bauxite proppants stream, SO_3 splitting catalytic systems shaped in the form of particles and flow-through honeycombs and foams, intended to comprise the reactor's non-moving catalyst bed, were prepared and tested. The iron oxide-coated SiC foams in particular demonstrated reproducibly high conversion under a wide range of sulfuric acid flow rates, combined with very low-pressure drop, even under high catalyst loadings (35-45 weight %).

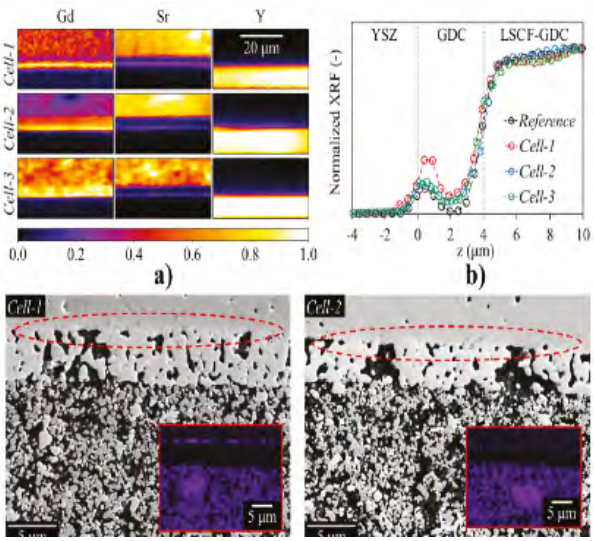
Comparative SO_3 splitting screening experiments with catalyst-coated SiSiC honeycombs, for periods between 100-950 hours onstream, demonstrated that platinum-based catalysts did not exhibit satisfactory conversion and suffered deactivation with prolonged on-stream exposure times at 650°C. This was attributed to extensive sulfation of the alumina support as determined with post-reaction analysis. These drawbacks did not justify their higher cost at this developmental stage, and thus emphasis was placed on the development of catalytic systems based on inexpensive iron oxide.

Structured catalysts involving iron oxide-coated SiSiC honeycombs and foams, as well as particles and foams made entirely of iron oxide, were long-term tested with respect to their SO_3 splitting capability. Conversions close to equilibrium could be achieved at 850°C with iron oxide-coated SiSiC foams, within a wide range of sulfuric acid flow rates. Furthermore, such foams exhibited very low-pressure drop even under high catalyst loadings.

The Korean roadmap on hydrogen released in 2019 has provided impetus to activities on hydrogen production. In this context, KAERI has conducted simulations on coupling various hydrogen production processes to a 350 MWth HTGR. Hydrogen production processes include steam methane reforming, HTSE and the sulfur-iodine process. KAERI has launched a new project related to nuclear hydrogen production, focused on the integration of HTSE and a high-temperature system, and the development of an analysis of the coupling of the reactor and HTSE system. KAERI is utilizing the helium loop facility, which has a maximum power of 600 kilowatts electric (kWe) at 950°C (see Figure VHTR-10), for the integral test of 30 kWe HTSE.

The JAEA has studied the hydrogen production technology from water that uses the thermochemical iodine-sulfur (IS) process with the aim of realizing the heat utilization of a HTGR. A test facility for producing hydrogen via thermochemical IS process was constructed using industrial materials (lining materials, metals and ceramics) with heat and corrosion-resistant components. Figure VHTR-10 shows a simplified flowsheet. The main function of three steps, namely the Bunsen reaction, sulfuric acid decomposition and hydrogen iodide decomposition, is to split water to produce hydrogen and oxygen. During the trial runs, technical problems, including leakages and pipe clogging, were clarified for a more stable operation. As a countermeasure, a shaft sealing system for a hydrogen solution pump was developed to suppress the solidification of iodine

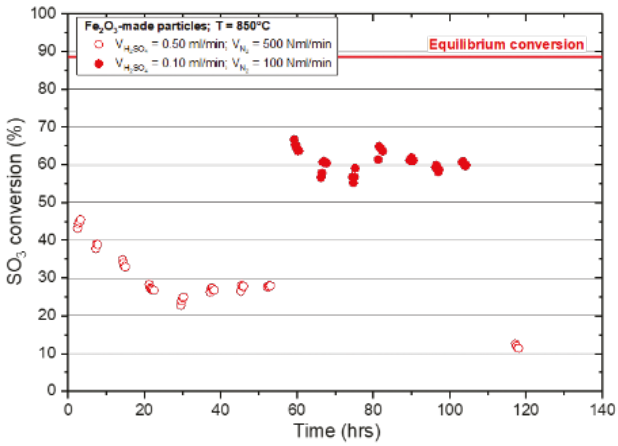
Figure VHTR-8: Analysis of operated cell



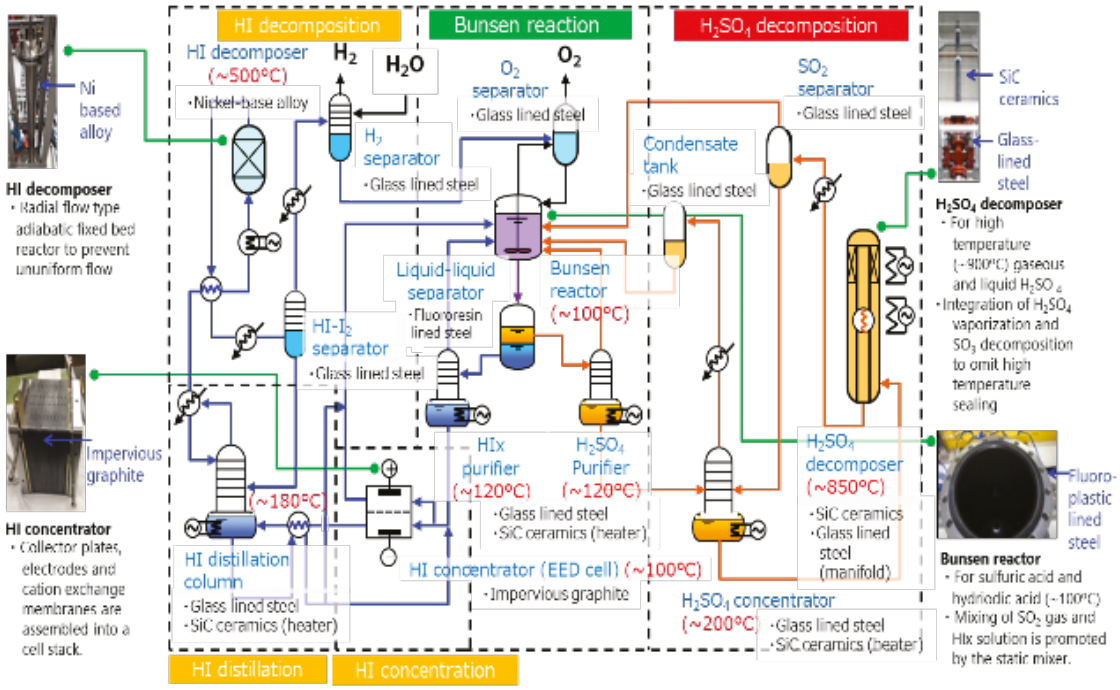
a) Normalized 2D element distribution obtained from the WRF signal at the electrode/barrier layer/electrolyte interface for cell-1, cell-2 and cell-3; b) Comparison of the integrated XRF signal of Sr between the pristine cell and the aged samples; and c) SEM images of the LSCF-GDC/GDC/8YSZ region for the cells aged in electrolysis mode at 850°C (cell-1) and at 750°C (cell-2). The inserts show the EDX signal of Sr in the same interface.

Source: Monaco, F et al. (2021).

Figure VHTR-9: Prolonged SO3 dissociation testing of iron-oxide made particles, at 850 °C: SOC conversion vs. reaction time as a function of sulfuric acid flow rate (0.10 and 0.50 mL/min)



Source: Agrafiotis, C. et al. (2021).



Source: Noguchi, H. et al. (2021).

Figure VHTR-10: Simplified flowsheet of thermochemical iodine-sulfur process, brief of major chemical reactors, structural materials of each component

and enable the stable delivery of the hydrogen solution. In addition, the quality-improved glass-lined thermometer sheath was manufactured and used to prevent the leakage of the hydrogen solution. By incorporating these developed technologies into the facility and solving technical issues (e.g. leaks and pipe clogging), the JAEA succeeded in continuously producing hydrogen for 150 h at the rate of 30L/h. Using the hydrogen production tests to evaluate the entire process, the manufacturability and function of reactors made of industrial materials were confirmed, and the prospect of a practical use of industrial material components was also confirmed.

The United States' efforts have continued on the development and application of HTSE for hydrogen production in the context of being part of a dynamic and flexible integrated nuclear energy system. Advanced reactors and renewable energy sources will provide heat and electricity for this integrated system, supporting the production of hydrogen and transport fuel, the electric grid, industrial needs, clean water production and new chemical processes. Advanced nuclear reactor systems under consideration would range from microreactors (1 to 20 MW) and SMRs (20 to 300 MW) to full-sized reactors (300 to 1 000 MW).

The objectives with respect to HTSE have been to verify operation of the solid oxide electrolysis cell stacks from US suppliers, qualify them for use in nuclear hydrogen demonstrations and benchmark stack performance in a laboratory environment for industrial applications. In this effort, a 25 kWe HTSE test facility was commissioned with some 1 000 initial tests on a 5 kWe stack. Remote supervisory control of stack operation, including multiple voltage-current sweeps, has been conducted. There is also a plan being developed for the demonstration of a 250 kWe integrated HTSE system.

Technical highlights – computational methods validation, and benchmarks project

Validation of new computational methods and codes in the areas of thermal hydraulics, thermal-mechanics, core physics and chemical transport are needed for the design and licensing assessment of reactor performance in normal, upset and accident conditions. Code validation needs will be carried out through benchmark tests and code-to-code comparisons, from basic phenomena to integrated experiments, supported by current HTTR, HTR-10 and HTR-PM test data or historic HTR data (AVR, the thorium high-temperature reactor and Fort Saint-Vrain). Computational methods will also facilitate the elimination of unnecessary design conservatisms and improve cost estimates and safety margins.

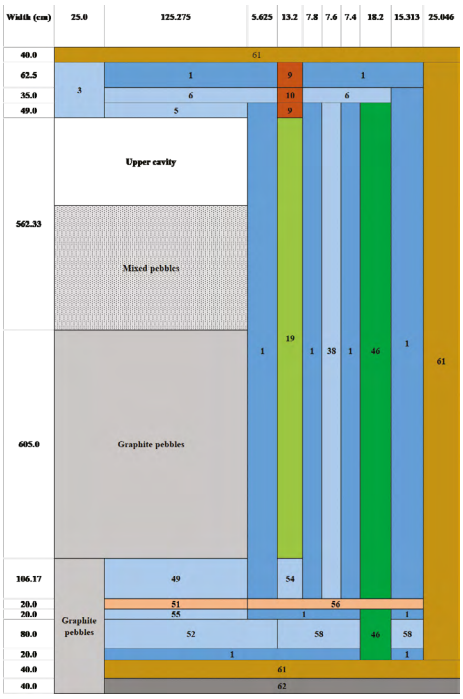
In China, the HTR-PM demonstration project is in its commissioning stage with nuclear fuel. The operating license was issued by the National Nuclear Safety Administration on 20 August 2021, and the first fuel loading began one day later. Following this, the first module and the second module of the

reactor reached first criticality on 12 September and 1 November, respectively. On 20 December 2021, the HTR-PM was connected to the electricity grid. The design of the HTR-PM600, a 600 MWe commercial plant with six nuclear steam supply system modules proved by the HTR-PM, was continued and expanded, and the basic design was completed in 2021. On such a basis, the preliminary safety analysis report has been prepared and is ready to be submitted to the regulatory body for safety review. At the same time, the detailed design is planned in 2022. At INET, the self-reliant HTR design software package, covering the fields of reactor physics, thermal hydraulics and source term analysis, is under development and assessment. Comprehensive verification and validation continued to be carried out in 2021 for the in-house version of the domestic codes, using the test data or benchmark cases defined, based on the HTR-10, HTR-PM, AVR, Proteus, ASTRA, etc. The simulation results of HTR-10 operation history with PANGU code agreed well with the experiment data. In the HTR-PM criticality experiment, PANGU predicted the critical result with about -0.6% difference in keff (see Figure VHTR-11), agreeing very well with the experimental data. Domestic codes should be used in the design verification of HTR-PM600 as the first step of the application after obtaining licenses from the safety authority in 2022.

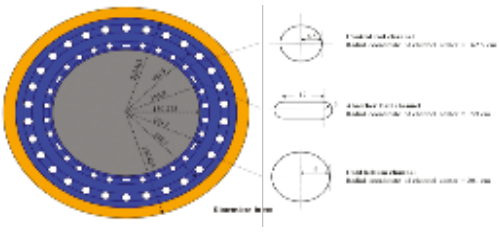
In Japan, the JAEA restarted the operation of the HTTR, a 30 MW research reactor on 30 July 2021, representing another major milestone for international HTGR activities in 2021. Safety demonstration tests under the framework of a OECD/NEA project will be carried out in 2022. There are also plans to perform various tests concerning safety, core physics, thermal-fluid characteristics, fuel performance, etc. by using the HTTR. The JAEA's R&D in code and calculation methodology developments are expected to contribute to computational methods, validation and benchmarking (CMVB) activities, such as a benchmark activity using the US advanced test reactor TRISO irradiation data.

The VHTR R&D program in Korea aims at improving very high-temperature system (VHTS) key technologies in terms of the design code development and assessment. A sub-project focuses on the development of coupled analysis technologies between the VHTS and the HTSE hydrogen production system. Some specific CMVB related R&D activities include the use of cross sections based on triangular node in CAPP code and the simulation of design-basis accidents for a 350 MWt VHTR by using the neutronics and system code coupled system (CAPP/GAMMA+). The Monte Carlo-based code, McCARD, has been updated to calculate tritium inventory more precisely with the aim of developing system performance evaluation and verifying the materials performance for the VHTR. Some specific CMVB related R&D activities include scale-down standard fuel block tests to validate the CORONA code, cross-section generation based on triangular node in DeCART2D code and simulation of the total control rod withdrawal transient for PBMR400 benchmark problems by using the neutronics and system code coupled system. In addition, fission product release from fuel to coolant under core heat-up and air-ingress accident conditions are investigated.

Figure VHTR-11: HTR-PM first-criticality calculation model by PANGU code



ID	Description	Composition (n/barn/cm)	
		C12	Boron
1	Standard reflector material (IG110, 1.781 g/cm^3, 0.445 ppm EBC)	8.92947E-02	4.41429E-08
3	Top reflector with charge tube	0.91 * IG110	
5	Top reflector with cold helium channel	0.84 * IG110	
6	Top reflector with cold helium chamber	0.6286 * IG110	
19	Side reflector with control rod channel	0.719 * IG110	
38	Side reflector with gap	0.99 * IG110	
46	Side reflector with cold helium channel	0.6679 * IG110	
49	Bottom reflector with hot helium channel	0.7135 * IG110	
52	Bottom reflector with hot helium chamber	0.5079 * IG110	
54	Bottom reflector with control rod channel	0.8315 * IG110	
55	Bottom reflector with hot helium channel	0.8741 * IG110	
58	Bottom reflector with hot helium guide tube	0.9317 * IG110	
9	Top reflector with control rod structure	5.62021E-02	3.69779E-06
10	Top reflector with control rod structure	2.30380E-02	3.68139E-06
51	Bottom reflector with B_4C	7.49075E-02	1.26898E-04
56	Bottom reflector with B_4C	8.14783E-02	1.22899E-04
61	Borated carbon bricks	8.39217E-02	3.79675E-03
62	Non-borated carbon bricks	8.53015E-02	3.79190E-05



Source: She, D. et al. (2021).

In the United States, the Idaho National Laboratory (INL) and Argonne National Laboratory (ANL) started several collaborations with HTGR vendors through the Advanced Reactor Demonstration Program (ARDP) to support the deployment of their technology to the market. Several collaborations with US universities are ongoing through the Nuclear Energy University Program to produce validation data in the fields of pebble random distribution reconstruction, natural circulation in the reactor cavity and gas stratification. The ANL continued development of a model of the High Temperature Test Facility and a system-level model of the PBMR400 benchmark with the system analysis module (SAM) code (Hu, et al., 2020). In the high-fidelity field, the ANL successfully validated NEK5000 against the experimental data produced by the 1/16th scale VHTR upper plenum facility for single and multiple jet and used machine learning to correctly predict deteriorated turbulent heat transfer for up-flow In a circular tube. At the ANL, the conversion of the Natural Convection Shutdown Heat Removal Test Facility to water loops is complete, and validation of STARCCM+ and RELAP5-3D models is ongoing (Lv, et al., 2021). The ANL and INL successfully validated the new set of equations of the intermediate fidelity porous media modeling tool "Pronghorn" for natural circulation in the upper plenum of a pebble-bed reactor using the SANA cases with 1/3 top cavity (Novak, et al., 2021). A new multiphysics equilibrium core calculator has been developed within the INL/ANL Griffin code coupled with Pronghorn. The code has been tested with an HTR-PM model, based on the available open literature, and is showing promising results once compared with the reference solutions (Ortensi, et al., 2021). The INL successfully modeled the Japanese HTTR steady state and transient scenarios (9MW and 30 MW LOFC) using applications based on a multi-physics object oriented simulation environment, developed by DOE Nuclear Energy Advanced Modeling and Simulation Program (NEAMS). The Griffin code was utilized to model 3D neutron transport using ten-group diffusion, Relap7 to represent the helium channels in the blocks and BISON to simulate a representative fuel stack per block and the full core conduction radiation from the core center to the reactor cavity cooling system. The preliminary results show relatively good agreement with the measured data for the 9 MW LOFC transients, while the 30 MW LOFC prediction showed typical behavior for that level of power and boundary conditions (Laboure, et al., 2021).

References

Agrafiotis, C. et al. (2021), "Materials for concentrated solar energy-driven sulphur-based thermochemical cycles", virtual MRS Meeting & Exhibit held on 17-23 April 2021.

GIF (2018), GIF R&D Outlook for Generation IV Nuclear Energy Systems: 2018 Update, GIF, Paris.

Gougar, H., (2014). "NGNP Program 2013 Status and Path Forward', INL/EXT-14-31035, Idaho National Laboratory, available at: https://inldigitallibrary.inl.gov/sites/sti/sti/5998116.pdf.

Hu, R, et al. (2020), "FY20 SAM Code Developments and Validations for Transient Safety Analysis of Advanced non-LWRs", ANL/NSE-2-/50, Argonne National Laboratory.

Laboure, V., et al. (2021), "FY21 Status Report on the ART-GCR CMVB and CNWG International Collaborations", INL/EXT-21-64241, Idaho National Laboratory.

Lv, Q., et al. (2021), "FY21 Progress on Computational Modeling of WaterBased NSTF", ANL-ART-235, Argonne National Laboratory.

Monaco, F et al. (2021), "Oxygen electrode degradation in solid oxide cells operating in electrolysis and fuel cell modes: LSCF destabilization and interdiffusion at the electrode/electrolyte interface", *Int. J. of Hydrogen Energy*, Vol. 42, Issue 62, p31533-31549.

Noguchi H. et al. (2021), "Hydrogen Production using thermochemical water-splitting Iodine-Sulfur process test facility made of industrial materials: Engineering solutions to prevent iodine precipitation", *Int. Journal Hydrogen Energy*, Vol. 46, 22328-43.

Novak, A. J., et al. (2021), "Pronghorn: A Multidimensional Coarse-Mesh Application for Advanced Reactor Thermal Hydraulics", *Nuclear Technology*, 207:7, 1015-1046, DOI: 10.1080/00295450.2020.1825307.

Ortensi, J., P. Balestra (2021), "Initial study on cross section generation requirements for a PBR equilibrium core", INL/EXT-21-64242, Idaho National Laboratory.

Pham, B.T., et al. (2021), "AGR-5/6/7 Irradiation Test Final As-run Report," INL/EXT-21-64221, available at: www.osti.gov/biblio/1822447-agr-triso-fuel-post-irradiation-examination-final-report.

She, D., et al. (2021), Prediction calculations for the first criticality of the HTR-PM using the PANGU code, *Nucl. Sci. Tech.*, 32:90.

Stempien, J. D., et al. (2021), "AGR-2 TRISO Fuel Post-Irradiation Examination Final Report," INL/EXT-21-64279, available at: www.osti.gov/biblio/1822447-agr-triso-fuel-post-irradiation-examination-final-report.

Stempien, J. D., et al. (2021), "Reirradiation and heating testing of AGR-3/4 TRISO fuels," Paper HTR 2021-3004, Proceedings of HTR 2021, virtual event held on 2-4 June 2021.

Gerhard Strydom
Chair of the VHTR SSC,
with contributions from
VHTR members