Very-high-temperature reactor

High-temperature gas-cooled reactors (HTRs or HTGRs) are helium-cooled graphite-moderated nuclear fission reactors using fully ceramic 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 750°C and 850°C, thus enabling power conversion efficiencies up to 48%. The very-high-temperature reactor (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 950°C or even higher. Above 950°C, however, and such reactors will require the use of new structural materials.

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 sulphur-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 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.

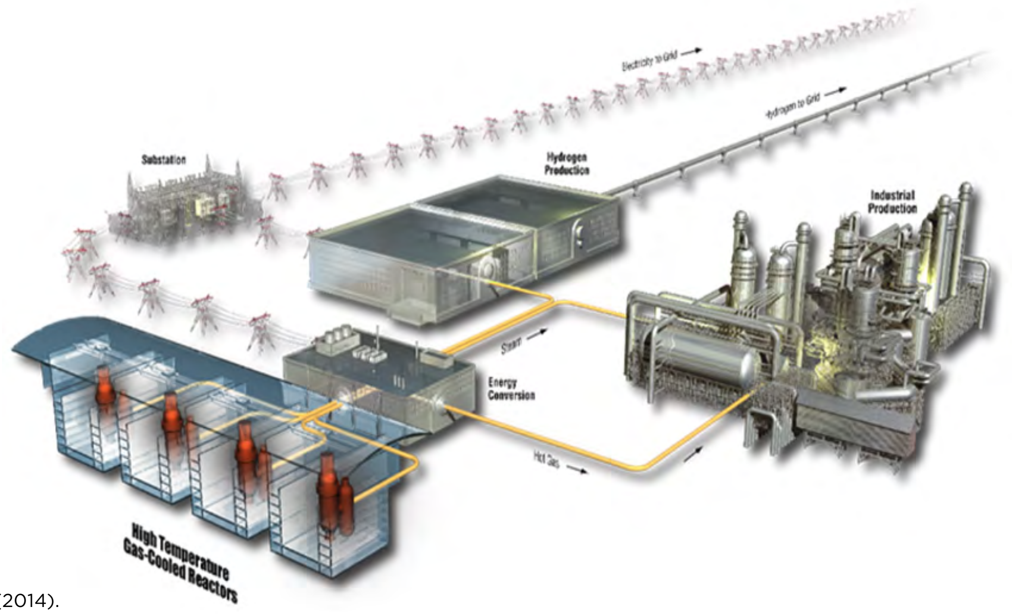
Cogeneration of heat and power makes HTRs and VHTRs attractive heat sources for big industrial complexes, such as chemical plants, to substitute large amounts of process heat at different temperatures, which are today produced by fossil fuels. Depending on the coolant outlet temperature, such reactors can be employed to produce hydrogen from heat and water by using thermo-chemical, electro-chemical or hybrid processes with largely reduced CO_2 emissions. Typical HTR coolant outlet temperatures range from below 750 to 850°C, thus enabling power conversion efficiencies up to 48% in pure power generation and even much higher in the combined heat and power (CHP) mode.

HTRs and VHTRs can be operated with a once-through LEU (<20% 235U) fuel cycle and with a closed fuel cycle (improved sustainability). This reactor type was identified quite early as particularly suitable for the Th-U fuel cycle, while potential symbiotic fuel cycles with other reactor types (especially light water and fast reactors) are also an option.

The operational temperatures of HTRs and VHTRs can be adapted to specific end-user needs. Thermal reactor power is limited by the requirement for fully passive heat removal in accident conditions. The different core pressure drops, which govern the capacity for passive heat removal, translates to <250 MWth for pebble bed reactors and <625 MWth for hexagonal block type reactors. The actual 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 walk-away safety in accident conditions. The potential for high fuel burn-up (150-200 GWd/tHM), high efficiency, high market potential, low operational and maintenance costs, as well as modular construction, all constitute advantages favouring commercial deployment.

The basic technology has been established in former high-temperature gas-cooled reactor plants, starting with the NEA DRAGON project, which led to the development of coated-particle fuel and demonstrated the safety features of HTRs, including a final core heat-up experiment. Later, the United States Peach Bottom and Fort Saint-Vrain plants were built, as well as the German AVR and THTR prototypes, which produced high-quality steam up to 550°C. After resolving some initial problems, the technology has advanced through near- and medium-term projects led by several plant vendors and national laboratories, such as HTR-PM (China), PBMR (South Africa), the gas turbine high-temperature reactor 300 for cogeneration (GTHTR-300C, Japan), Antares project (France), the Nuclear Hydrogen Production Project (NHDD, Korea), the gas turbine modular helium reactor (GT-MHR, US and Russia) and the next generation nuclear plant (NGNP, US). Experimental reactors such as the HTTR (Japan, 30 MWth) and HTR-10 (China, 10 MWth) support technology development, including CHP, hydrogen production and other nuclear heat applications.

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 tri-structural isotropic (TRISO) coated particles (UO_2 kernel, buffer/iPyC/SiC/oPyC coatings) embedded in a graphite matrix, which is then formed either into pebbles (tennis ball size spheres) or into compacts (thumb-size rodlets). This fuel form exhibits a demonstrated long-term temperature tolerance of 1 600°C in accident situations. This safety performance may be further enhanced, for example through the use of a uranium-carbon-oxygen 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 burn-up, 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



Gougar, H., (2014).

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

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 sulphur-iodine process, hybrid cycles or steam methane reforming. The transfer of heat to a user facility over a distance of several kilometres 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 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).

Status of cooperation

The VHTR system arrangement was signed in November 2006 by Canada, Euratom, France, Japan, Korea, Switzerland and the United States. In October 2008, China formally signed the VHTR system arrangement. South Africa formally acceded to the GIF framework agreement in 2008, but announced in December 2011 that it no longer intended to accede to the VHTR SA. Canada withdrew from the SA at the end of 2012 but is again an observer and remained active in the hydrogen production project. The SA was subsequently signed by Australia (December 2017) and the United Kingdom (January 2019). In 2020, the VHTR System Steering Committee updated its work plan of the high-level system R&D for the development of the VHTR in support of national or international VHTR demonstrator projects and enhanced performance capability in the long term.

The fuel and fuel cycle project arrangement became effective on 30 January 2008, with implementing agents from Euratom, France, Japan, Korea and the United States. The project arrangement (PA) has been extended to include input from China and was amended in 2013. The project was extended in 2018 for a period of ten years.

Although the term of the original VHTR materials project plan was completed in 2012, the materials PA continued through 2019 under its first amendment, which added China as a signatory. On 27 April 2020, the second amendment of the PA became effective. It incorporated a new project plan for technical activities and planned contributions from 2018-2022, and added Australia as an additional signatory. It also extended the term of the PA through April 2030. Contributions to the new PP for 2018-2022 were developed by all previous signatories (China, the European Union, France, Japan, Korea, Switzerland, and United States), as well as Australia, the newest member. In 2020, the United Kingdom expressed an interest in joining the PA, and Canada also expressed an interest in joining the PA once again, after its earlier withdrawal. Both countries presented their capabilities and potential contributions for VHTR materials to the Materials Project Management Board (PMB), which invited both of these countries to prepare formal planned contributions to the PP. These planned contributions are expected to be completed in 2021. If accepted, Canada and the United Kingdom will be invited to join the third amendment to the PA. It is also anticipated that the current PP will be extended through 2024, with augmented contributions from all existing signatories.

The hydrogen production PA became effective on 19 March 2008, with implementing agents from Canada, Euratom, France, Japan, Korea and the United States. In 2020, the forthcoming five-year project plan was prepared to incorporate contributions from China and updated contributions from other countries, under the consensus of the PMB. The amendment of the hydrogen production PA to welcome China's Institute of Nuclear and New Energy Technology (INET) as a member of the PMB is expected in 2021.

The computational methods validation and benchmarks (CMVB) PA remained provisional in 2020. After the draft PP was approved by the VHTR System Steering Committee (SSC), the draft PA was confirmed by each signatory. The signatories are now ready to pursue the signature of the PA expected in 2021.

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, burn-up), coupling with process heat applications, cogeneration of heat and power, and the resolution of potential conflicts between these challenging R&D goals.

The VHTR system research plan 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*. From the six projects outlined in the SRP, three are effective and one is provisional, as discussed below. Today, most of the performed activities are licensing-relevant:

- Fuel and fuel cycle (FFC) 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 uranium-oxycarbide 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 plutonium and minor actinides (MAs) in support of a closed cycle.
- Materials (MAT) development and qualification, design codes and standards, as well as manufacturing methodologies, are essential for VHTR system development. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in operating environments. For core coolant outlet temperatures up to 950°C, it is envisaged to use existing materials; however, the stretch goal of 1 000°C, including safe operation under off-normal conditions and involving corrosive process fluids, requires the development and qualification of new materials. Multi-scale modelling is needed to support improved design methods. In addition to high-temperature heat exchangers, additional attention is being paid to metal performance in steam generators, which reflects the current interest in steam-based process applications at somewhat lower core outlet temperature of 750°C to 850°C. Structural materials are considered in three categories: graphite for core structures, or for fuel matrix; very/medium-high-temperature metals; and ceramics and composites. A materials handbook has been developed and is being used to store and manage VHTR data, facilitate international R&D co-ordination, and support modelling to predict damage and lifetime assessment.
- For hydrogen production, two main processes for splitting water were originally considered: the sulphur/iodine thermo-chemical cycle and the high-temperature steam electrolysis process. Evaluation of additional cycles has resulted in focused interest on two additional cycles with lower temperature: the hybrid copper-chlorine thermo-chemical cycle and the hybrid sulphur cycle. R&D efforts in this PMB address feasibility, optimization, efficiency and economics evaluation for small- and large-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 such as advanced process heat exchangers. Hydrogen process coupling technology with the nuclear reactor is also being investigated, and design-associated risk analysis is being performed, covering potential interactions between nuclear and non-nuclear systems. Thermo-chemical or hybrid cycles are examined in terms of technical and economic feasibility in dedicated or cogeneration hydrogen production modes, aiming to lower operating temperature requirements in view of making them compatible with other Gen-IV nuclear reactor systems.
- CMVB in the areas of thermal-hydraulics, thermal-mechanics, core physics and chemical transport, are major activities. They are needed for the assessment of reactor performance in normal, upset and accident conditions and for licensing. Code validation needs to be carried out through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported by HTTR, HTR-10 and

HTR-PM tests or by past HT reactor data (AVR, THTR and Fort Saint-Vrain). Computational methods will also facilitate the elimination of unnecessary design conservatisms and improve construction cost estimates.

Even though they may not be implemented, the development of components needs to be addressed for key reactor systems (core structures, absorber rods, core barrel, pressure vessel, etc.) and for the energy conversion or coupling processes (e.g. steam generators, heat exchangers, hot ducts, valves, instrumentation and turbo machinery). Some components will require advances in manufacturing and on-site construction techniques, including new welding and post-weld heat treatment techniques. Such components will also need to be tested in dedicated large-scale helium test loops simulating normal and off-normal events. The project on components should address development needs that are partly common to the GFR, so that common R&D can be envisaged for specific requirements.

SIA is necessary to guide R&D so as to meet the needs of different VHTR baseline concepts and new applications (cogeneration and hydrogen production). Near and medium-term projects should provide information on their designs to identify the potential for further technology and economic improvements. This topic is directly addressed by the SSC.

Milestones

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 development of HT alloys, qualification of new graphite types and the development of composite ceramic materials. Lower-temperature versions of HTRs (from 700°C to 950°C) will enter the demonstration phase, based on HTR-PM experience in China, which is scheduled to reach the operation stage in 2021. A future higher temperature version (1 000°C and above) will require more research.

Main activities and outcomes

Fuel and fuel cycle project

The VHTR fuel and fuel cycle (FFC) project is intended to provide demonstrated solutions for VHTR fuel (design, fabrication and qualification) and for its back-end management, including novel fuel cycle options.

TRISO coated particles need to be qualified for relevant service conditions. Furthermore, its standard design (UO_2 kernels surrounded by successive layers of porous graphite, dense pyrocarbon (PyC), silicon carbide (SiC) and then PyC) could evolve through the use of a uranium oxycarbide (UCO) kernels or a zirconium carbide (ZrC) coating for enhanced burn-up capability, minimized fission product release, and increased resistance to core heat-up accidents. Fuel characterization work, post-irradiation examinations (PIE), safety testing and fission product release evaluations, as well as the measurement of chemical and thermo-mechanical material properties in representative conditions, will feed a fuel materials database. Further development of physical models enables assessment of in-pile fuel behavior under normal and off-normal conditions.

The back-end of the fuel cycle encompasses spent-fuel treatment and disposal, as well as used-graphite management. An optimized approach for dealing with the graphite needs to be defined. Although a once-through cycle is envisaged, the potential for the deep burn of plutonium and minor actinides in a VHTR, as well as the use of thorium-based fuels, will be accounted for as an evolution towards a closed cycle.

Irradiation and PIE

Irradiation and PIE includes activities on fuel irradiation testing, PIE facility and equipment development and post-irradiation examination of fuel specimens, with activity currently taking place in China and the United States.

PIE on the AGR-2 fuel (including both UCO and UO_2 TRISO particles) has continued in the United States and is near completion, with a final report expected in 2021. This work includes extensive destructive examination of fuel compacts and particles. Up to this time, 12 UCO and 2 UO_2 compacts have been examined, providing information on fission product retention in the particles and compacts during irradiation, and detailed microstructural information on the condition of the coating layers and migration of fission products in the layer.

The US AGR-5/6/7 irradiation of UCO TRISO fuel into the advanced test reactor was completed in July 2020 after achieving approximately 360 effective full power days in the reactor and a peak fuel burn-up of ~15% FIMA. This experiment is both the final fuel qualification irradiation and a separate HT fuel performance margin test (peak temperatures of ~1 500°C) and contains approximately 570 000 fuel particles in 194 fuel compacts. PIE is expected to begin around April 2021. Development of PIE capabilities with the newly established INET hot cells continues, after installation of an Irradiated Microsphere Gamma Analyzer (IMGA) apparatus and pebble deconsolidation equipment was completed. This equipment will be used to perform destructive examination on irradiated fuel pebbles.

Safety

A fuel pebble of HTR-10 production, which was irradiated previously in the HFR-EU1 experiment in HFR Petten, was heated in the KÜFA facility at JRC Karlsruhe to evaluate fission product release at elevated temperatures. The specimen was held at temperatures of 1 620°C, 1 700°C, and 1 800°C for 150 hours at each temperature. Release of Kr-85, Cs-134, and Cs-137 were measured. A new KÜFA furnace system similar to the one deployed in Karlsruhe has been installed in INET hot cells (see Figure VHTR-2) Hot testing of the system has been delayed by the COVID-19 pandemic, but is planned for 2021 using low-burn-up fuel pebbles discharged from HTR-10.

The United States is continuing to perform PIE on the AGR-3/4 irradiation experiment components and heating tests on AGR-3/4 TRISO fuel compacts. These compacts contain about 1 900 TRISO fuel particles, and 20 "designed-to-fail" particles that experience coating failure during the irradiation. The PIE of these materials thus helps to understand fission product transport in fuel matrix and graphite materials, and will be used to refine fission product transport models that are critical for reactor safety analyses. Work in 2020 focused on destructive examination and heating tests of the AGR-3/4 fuel compacts. Heating tests have been performed on a total of seven irradiated AGR-3/4 compacts in which the fuel is heated at temperatures between 1 100°C and 1 600°C while fission product release is monitored. In three of these tests, the fuel specimens were re-irradiated in the neutron radiograph reactor (NRAD) to generate short-lived iodine-131 (I-131) and xenon-133 (Xe-133) prior to the heating tests. These tests provide data on the release of short-lived fission products (including I-131) that can be significant contributors to off-site dose during reactor accidents. Accurately quantifying the release from the kernels requires measurements of the fission product inventory in the compact matrix. In order to avoid the exposed kernels that lie roughly along the compact axial centerline, the fuel compacts are deconsolidated by rotating the compact during the process to remove successive layers of particles (~1 mm-thick layers).

The United States also continues with the development of a dedicated furnace to heat irradiated TRISO fuel specimens up to 1 600°C in oxidizing atmospheres. The system will be used to test oxidation behavior of fuel and fuel materials in air/He and moisture/He gas mixtures, while monitoring online the release of fission products and reaction products. The system is expected to be deployed in 2021.

Post-irradiation heating tests of loose TRISO particles in helium for very long durations (up to 1 500 h) under a range of temperatures (1 150°C to 1 600°C) are being performed in the United States with the goal of quantifying the release of certain fission products through intact particle coatings. This study focuses in particular on silver (Ag) and europium (Eu), as these have been observed to be released in significant (Ag) or modest (Eu) amounts depending on the fuel irradiation temperature. The majority of the experimental matrix has been completed, with final tests to be performed in 2021.

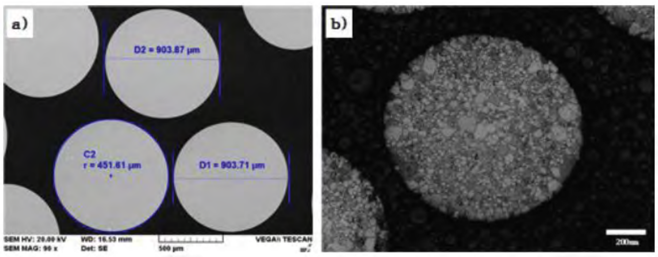
Enhanced and advanced fuel fabrication

Work to develop advanced TRISO fuels, in some cases involving varying particle designs or new coating materials, is in progress in several member countries. In Korea, KAERI has been working to develop fabrication methods for UO_2 kernels with diameters significantly larger than the conventional 500 µm used in most UO_2 TRISO particles. With some process development, kernels with diameters of ~900 µm and good sphericity and density have been fabricated (see Figure VHTR-3). These kernels are envisaged for use as accident-tolerant fuels for light water reactors. KAERI is planning to refine the particle coating process to accommodate these larger kernels. In addition, this work has involved computational studies to optimize the particle dimensions. Future work is planned on developing a composite double layer ZrC/SiC coating for TRISO, which may result in improved performance.

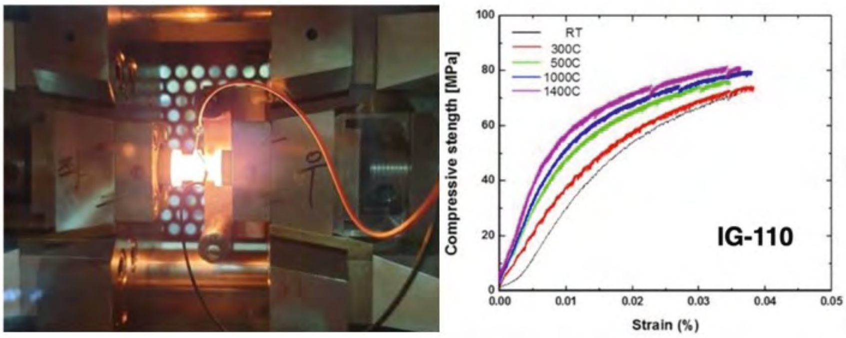
In China, work has been progressing to develop processing methods for UCO kernels, which may offer enhanced fuel performance compared to the conventional UO_2 fuel to be used in the first loading of the HTR-PM reactor.

Figure VHTR-2. KÜFA furnace installed in INET hot cells

Figure VHTR-3. Example of large-diameter UO_2 kernels fabricated at KAERI



Mean diameter(µm)	Sphericity	Aspect ratio	Density(g/㎤)
907.5	0.948	0.947	10.78



Source: Xu, A. et al. (2021).

한국원자력연구원
KAERI

Figure VHTR-4. Very high-temperature compression testing of graphite

Other activities

While there has been little VHTR fuel work taking place in the European Union in recent years, representatives from the National Centre for Nuclear Research (NCBJ) in Poland attended the September 2020 FFC PMB meeting as observers and discussed plans for TRISO fuel development in Poland. The Polish representative has been appointed as a Euratom member to the VHTR PMB. A representative from the United Kingdom also attended the FFC PMB meeting and provided a presentation on TRISO capabilities. It is expected that the United Kingdom will become a new signatory to the FFC PMB in the near future.

Materials project

As part of the new PA, a thorough review was made of all high-level deliverables (HLDs). Additionally, by the end of 2019, over 450 technical reports and over 30 000 materials test records, including contributions from all signatories, had been uploaded into the Gen-IV Materials Handbook (i.e. the database used to share materials information within the PMB). This reflects the outstanding technical output of the membership, which has now been shared to support system design and codes and standards development.

In 2020, research activities continued to focus on near- and medium-term project needs (i.e. graphite and HT metallic alloys), with limited activities related to ceramics and composites.

Additional characterization and analysis of selected baseline data, and its inherent scatter of candidate grades of graphite, were performed by multiple members. Mechanical, physical and fracture property behavior was examined for numerous grades. Graphite irradiations, PIE and other analyses 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 to graphite. Tests on the use of boron coatings to minimize the impact of oxidation on graphite core components were conducted. Examination and validation of the multi-axial loading response of graphite from dimensional changes and seismic events, using large-scale experiments on graphite blocks, continued. An example of very-high-temperature testing of graphite mechanical properties is shown in Figure VHTR-4.

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 International) and American Society of Mechanical Engineers (ASME) codes and standards required for the use of nuclear graphite, which continue to be updated and improved.

Examination of HT alloys provided valuable information for their use in a heat exchanger and steam generator. These studies included an evaluation of the existing database and its extension through ageing, creep, creep-fatigue and creep crack growth rate testing to 950°C for alloys 800H and 617. Welding studies on 617, 800H, and dissimilar welds of T22 to 800H were performed. Examination of enhanced diffusion bonding techniques for construction of compact heat exchangers (CHEs) showed promising results, and extensive modelling and testing of CHEs are laying the groundwork for their qualification in VHTRs. Testing to qualify new metallic materials (alloy 709, high entropy alloys, oxide dispersion-strengthened [ODS] alloys) for construction of high-temperature nuclear components was pursued.

A new thrust to develop and qualify advanced manufacturing methods for nuclear components (laser fusion, consolidation of metal powders, direct deposition, etc.) was extensively investigated by several signatories. Additionally, new approaches to the synthesis of novel HT structural materials were explored.

Advanced characterization techniques are being used to evaluate the impact of irradiation effects on HT structural materials. Figure VHTR-5 illustrates the novel use of nano-scale tensile specimens fabricated in situ to assess the degradation of tensile properties in ODS alloy (MA957) specimens exposed to ion-beam irradiation. The micro-scale region of maximum damage in ion-beam-irradiated samples can be evaluated using such nano-scale specimens.

For the near/medium term, metallic alloys are considered as the main option for control rods and internals in VHTRs, 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 thermo-mechanical properties of SiC and SiC-SiC composites, including irradiation-creep effects and the oxidation in carbon- carbon (C-C) composites. Studies to evaluate radiation damage and examine the fracture behavior of C-C composites have 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 to examine irradiation effects on ceramic composites.

Hydrogen production project

Canada has continued its efforts to demonstrate an integrated copper-chlorine (Cu-Cl) cycle for the production of 50 NL/h hydrogen. Continuing advancements in the four steps of the cycle has also been carried out during 2020. The equipment being assembled for integrating the whole cycle are shown in Figure VHTR-6.

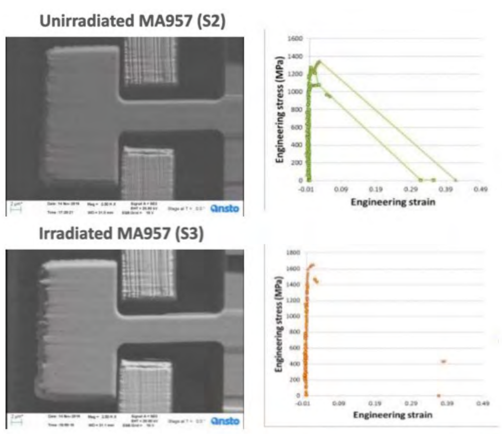
A detailed flowsheet analysis has also been carried out, and the ancillary components required for rendering the process a closed cycle were defined. Optimization of the process with respect to operating expenses (OPEX) and capital expenses (CAPEX) is being undertaken to fully understand their impact on the cost of hydrogen. As expected, the thermal and electrical demand of the process significantly affects the OPEX. A balance between the CAPEX and OPEX has been found to be necessary to arrive at a reasonable cost for the hydrogen produced. Further refinements of the analysis are being carried out with a more detailed look at large-scale hydrogen production.

Chinese efforts with regard to nuclear hydrogen production in the past years have focused on the development of the components of both the sulphur-iodine (S-I) and hybrid sulphur (HyS) processes. In the development of the components of the S-I process, two reactors, which intend to use the heat from the HTGR (the sulphuric-acid [H2SO4] decomposer and hydriodic acid [HI] decomposer), were designed and constructed. The SA decomposer was designed as a shell-and-tube heat exchanger, with the bayonet type silica carbide (SiC) tube as the reaction zone for SA decomposition, as well as the pressure boundary, and the integrity of SiC component has been verified with 100 hours of testing. In addition, the lifetime test of the catalyst for the SA decomposition reaction has been conducted for more than 700 hours and will be continued. The prototype SA decomposer is being manufactured and will be completed in several months. The HI decomposer, composed of an evaporator and an adiabatic reactor, is also being produced. At the same time, a high-temperature helium loop (>900°C, 100 kW) was designed to provide heat for the performance test of those components of the S-I process (see figure VHTR-7).

In the development of the HyS process, efforts have gone into the development of the sulphur dioxide (SO_2)-depolarized electrolyzer (SDE) stack, as well as the auxiliary facility for the test and operation of the SDE stack, particularly under enhanced pressure. A SDE stack with an H2 production rate of 100 NL/h has been developed and tested. Currently, the scaling-up of the stack is in progress.

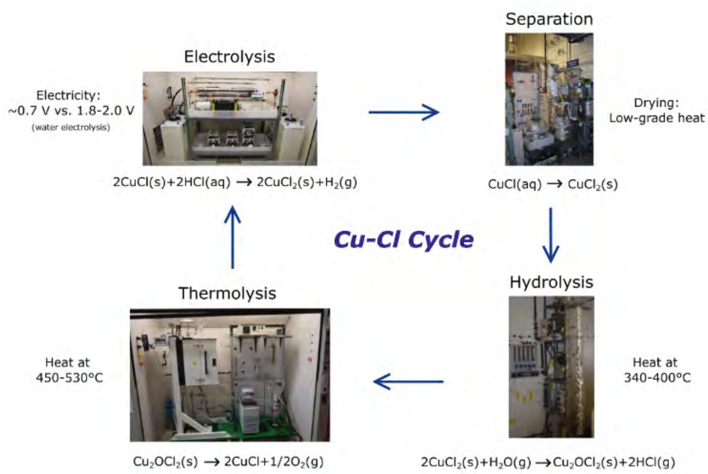
With the announcement of an ambitious recovery plan of EUR 7 billion over a ten-year period,

Figure VHTR-5. Use of in-situ nano-scale tensile specimens to evaluate irradiation effects in ion-beam exposed samples of ODS alloy MA957



Xu, A. et al. (2021).

Figure VHTR-6. Equipment used for the four steps of the Cu-Cl Cycle integration in Canada



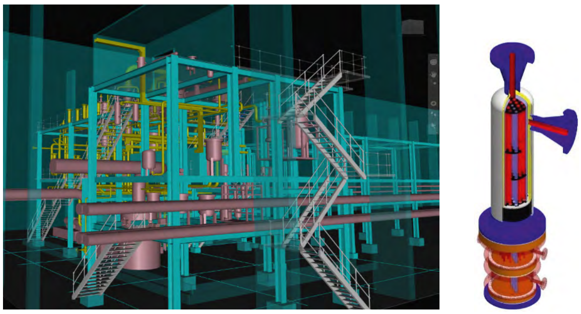
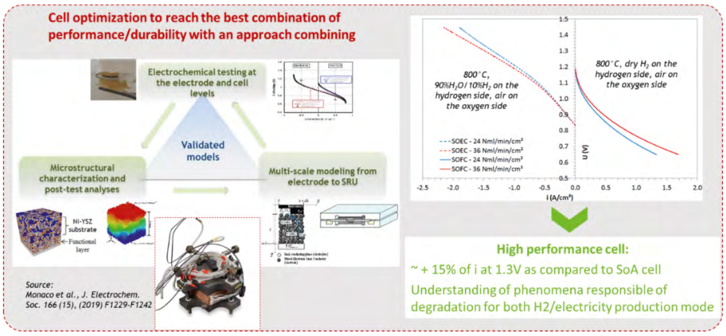


Figure VHTR-7. Sulphuric-acid and hydriodic-acid decomposition facilities and the high-temperature helium loop



Monaco, F. et al. (2019).

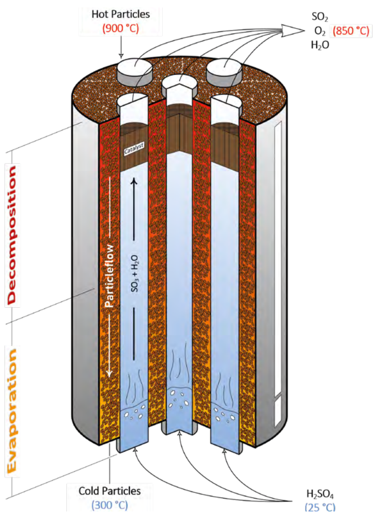
Figure VHTR-8. CEA activities to develop high-temperature steam electrolysis

hydrogen is a pillar of the energy transition in France and a market priority. Aligned with the EU Green Deal, the plan proposes financial incentives to foster clean H2 in industry (e.g. refinery and steel) and the transport sector (heavy or intensive duty vehicle). The electrolysis system, in particular high-temperature steam electrolysis (HTSE), is going to play a major role in producing clean hydrogen, thanks to the mix of nuclear energy and renewable energy. 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 for developing a pilot line and producing 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 modelling and characterization of the microstructure using the European Synchrotron Radiation Facility (ESRF), it has been possible to predict the performances of the cell by incorporating a mass transportation model. The CEA has also identified possible impacts of the electrical polarity in addition to the high temperature on the aggregation of the Ni phase on the electrode catalyst. This can explain the higher degradation of the performance of the cell in electrolysis mode compared to performance in fuel cell mode. Thanks to the recovery plan, research on the second generation of cell and stack has been moving forward at good speed with significant progress. Figure VHTR-8 shows a pictorial view of the overall activities at CEA in the development of the HTSE technology.

In the area of sulphur-based hydrogen production process development, different reactor concepts were developed and evaluated in the European PEGASUS project to demonstrate the feasibility of sulphuric-acid decomposition with high-temperature heat absorbed by particles. In the final design, an indirect contact approach, in a strictly counter-current moving bed heat exchanger (MBHE), was chosen. A proof of concept (POC) sulphuric-acid splitting/decomposition prototype driven by hot bauxite particles was developed and designed. The laboratory-scale test reactor is a novel counter-current flow shell-and-tube heat exchanger with particles on the shell side and sulphuric acid on the tube side, and with mass flow rates of 10 and 2 kg/h, respectively. A one-dimensional heat transfer model was developed based on correlations of the flowing fluid boiling heat transfer coefficient and particle bed heat transfer coefficient for sizing the shell-and-tube heat exchanger. A detailed study was carried out in order to choose suitable materials, particularly in the sulphuric-acid inlet and evaporation section. A new concept of an electrically heated, continuously operated particle heating system was designed and developed to provide the splitting reactor with hot particles. Different cases were studied using a finite element method analysis to qualify the particle heater and examine its thermo-mechanical stability.

A kinetic study of the sulphur-trioxide decomposition in the particle heated laboratory reactor for the EU research project PEGASUS was also carried out. The reactor (see Figure VHTR-9) was developed for the use of hot ceramic particles to evaporate sulphuric acid and dissociate into sulphur trioxide (SO_3) and water, and then further decompose SO_3 into SO_2 and oxygen in a sulphur-based thermo-chemical energy-storage cycle (TCES) for concentrated solar power (CSP) plants. The kinetic study is separated into two parts. First a literature search was conducted on available

Figure VHTR-9. Particles driven Sulphuric-acid splitting reactor



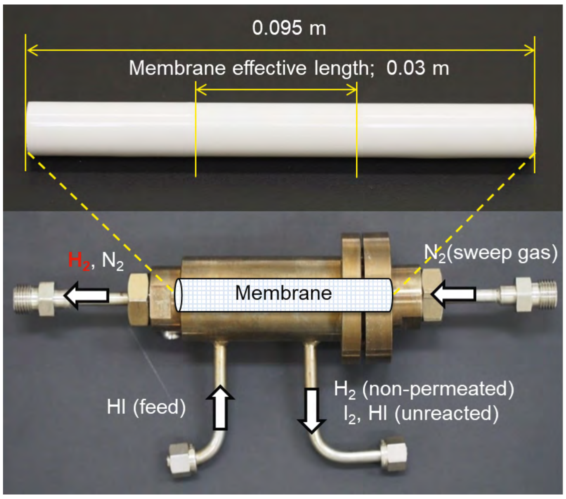
Thanda, V. K. et al. (2020).

kinetic models that describe the sulphur-trioxide decomposition, and these were evaluated. The kinetic models were found to be strongly limited with respect to their application range and several influencing factors such as active catalyst surface area, catalyst loading, and operating pressure – however, equilibrium effects were not considered by the kinetic models. Then, the kinetic models were implemented in a discretized model of the SO_3 decomposition section of the particle heated laboratory reactor. The simulation results were used to evaluate the reactor design, to find favourable operating conditions for the upcoming experiments and to develop the evaluation method for the experiments. The influence of parameters such as gas temperature at the catalyst inlet, the temperature difference between the particles and the gas, and the mass flow in the decomposer on the SO_3 decomposition were varied and the conversions calculated. With the simulation results, initial test plans for the upcoming experiments with a H_2SO_4 decomposition reactor were developed. Furthermore, the evaluation method for the experiments was defined based on the model equations for the decomposer. The evaluation method was implemented in software modules to prepare the interface of the evaluation method to the overall measurement and control software.

The JAEA has continued working on essential R&D tasks of the S-I process to verify the integrity of components made of practical structural materials and the stability of hydrogen production operation in harsh working conditions. For stable hydrogen production, technical issues for instrumental improvements (stable pumping of HI-I2-H2O solution, prevention of leakage, prevention of I2 precipitation) were resolved. In parallel, the JAEA has also focused on the development of membranes and a separation materials database, including selection development and performance assessments of separation techniques and materials. If hydrogen can be separated effectively from product gases such as HI and I2 without phase change, the thermal efficiency of the total IS process would increase and the cost of hydrogen would decrease. Therefore, the successful development of a hydrogen-separation membrane for the hydrogen iodine (HI) molecule decomposition is significant. The objective is to test the membrane and to investigate separation performance in HI decomposition. Silica ceramic membranes were selected for their thermal and chemical stability, thickness control capability, access to high permeation flux and high selectivity, as well as their feasibility of controlling the porosity structure. The JAEA succeeded in the preparation of silica ceramic membranes and the demonstration of a lab-scale catalytic membrane reactor for HI decomposition, as shown in Figure VHTR-10.

The Korean government released two roadmaps in 2019: 1) "Hydrogen Economy Roadmap" to drive a new growth engine and turn Korea into a society fuelled by eco-friendly energy; and 2) "Hydrogen Technology Development Roadmap" for technology development across ministries to support the implementation of the hydrogen economy by enhancing domestic technological competitiveness in the hydrogen energy sector. The establishment of these roadmaps has provided impetus to activities on hydrogen production. 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 S-I process. KAERI has been planning 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 considering use of the available helium loop facility for the integral test, which operates at 600 kWe and 950°C (see Figure VHTR-11).

Figure VHTR-10. A lab-scale catalytic membrane reactor for HI decomposition

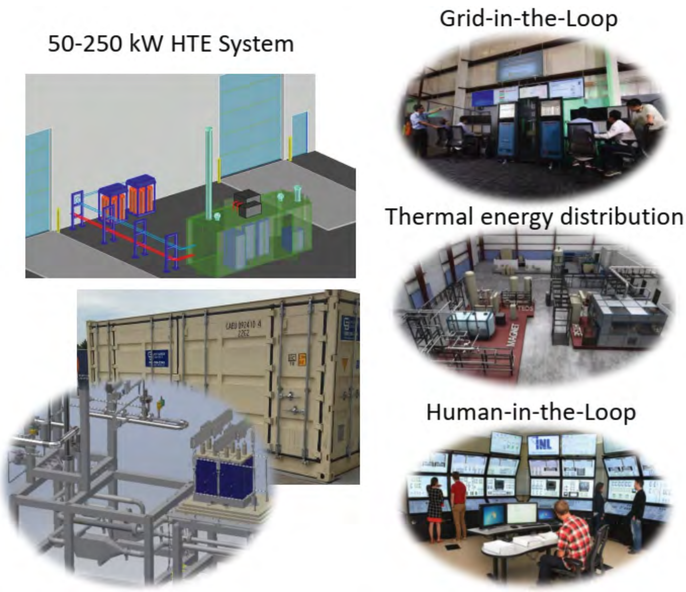


Myagmarjav, O. et al. (2019).

Figure VHTR-11. Component scale helium gas loop

The United States' efforts have continued on the development and application of HTSE for hydrogen production in the context of being a dynamic and flexible part of an 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 micro reactors (1 to 20 MW) and small modular reactors (20 to 300 MW) to full-size reactors (300 to 1 000 MW).

The objectives with respect to the HTSE have been to verify operation of solid oxide electrolysis cell (SOEC) stacks from US suppliers, qualify them for use in nuclear hydrogen demonstrations and benchmark stack performance under 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 (see Figure VHTR-12).



Boardman, R. (2020).

Figure VHTR-12. 250 kWe integrated HTSE system development

Computational methods validation and benchmarks

The computational methods validation and benchmarks (CMVB) project was restarted in 2014. From 2015 to 2020, a total of 11 meetings were organized by the CMVB pPMB and held in turn in different participating countries. The main activities resulting from these meetings included discussions and confirmation of the research tasks in each work package (WP), review and approval of the draft project plan, of which the final version is the indispensable annex of the project arrangement (PA), discussions on some common topics and potential test facilities that will be fundamental resources of this project, and the process and guidelines to launch the PA. To date, the PP has been approved by all SSC members. In 2020, confirmation of the PA was under processing by each signatory (China, the EU, Japan, Korea and the United States). Signatories are ready to pursue the signature for the PA.

Because of the impact of COVID-19, the originally planned 22nd CMVB pPMB meeting in the spring of 2020 (to be held in China) was postponed. Instead, a video conference was held online. Considering the status of the CMVB PA and PP, the main objective of the meeting was to review the CMVB pPMB and receive updates on the status of CMVB R&D work from each participant. The United Kingdom showed interest in joining the project. The United Kingdom signed the GIF VHTR system arrangement in 2019. UK representatives were invited as observers to discuss potential interest in collaborating within GIF, and these representatives presented the UK's VHTR CMVB activities, identifying what the United Kingdom could potentially contribute to the current project. CMVB members are now ready to sign the PA. Through pPMB meetings, past, current and new test facilities and projects have been identified, proposed and confirmed as fundamental resources for the development and assessment of codes and models covering HTR physics, TH, CFD, fission product transport, plant dynamics, etc.

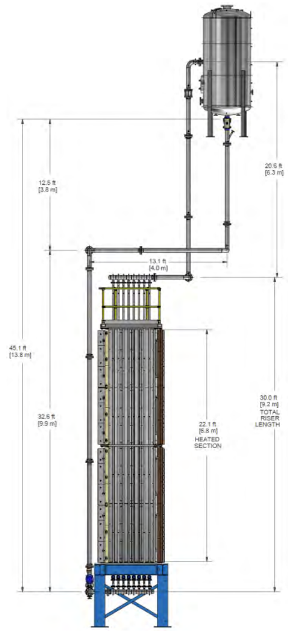
Table VHTR-1. Work Package titles of the VHTR CMVB PMB

WP No.	WP title	Lead
1	Phenomena identification and ranking table (PIRT) methodology	DOE (US)
2	Computational fluid dynamics (CFD)	INET (CHINA)
3	Reactor core physics and nuclear data	DOE (US)
4	Chemistry and transport	INET (CHINA)
5	Reactor and plant dynamics	INET (CHINA)

In China, the HTR-PM demonstration project is in its commissioning stage. In 2020, the reactor pressure vessel, the steam generator pressure vessel and the hot gas duct pressure vessel were connected in both of the two NSSS modules. The two modules underwent helium leak detection of the pressure boundary and cold tests, including strength and leakage tests. Then, the primary circuits were heated by the helium blower. On 30 December, the two modules achieved the status of 250°C and 7 MPa, under which the hot test could be performed. The first fuel loading is planned for the first half of 2021. The design of the HTR-PM600, a

600 MWe commercial plant, was pursued in 2020. The emphasis of the HTR-PM600 project in 2020 was on the feasibility study, preliminary design and preparation of the PSAR. Regarding HTR-10, the in-core temperature measurement experiment was conducted and completed after it was restarted, with the purpose of determining temperatures inside the fuel elements (see Figure VHTR-14). The experimental results have been summarized and used to support the safety review of the HTR-PM. In the Institute of Nuclear and New Energy Technology (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 (V&V) was carried out in 2020 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 domestic codes are supposed to be used in the design verification of HTR-PM600 as the first step of their application.

In the EU, most of the current (V)HTR-related activities are taking place in the Euratom Horizon 2020 project GEMINI+, which is supporting the demonstration of an HTGR nuclear cogeneration system. The outcome will be submitted to the CMVB PMB as the GIF contribution, in addition to the (V)HTR-related projects in past Euratom Framework Programmes (and EC national projects). Some additional, national HTR-related projects will deliver contributions to the CMVB: the Polish projects GOSPOSTRATEG – HTR (2019-2022), and the NOMATEN Centre of Excellence (2019-2026).

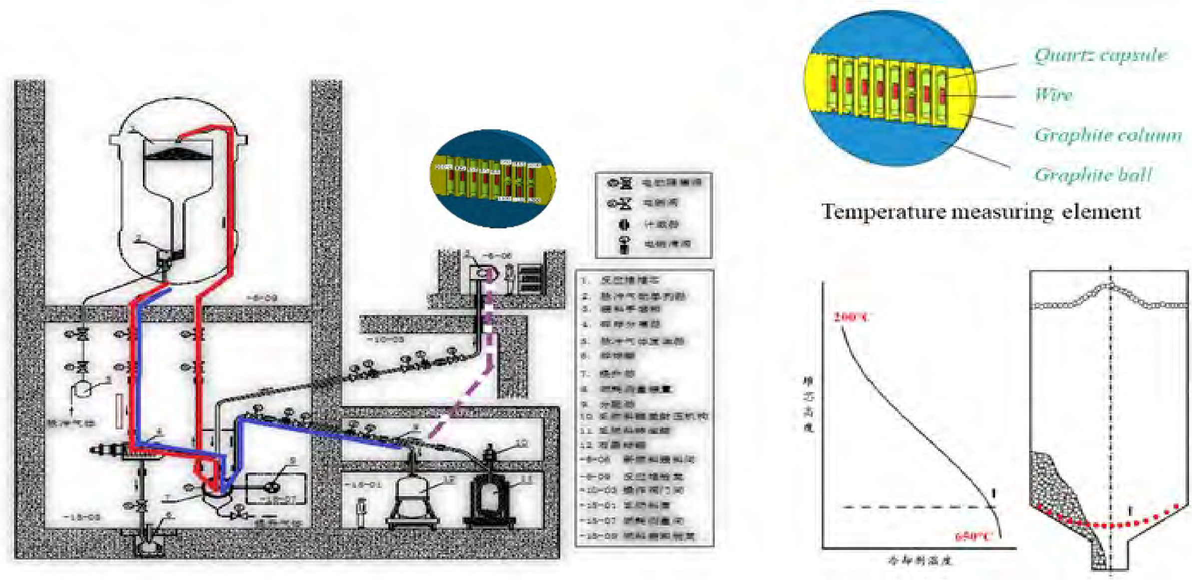


Zhang, Y. et al. (2021).

Figure VHTR-13. Schematic diagram of water-cooled natural convection shutdown heat removal test facility

The JAEA is making a strong effort to restart the HTTR. It is developing and benchmarking various models and analysis methodologies, as well as codes for reactor physics, thermal fluids, etc. JAEA R&D is expected to support planning of the CMVB co-operative activities, such as a benchmark activity using ATR irradiation data. Consequently, the JAEA defined a provisional calculation problem to verify the geometrical model for MVP code, based on an ATR critical experiment in 2020. R&D on a system analysis code based on RELAP5/MOD3 code was

Figure VHTR-14. Schematic diagram of in-core temperature measurement experiment of HTR-10[1]



Lisowski, D. et al. (2017).

1. Launch temperature measuring elements from the top of the core, then start to shuffle the fuel elements and temperature measuring elements to a certain position. After running the reactor at 3 MW for 10 hours, then shutdown the reactor and discharge the elements. If the temperature measuring element is identified by the X-ray machine, it will be removed from the discharged loop for further analyses.

performed for transient thermal-hydraulic behavior in a prismatic-type VHTR. The flow distribution analysis model and molecular diffusion model were newly developed and validated for the code to simulate key thermal-hydraulic phenomena in a prismatic-type VHTR. Based on this study, the JAEA will explore the possibility of further collaboration, such as the launch of a new benchmark task in CMVB.

The VHTR R&D program in Korea aims at improving high-temperature system key technologies in terms of the design code development and assessment, and also high-temperature materials performance verification. With regard to its VHTR program, the five-year project on the development of HT system key technologies was launched in 2020, with the aim of developing the HT system performance evaluation technology and verifying the materials performance for VHTR. Some specific CMVB related R&D activities include scale-down standard fuel block tests to validate 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.

Regarding CMVB more specifically, neutronics code improvement is underway to predict power distribution precisely. Neutronics/system code coupled calculations have been updated to enhance the thermal margin. Fission product transport from fuel to containment will be assessed under normal and accident conditions. An HTSE system analysis/experiment project will be launched in 2021.

In the United States, the latest progress was from metals and TRISO fuel-related activities. Alloy 617 has been approved by the ASME for inclusion in its boiler and pressure-vessel code. This means the alloy, which was tested by Idaho National Laboratory (INL), can be used in proposed molten salt, HT, gas-cooled, or sodium reactors. It is the first new material to be added to the code in 30 years. The UCO TRISO fuel performance topical report submitted by the Electric Power Research Institute (EPRI) is being reviewed and approved by the regulator (i.e. NRC). In addition, modelling of the HT test facility (HTTF) with RELAP5-3D was performed, and the two-phase testing at water-cooled Natural Convection Shutdown Heat Removal Test Facility (NSTF) was ongoing.

Michael A. Fütterer
Chair of the VHTR SSC, with contributions from VHTR members