Very-high-temperature reactor (VHTR)

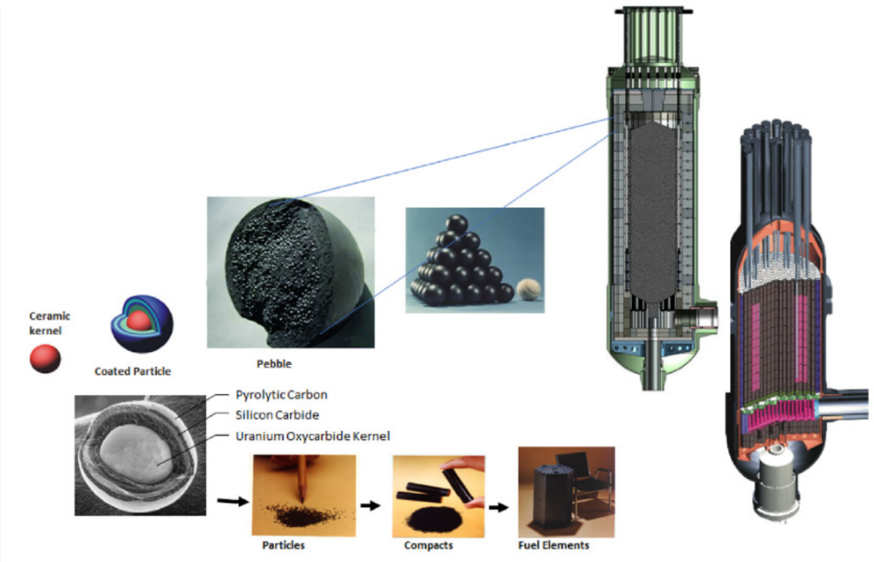
Main characteristics of the system

The very-high-temperature reactors are the descendants of the high-temperature reactors developed in the 1970s-1980s. They are characterized by a fully ceramic coated-particle fuel, the use of graphite as neutron moderators, and helium as coolant, self-acting decay heat removal capability, resulting in inherent safety and process heat application capability.

Use of helium as coolant and ceramics as core structure material allows operation temperature at core outlet of 850°C or above allowing for hydrogen production using processes with no greenhouse gas emission, such as thermo-chemical cycles (Sulphur-Iodine process) or High-Temperature Steam Electrolysis (HTSE). Beyond electricity generation and hydrogen production, high-temperature reactors can provide process heat for use in other industries, substituting fossil fuel applications

As previously noted, the basic technology for the VHTR has been established in former high-temperature gas-cooled reactors such as the US Peach Bottom and Fort Saint-Vrain plants as well as the German AVR and THTR prototypes, also the test reactors HTTR in Japan and HTR-10 in China. These reactors represent the two baseline concepts for the VHTR core: the prismatic block-type and the pebble bed-type (see **Figure VHTR 1**).

Figure VHTR 1. **TRISO coated-particle fuel as the basis for hexagonal block and pebble bed core designs**



The fuel cycle will initially be once-through with low-enriched uranium fuel and very-high-fuel burn-up, with plutonium or thorium-based fuels as alternatives. Solutions need to be developed to adequately manage the back-end of the fuel cycle. The potential for a closed fuel cycle needs to be fully established. Although various fuel designs are considered within the VHTR systems, all concepts exhibit extensive similarities allowing for a coherent R&D approach, as the TRISO coated-particle fuel form is the common denominator for all. This fuel consists of small particles of nuclear material, surrounded by a porous carbon buffer, coated with three layers: pyro-carbon/silicon carbide/pyro-carbon. These coatings represent the first barrier against fission product release under normal operation and accident conditions.

Former HTR reactors, such as AVR and HTTR, were already operated at temperatures up to 950°C. The VHTR can now supply heat and electricity over a range of core outlet temperatures between 700 and 950°C, or more than 1 000°C in future. The available high-temperature alloys used for heat exchangers and metallic components determine the current temperature range of VHTR (~700-950°C). The final target for GIF VHTR has been set at 1 000°C or above, which implies the development of innovative materials such as new super alloys, ceramics and compounds. This is especially needed for some non-electric applications, where a very high temperature at the core outlet is required to fulfil the VHTR objective of providing industry with very-high-temperature process heat.

In the current projects of VHTR, the electric power conversion unit is an indirect Rankine cycle applying the latest technology of conventional power plants, as this technology is available. Direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles can be envisaged in the longer term.

The experimental reactors HTTR (Japan, 30 MWth) and HTR-10 (China, 10 MWth) support the advanced reactor concept development for the VHTR. They provide important information for the demonstration and analysis of safety and operational features of VHTRs. This allows improving the analytical tools for the design and licensing of commercial-size demonstration VHTRs. The HTTR, in particular, will provide a platform for coupling advanced hydrogen production technologies with a nuclear heat source at a temperature up to 950°C.

The technology is being advanced through near and medium-term projects, such as HTR-PM, NGNP, GT-MHR, NHDD, and GTHTR300C, led by several plant vendors and national laboratories respectively in China, the United States, Korea and Japan. The construction of the HTR-PM demonstration plant (two pebble bed reactor modules with one super heated steam turbine generating 210 MWe) is currently being finalized. Each reactor module has a thermal power of 250 MWth. The coolant gas temperature will be 750°C, which represents the current state of the art for materials and the requirement of high-temperature steam generation. High quality steam of 566°C from either reactor will be fed into a common steam header and turbo generator set. The HTR-PM demonstration plant will be connected to the grid in 2020, representing a major step towards a Generation IV demonstration plant.

Status of co-operation

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 SA during the Policy Group meeting held in Beijing. South Africa, formally acceded to the GIF Framework Agreement in 2008, but announced in December 2011 that it no longer intends 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 new members of the system arrangement was subsequently signed by Australia (December 2017) and the United Kingdom (January 2019).

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 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 Materials Project Plan (PP) was completed in 2012, the Materials Project Arrangement (PA) continued through 2019 under its 1st amendment, which added China as a Signatory, while simultaneously pursuing a 2nd amendment that would incorporate a new PP for activities from 2018-2022 and add Australia as another Signatory. Contributions to the new PP for 2018-2022 were developed by the current seven Signatories (China, European Union, France, Japan, Korea, Switzerland, and United States), as well as Australia, which will be joining the PA. This 2nd amendment of the PA (incorporating the new PP and Australia) was approved by the SSC in April 2019 and was distributed by NEA for signature on 20 November 2019.

The hydrogen production PA became effective on March 2008 with implementing agents from Canada, France, Japan, Korea, the United States and Euratom. In 2019, the forthcoming five-year Project Plan was prepared to incorporate Chinese contributions and other countries' updated contributions. The finalized Project Plan is expected in early 2020.

The computational methods validation and benchmarks (CMVB) PA remained provisional. In 2019, detailed discussions on finalizing a multi-year work plan were performed by the participants. The PA is now ready for final approval by the VHTR SSC.

R&D objectives

Even if the VHTR development is mainly driven by the achievement of very-high-temperatures providing higher thermal efficiency for new applications, 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 those challenging R&D goals.

The VHTR system research plan describes the R&D programme to establish the basic technology of the VHTR system. As such, it is intended to cover the needs of the viability and performance phases of the development plan described in the Generation IV Technology Roadmap and in the GIF R&D Outlook (2018 Update). From the six projects outlined in the VHTR SRP, three are effective, and one is provisional, as discussed below:

- Fuel and fuel cycle (FFC) investigations are focusing on the performance of the TRISO coated particles, which are 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-oxicarbide 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 (MA) in support of a closed cycle.
- Materials (MAT) development and qualification, design codes and standards, as well as manufacturing methodologies, are essential for the VHTR system development. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in the operating environments. For core coolant outlet temperatures up to around 950°C, it is envisioned 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. Improved multi-scale modelling is needed to support inelastic finite element design analyses. In addition to other high-temperature heat exchangers, additional attention is being paid to the metal performance in steam generators, which reflects the current interest in steam-based process applications at somewhat lower core outlet temperature of 750 to 850°C. Structural materials are considered in three categories: graphite (for core structures, fuel matrix, etc.), very/medium-high-temperature metals, and ceramics & composites. A materials handbook has been developed and is being used to efficiently store and manage VHTR data, facilitate international R&D co-ordination, and support modelling to predict damage and lifetime assessment.
- For hydrogen production (HP), 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 requirements: 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 will be 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 will also be investigated and design-associated risk analysis will be 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 order to make them compatible also with other Generation IV nuclear reactor systems dealing with a lower temperature range.

- Computational Methods Validation and Benchmarks (CMVB) in the areas of thermal-hydraulics, thermal-mechanics, core physics, and chemical transport are major activities needed for the assessment of the reactor performance in normal, upset and accident conditions and for licensing. Codes validation needs to be carried out through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported by HTTR and HTR-10 tests or by past high-temperature reactor data (e.g. AVR, THTR and Fort Saint-Vrain). Improved computational methods will also facilitate the elimination of unnecessary design conservatisms and improve construction cost estimates.
- Even though it is not currently implemented, the development of components needs to be addressed for the key reactor systems (core structures, absorber rods, core barrel, pressure vessel, etc.) and for the energy conversion or coupling processes (such as 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, capable of simulating normal and off-normal events. The project on components should address development needs that are in part common to those of the GFR, so that common R&D could be envisioned for specific requirements, when identified.

System integration and assessment (SIA) is necessary to guide the R&D to meet the needs of different VHTR baseline concepts and new applications such as cogeneration and hydrogen production. Near- and medium-term projects should provide information on their designs to identify potentials for further technology and economic improvements. At the moment, this topic is directly addressed by the System Steering Committee.

Milestones

In the near term, lower-temperature demonstration projects (from 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 high-temperature alloys, qualification of new graphite types and development of composite ceramic materials. Lower temperature version of VHTR (from 700°C to 950°C) will enter the demonstration phase around 2020, based on HTR-PM experience in China which is scheduled to operate in 2020. A future higher temperature version (1 000°C and above) will require more research.

Main activities and outcomes

Fuel and fuel cycle (FFC) project: The Very-High-Temperature Reactor (VHTR) Fuel and Fuel Cycle (FFC) project is intended to provide demonstrated solutions for the VHTR fuel (design, fabrication, and qualification) and for its back-end management, including novel fuel cycle options.

Tri-structural isotropic (TRISO) coated particles, which are the basic fuel concept for the VHTR, need to be qualified for relevant service conditions. Furthermore, its standard design – uranium dioxide (UO_2) kernel surrounded by successive layers of porous graphite, dense pyro-carbon (PyC), silicon carbide (SiC), then PyC – could evolve along with the improvement of its performance through the use of a uranium oxycarbide (UCO) kernel or a zirconium carbide (ZrC) coating for enhanced burn-up capability, minimized fission product release, and increased resistance to core heat-up accidents (above 1 600°C). Fuel characterization work, post-irradiation examinations (PIE), safety testing, fission product release evaluation, as well as the measurement of chemical and thermo-mechanical material properties in representative conditions will feed a fuel material data base. Further development of physical models enables assessment of in-pile fuel behavior under normal and off normal conditions.

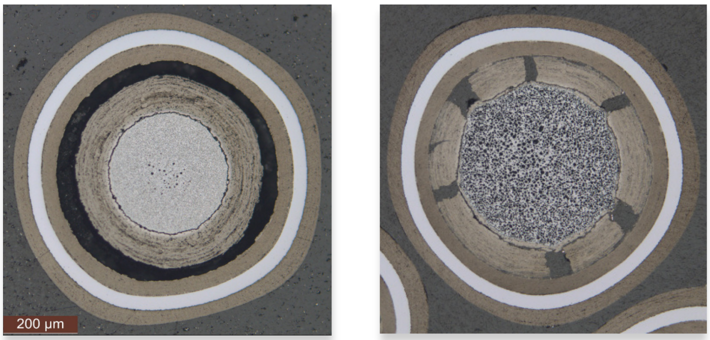
Fuel cycle back-end encompasses spent fuel treatment and disposal, as well as used graphite management. An optimized approach for dealing with the graphite needs to be defined. Although a once-through cycle is envisioned initially, the potential for deep burn of plutonium and minor actinides in a VHTR,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

Irradiation and PIE

This work package includes the activities of fuel irradiation testing, PIE facility and equipment development, and post-irradiation examination of fuel specimens. Work in China has continued to develop domestic fuel post-irradiation examination capabilities. This includes hot cells and equipment for fuel heating tests.

Post-irradiation examination on the AGR-2 fuel (including both UCO and UO_2 TRISO particles) has continued in the United States. This includes destructive examination of fuel compacts and particles. Up to this time, 11 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 the migration of fission products in the layer (see **Figure VHTR 2**).

Figure VHTR 2. **Micrographs of UCO TRISO particles from an AGR-2 compact irradiated to an average burn-up of 12.0% FIMA**



The US AGR-5/6/7 irradiation of UCO TRISO fuel continues in the Advanced Test Reactor. This experiment is both the final fuel qualification irradiation and a separate high-temperature fuel performance margin test (peak temperatures of ~1 500°C) and contains approximately 570 000 fuel particles in 194 fuel compacts. The irradiation is roughly half complete.

The Unites States has also recently developed – and is currently using – the capability to re-irradiate fuel specimens prior to performing heating tests. This capability is essential for measuring the release of short-lived fission products (including 131I) that can be significant contributors to off-site dose during reactor accidents. The fuel specimens (previously irradiated in the Advanced Test Reactor), are re-irradiated in the Neutron Radiograph (NRAD) reactor located at the Hot Fuel Examination Facility at INL, where they can quickly be removed from the reactor and transported to the hot cell for heating tests.

Fuel attributes and material properties

The FFC PMB organized the 5th Workshop on High-Temperature Gas-Cooled Reactor SiC Material Properties in conjunction with the 15th official meeting of the PMB at ORNL in May 2019 (34 people participated from five different countries). The participants ranged from members of academia, industry, national laboratories, and intergovernmental agencies. The meeting was divided into technical sessions including 16 technical presentations along with significant discussion focusing on scientific challenges facing tri-structural-isotopic (TRISO) fuel for HTGR applications. Technical topics broadly covered two different areas: issues surround SiC coating layers in TRISO fuel, and oxidation of TRISO fuel materials. The meeting also included a series of tours focusing on ORNL's past and present nuclear research and development capabilities.

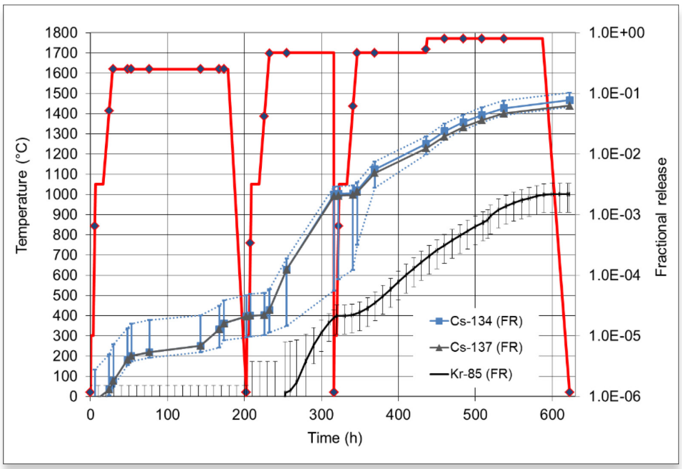
The United States, China, and Korea are completing the final stages of an as-fabricated fuel characterization "round robin" experiment. The work involves performing leach-burn-leach analysis on well-characterized particle specimens to detect defective SiC coatings and complete through-layer coating defects. The specimens were provided by the United States and China. As of the end of 2019, all of the experimental work has been completed. The United States is preparing a final report that summarizes these results.

The United States, Japan, and Korea have also completed a TRISO fuel accident test computational benchmark task. All three participants used fuel performance models to predict fission product release from TRISO fuel during heating tests in pure helium. The results of the predictions were compared with the experimental results from the safety tests performed on fuel in the United States and the EU. A draft report has been prepared and will be issued in 2020.

Safety

High-temperature safety tests are being performed at JRC Karlsruhe on HTR-PM spheres that were previously irradiated in HFR. The tests have been performed at temperatures ranging from 1 620 to 1 770°C for a total of 450 hours for each sphere. A total of four tests have been completed. Krypton release during the tests remained below the level of a single particle, indicating no particle with complete TRISO failure. Cesium releases were below ~2×10^{-5} for 150 h at 1 620°C, but increased at longer exposures and higher temperatures, indicating gradual degradation of the SiC layers. And example of the cesium and krypton release data is shown in **Figure VHTR 3**. In addition to the HTR-PM sphere tests at Karlsruhe, China is also deploying a KÜFA heating test capability at the hot cells at INET. The system has been installed in the hot cells and is undergoing testing.

Figure VHTR 3. **Fractional release of fission products Cs-134, Cs-137, and Kr-85 from an HTR-PM fuel specimen irradiated in HFR Petten and heated in the KÜFA facility. The heating program is shown in red**



In the United States, high-temperature safety tests of AGR-2 UCO and UO_2 fuel compacts in pure helium have continued. A total of 16 safety tests have been performed at temperatures ranging from 1 500°C to 1 800°C. One of these tests was performed with a test temperature that varied over time in a manner similar to the predicted peak fuel temperature in a modular HTGR during a depressurized loss of forced cooling. The results have indicated no TRISO failure during testing of UCO temperatures of 1 800°C for 300 h and testing of UO_2 at 1 700°C for 300 h. Cesium release from UCO fuel remains low during the tests (highest releases are ~3×10^{-4} after 300 h at 1 800°C), but somewhat higher for UO_2 fuel (release as high as 9×10^{-2} observed after 300 h at 1 700°C).

The United States is also performing 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. Some of these compacts have been re-irradiated in the NRAD reactor to generate short-lived ^{131}I prior to the heating tests. These tests are therefore being used to assess fission product release from exposed kernels.

A dedicated furnace designed to heat irradiated TRISO fuel specimen as high as 1 600°C in oxidizing atmospheres is currently being developed at INL in the United States. The system will be used to test oxidation behavior of fuel and fuel materials in air/He and moisture/He gas mixtures, while monitoring the release of fission products and reaction products in real time. The system is expected to be deployed in 2021.

The United States has also prepared a topical report on UCO TRISO fuel performance in co-operation with the Electric Power Research Institute (EPRI) that describes the results of the AGR-1 and AGR-2 irradiation experiments and subsequent PIE. The report was submitted to the Nuclear Regulatory Commission (NRC) for review. The objective of this report and NRC review is to obtain agreement from the US regulatory authority that the fuel performance data from these experiments can be used by future reactor designers in their licensing submissions.

In Japan, researchers are studying the oxidation of the TRISO SiC layer at the SiC-OPyC boundary. This includes computational modelling of the mechanism of oxidation and the influence of such parameters as temperature, O_2(g) concentration, and transport to the SiC layer through the OPyC. A series of experiments is proposed using TRISO particles with surrogate kernels at temperatures up to 1 600°C and O_2 concentrations of 20 ppm to 20%. JAEA has proposed a new computational benchmarking activity that will focus on the release behavior of short-lived fission gases.

Enhanced and advanced fuel fabrication

Development of fabrication of larger UO_2 kernel sizes that typically used in TRISO fuel is being pursued in Korea. Researchers are targeting sintered kernel sizes of 800 μm, for potential application in accident-tolerant fuels. Experiments have been successful in producing kernels in excess of 800 μm diameter, and work continues to refine the process to improve kernel properties. In conjunction with this effort, coating process for the larger kernels are also being developed. To date this has included computational modelling of the fluidized particle bed, and experiments are planned in the future. Finally, development of double-layer ZrC/SiC TRISO coatings with improved properties continues to be studied in Korea.

China is studying equipment and processes for fabricating ZrC coatings as a potential replacement for SiC in TRISO fuel. Fabrication of UCO kernels is also being pursued.

Significant recent work has been performed on PIE and safety testing of TRISO fuel and new PIE and safety testing capabilities are being developed by several members. The Project Management Board has produced results on two collaborative projects: an LBL round robin experiment and an accident testing computational benchmark. This has led to the creation of a third five-year plan.

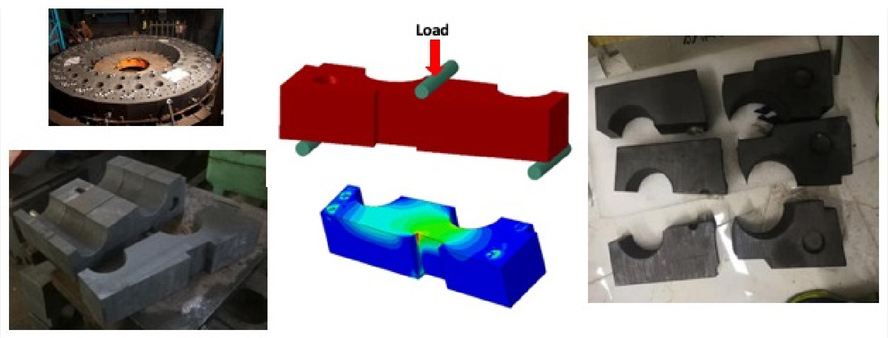
Materials project

As part of the development of the new Program Plan, a thorough review was made of all the High-Level Deliverables (HLDs). All HLDs scheduled for completion prior to the end of 2015 were adjusted for completion during the term of the new PP. Additionally, by the end of 2019, over 420 technical reports and over 10 000 materials test records describing contributions from all signatories had been uploaded into the Gen-IV Materials Handbook, the database used to share materials information within this PMB. This reflects the outstanding technical output of the membership that has now been shared to support system design and codes & standards development.

In 2019, research activities continued focused on near- and medium-term projects 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. Mechanical, physical, and fracture properties behavior were examined for numerous grades. Graphite irradiations and post-irradiation examinations & 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. One area of significant interest among signatories is the validation of the anticipated multi-axial loading response of graphite from dimensional changes and seismic events. A figure illustrating large-scale experiments on graphite blocks to validate design models is shown in **Figure VHTR 4.**

Figure VHTR 4. **Fracture testing of large graphite blocks with complex geometry to verify failure probability calculations for HTR-PM construction**

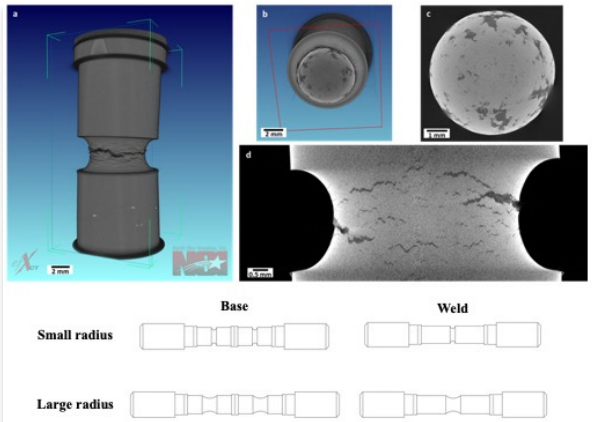


Courtesy of Institute of Nuclear and New Energy Technology.

Data to support graphite model development was generated in the areas of microstructural evolution, irradiation damage mechanisms, and creep. Support was provided for both ASTM and 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 (especially weld behavior in 800H and 617) provided very useful information for their use in heat exchanger and steam generator applications. These studies included an evaluation of the existing data base and an extension of it through aging, creep, creep-fatigue and creep crack growth rate testing to 950°C. Examination of enhanced diffusion bonding techniques for construction of compact heat exchangers is showing very promising results. The most significant outcome of high-temperature alloy work was approval of the ASME Code Case for the use of Alloy 617 as a new construction material for high-temperature nuclear components at temperatures to 950°C for 100 000 hrs. Data for the Code Case was contributed from multiple Signatories (DOE, KAERI & CEA). Other metallic materials were also examined. Irradiation and irradiation creep was studied on 9Cr-1Mo ferritic-martensitic steels and oxide-dispersion-strengthened steels, plus creep behavior was examined in 2.25Cr-1Mo steel for steam generator applications.

Input for improvements in High Temperature Design Methodology (HTDM) were also contributed by participants. Removing unnecessary conservatism through improvement in analysis procedures and extending the applicability of the rules to longer life times or to a wider selection of materials could provide greater design flexibility and potential innovative designs to enhance safety or reduce construction costs. During 2019, constitutive models and inelastic analysis methods were developed to better define and extend the applicability of simplified design methods to maximum upper temperature limits. An example of experimental support needed for the HTDM improvements concerning multi-axial loading is illustrated in **Figure VHTR 5**. Creep testing and failure process assessment of different V- and U-notched specimens were performed to determine if a crossover from notch-strengthing to notch-weaking occurs in Alloy 617 base and weld metal at times up to 100 000 hrs.

Figure VHTR 5. **Specimens and example examination for creep testing of Alloy 617 at 800°C to assess effects of notch strengthening versus notch weakening**



Courtesy of Idaho National Laboratory.

In the near/medium term, metallic alloys are considered as the main option for control rods and internals in VHTR projects, which target 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. Limited work continues to examine the thermo-mechanical properties of SiC and SiC-SiC composites and oxidation in C-C composites. Studies of fabrication, architecture, and processing on the properties and fracture mechanisms of the composites is being investigated. The results of this work is being actively incorporated into developing testing standards and design codes for composite materials, and to examine irradiation effects on ceramic composites for these types of applications. A significant milestone in this area occurred in 2019 with drafts on all articles related to General Requirements and Design Rules for Ceramic Components of ASME Code Section III Division 5 (Rules for Construction of High Temperature Reactor Components) having been completed and submitted for ballot.

Hydrogen production project

In 2019, the international hydrogen community saw a tremendous interest towards bringing hydrogen economy to reality through a range of applications led by the transportation sector. This enthusiasm was also apparent in the progress reported by the signatories at 19th and 20th Official Hydrogen PMB meetings held in Grenoble and Shanghai, respectively, during the year.

Canadian efforts on hydrogen production focused mainly towards the demonstration of an integrated Copper-Chlorine Cycle (hybrid thermo-chemical process) at a laboratory scale (50 L/h H_2 production) by 2021 March. Experimental development of equipment required to carry out each of the four steps of the process led to the following advances during the year: an electrolyser (electrolysis of CuCl/HCl producing H_2) design capable of producing up to 100 L/h H_2 was operated over several weeks with consistent performance; separation of CuCl/$CuCl_2$ was simplified; and an innovative method for the decomposition of Cu_2OCl_2 (an intermediate product) to produce O_2 was successfully demonstrated. Development of an efficient method for the hydrolysis step (reaction between $CuCl_2$ and steam) is being investigated to complete the integration of all the steps involved. In parallel, modelling of the process using Aspen Plus V9 is being carried using an updated database for physical properties of the various species involved in the process.

In China, the HTGR reactor development continued, and when completed is expected to provide the power and heat required for the hydrogen production processes being developed, namely Sulphur-Iodine (S-I), High Temperature Steam Electrolysis (HTSE) and Hybrid Sulphur (HyS) Processes. In the development of the S-I process, focus has been on the construction and simulation of the sulphuric acid Bayonet-type decomposer, the construction of the hydrogen iodide decomposer for hydrogen production at 1 Nm^3/h and the intermediate He loop heat exchanger (100 kW) to satisfy the heat requirement of the S-I process. In the development of HyS process, efforts have gone into simulation of the process, fundamental studies, including simulation of the SO_2 depolarized electrolyser (SDE). A facility for testing a stack of six units (each 200 cm^2) of the SDE has been designed and built. In the meantime, an agreement has been signed by Tsinghua University, China National Nuclear Corporation and China Baowu Steel Corporation to jointly advance nuclear hydrogen technologies for application in steelmaking – an exciting development.

CEA in France is taking an integrated R&D approach for nuclear and renewable energy integration in establishing their overall energy system. The main emphasis of their Low Carbon Energies Division on hydrogen production is in the development of HTSE. The generic development of the cells and stacks have included optimization of the solid oxide cells through thicker oxygen electrodes and thinner barrier layers for performance enhancement and minimization of degradation of cell components for long-term durability. They have also adapted the original thick-cell stack designs to thinner cells and Solid Oxide Fuel Cell (SOFC) operation. During the year they focused on the development of reversible systems for nuclear coupling to allow switching of the electrolyser operation to low-power fuel cell mode when the nuclear reactor is not producing power. Their first reversible industrial system (supplied in 2018) with one stack producing 1 Nm^3/h of hydrogen and 1 kWe in fuel cell mode has continued to operate with electrical efficiency at 84% in electrolyser mode and 55% in fuel cell mode.

Hydrogen production technologies related developments from EU have focused on HTSE and HyS processes. Although the reported work focused on coupling of these processes to solar power production, the actual technical aspects of these hydrogen production processes apply equally to nuclear systems. A steam electrolyser system producing 6.7 NL/min of hydrogen has been built and operated at ~750°C at DLR (Deutsches Zentrum für Luft- und Raumfahrt). Developments on the hybrid sulphur process has progressed under the European research project SOL2HY2. In the first of the two main steps, sulphuric acid is decomposed at high temperatures forming oxygen as a product and SO_2 for the subsequent electrolysis step. SO_2 is then electrolysed at about 80°C with water to produce hydrogen as the main product. Because of the low voltage requirement for this electrolysis step, the power consumption is significantly lower compared to conventional water electrolysis, leading to a significant efficiency gain.

JAEA has been developing various corrosion resistant components for the S-I process, and have incorporated them for the latest 150 h test (completed in January 2019) of the integrated system for hydrogen production at 30 L/h. Following the test, they have been carrying out inspection of materials of components to investigate any corrosion that may have taken place during the test and its impacts. Initial observations have revealed that the improvements made on the glass-lined sheath in HI sections functioned well.

During the year, two roadmaps were released by the Korean government: 1) "Hydrogen Economy Roadmap" in January 2019 to drive a new growth engine and turn Korea into a society fueled by eco-friendly energy, and 2) "Hydrogen Technology Development Roadmap" in October 2019 for technology development across ministries to support the implementation of the hydrogen economy by enhancing domestic technological competitiveness in the hydrogen energy sector. This establishment of the roadmaps on hydrogen economy provided impetus to activities on hydrogen production reported at the Hydrogen PMB meetings during the year. Simulations have been carried out on coupling various hydrogen production processes to a 350 MWth HTGR. The hydrogen production processes included Steam Methane Reforming, HTSE and S-I Process. Component test facilities with nitrogen (**Figure VHTR 6**) and helium loops operating at 60 kWe and 600 kWe respectively, and 950°C have been used to derive databases for Code Verification and Validation. Studies included on sulphuric acid decomposer, corrosion resistance and SiC coating of fluid channels. Emphasis has also been placed on manufacturing of components and transfer of technologies to private industry.

Figure VHTR 6. **Small-scale nitrogen gas loop used for studies involving sulphuric acid decomposition and SiC coated process heat exchangers**

During the year, the American activities under DOE-NE Nuclear-Renewable Integrated Energy Systems (IES) have focused on modelling and simulation, demonstration/experimental systems and stakeholder engagement. INL has established a Dynamic Energy Transport and Integration Laboratory (DETAIL) that will consist of multiple heat and electricity producers, thermal and electrical storage, and multiple heat and electricity customers coupled via a thermal and electrical network (**Figure VHTR 7**). The combined system will provide a demonstration of real-time integration with electrical grid, renewable energy inputs, energy storage and energy users. The entire energy network can be simulated to understand how to optimize energy flows while maintaining stability and efficient operation of all assets in the system. Related to advanced hydrogen production research, a 25 kW high-temperature electrolysis research and demonstration facility (**Figure VHTR 8**) has been designed, installed and commissioned with initial testing at 5 kW scale. Focus has been applied to the actual electrolyser stack components production (interconnects, electrolyte and cells), stack assembly and testing with cycling and long-term operations. The plan is to couple a NuScale SMR Module to DETAIL for R&D activities.

Figure VHTR 7. Systems Integration Laboratory at INL

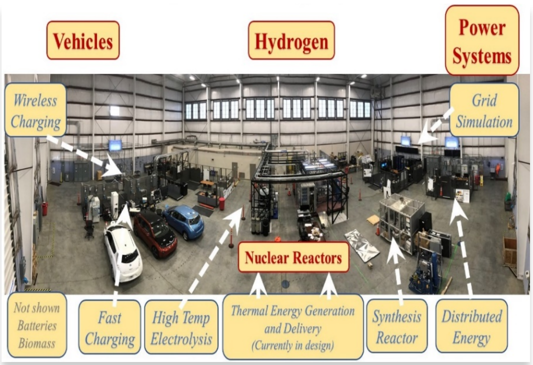


Figure VHTR 8. 25 kW HTSE Test Facility in DETAIL within the INL Energy Systems Laboratory

Computational methods validation and benchmarks Project

The Computational Methods Validation and Benchmarks (CMVB) project was restarted in 2014. From 2015 to 2018, eight meetings organized by the CMVB Provisional Project Management Board (pPMB) were held in turn in different participating countries. The main activities resulting from these meetings include discussion and confirmation of the research tasks in each work package (WP), review and approval of the draft project plan (PP) of which the final version is the indispensable annex of the project arrangement (PA), the discussion of some common topics and potential test facilities which will be the fundamental ressources of this project.

Table VHTR 1. **Work Package organization of the CMVB**

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

All the above-mentioned efforts were made to launch the signing process of the PA in 2019 focused on the review of the PA and discussion of how to carry out the PP. Through the pPMB meetings, the past, current, and new test facilities and projects have been identified, proposed and confirmed as fundamental resources to perform the development and assessment of codes and models covering HTR physics, thermal-hydraulics, CFD, fission products transport, etc.

In China, the demonstration project HTR-PM is under construction and commissioning. The installation of pressure vessels, steam generators, reactor internals and other important components have been finished. The standard design of the HTR-PM600, a commercial plant with an electricity power 600 MW, has been performed and reviewed by an independent nuclear engineering company. The engineering verification tests have been completed to support the HTR-PM project and such tests involve the main components of the HTR-PM, such as the helium circulator, the fuel handling system, the control rod driving system. Some benchmarking cases were defined and expected in the CMVB PP based on the HTR-PM future first criticality and low power physics tests. The HTR-10 was restarted and a temperature measurement experiment has been completed, whose purpose is to detect the temperatures inside the fuel elements. In addition, key operation parameters were monitored and one instance was the RCCS experimental data which will be used in the CMVB project to evaluate the capabilities of the system analysis tools to calculate the water-cooled RCCS behavior.

Figure VHTR 9. HTR-10 reactor coupled with RCCS

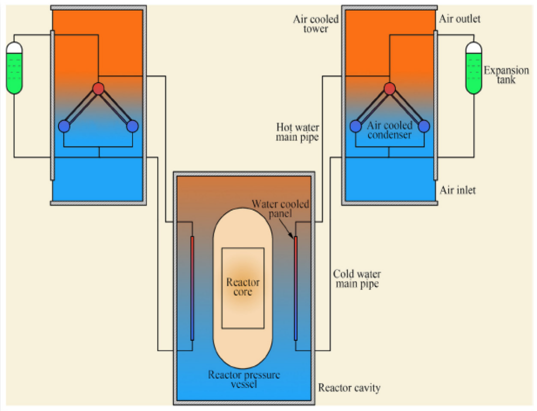
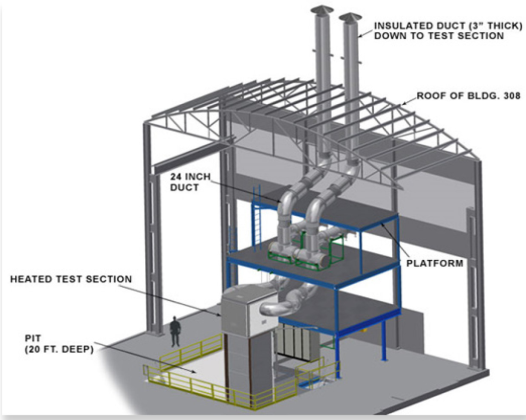


Figure VHTR 10. Natural Circulation Shutdown Heat Removal System at Argonne National Laboratory



In the United States, the advanced reactor technologies (ART) program is a strong support to CMVB project, since this program could provide data from the irradiation tests of fuel, graphite and also alloys. Regarding the design methods and validation, many concerned tests and benchmarks have been done through different facilities, e.g., the ANL Natural Circulation Shutdown Heat Removal Facility (NSTF, **Figure VHTR 10**), the High Temperature Test Facility (HTTF), Matched Index of Refraction Facility (MIR). Data from NSTF experiments is available for validation of air- and water-cooled RCCS models.

EU activities related to HTGR and CMVB include the GEMINI+ project which now is performing design iterations with thermal-hydraulics, neutronics and balance-of-plant calculations, previous Euratom Framework Program projects such as ARCHER, RAPHAEL, PUMA and NC2I-R, and also some past experiment projects such as NACOK, HELOKA, EVO, and HeFUS3. A new proposal for the Horizon 2020 Framework Program has been submitted: HYDRO-GeN-IV. If awarded funding Spring 2020, it will enable to continue and expand the work initiated in GEMINI+ after August 2020.

The VHTR R&D program in Korea aims at improving the VHTR key technologies in terms of the design codes development and assessment, and also high-temperature materials and component technologies. Some code validation work falling in the scope of the CMVB WPs has been completed, including scale-down standard fuel block tests, code-to-code comparisons for key design parameters.

JAEA is making a strong effort to restart the HTTR as early as possible. Based on design, construction and previous as well as future operation database of the HTTR, JAEA is developing and benchmarking various models and analysis methodologies and codes for reactor physics, thermal fluids, etc. The JAEA R&D in these areas is expected to support planning the CMVB co-operative activities such as benchmark activity using ATR irradiation data.

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Chair of the VHTR SSC and all Contributors