# 4 System reports

## Gas-cooled fast reactor

The gas-cooled fast reactor (GFR) system features a high-temperature helium-cooled fast spectrum reactor that can be part of a closed fuel cycle. The GFR, cooled with helium, is proposed as a longer-term alternative to liquid metal cooled fast reactors. This type of innovative nuclear system has several attractive features: the Helium is a single phase, chemically inert and transparent coolant. The high core outlet temperature, above 750°C and typically 800-850°C, is an added value of the GFR technology.

### Design objectives

High-outlet temperature (850°C) for high thermal efficiency and hydrogen production, and a direct cycle for compactness, are key reference objectives. Unit power will be considered in the range of 200 MWe (modularity), up to larger 1 500 MWe. Generation IV (Gen-IV) objectives for construction time and costs are therefore to be considered.

The objective of high fuel burn-up, together with actinide recycling, results in spent-fuel characteristics (isotopic composition) that are unattractive for handling. High burn-up is the final objective.

Consensus has been reached in the project to minimize feedstock usage with a self-sustaining cycle, which requires only depleted or reprocessed uranium feed. This would call for a self-generating core with a breeding gain near zero. So as not to penalize the long-term deployment of GFRs, and based on considerations regarding both the foreseen, available plutonium stockpiles (mainly derived from water reactors' irradiated fuel) and time for GFR fleet development, it is recommended that the initial Pu inventory in the GFR core not be much higher than 15 tonnes per GWe.

### Reference concept

The reference concept for the GFR is a 2 400 MWth plant having a breakeven core, operating with a core outlet temperature of 850°C that would enable an indirect, combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high-power density necessary for good neutron economics in a fast reactor core. The core is made up of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fuelled pins contained within a ceramic hextube. The favoured material for the pin clad and hextubes at the moment is silicon carbide fibre reinforced silicon carbide (SiCf/SiC). The entire primary circuit with three loops is contained within a secondary pressure boundary, the guard containment. The produced heat is converted into electricity in the indirect combined cycle, with three gas turbines and one steam turbine. The cycle efficiency is approximately 48%. A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle containing a helium-nitrogen mixture, which in turn drives a closed cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator, which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plants and so it represents an established technology, with the only difference in the case of the GFR being the use of a closed cycle gas turbine.

### The ALLEGRO gas-cooled fast reactor demonstrator project

The objectives of ALLEGRO are to demonstrate the viability and to qualify specific GFR technologies such as fuel, fuel elements, helium-related technologies and specific safety systems, in particular the decay heat removal function. It will also demonstrate that these features can be integrated successfully into a representative system. The demonstration of the GFR technology assumes that the basic features of the GFR commercial reactor can be tested in the 75 MWth ALLEGRO reactor.

The original design of ALLEGRO consists of two helium primary circuits, three decay heat removal (DHR) loops integrated into a pressurized cylindrical guard vessel (see Figure GFR-1). The two secondary gas circuits are connected to gas-air heat exchangers. The ALLEGRO reactor would serve not only as a demonstration reactor, hosting GFR technological experiments, but also as a test pad to:

- use the high-temperature coolant of the reactor in a heat exchanger to generate process heat for industrial applications;
- carry out research in a research facility which – thanks to the fast neutron spectrum – makes it attractive for fuel and materials development;
- test some of the special devices or other research work.

The 75 MWth reactor shall be operated with two different cores: the starting core, with uranium oxide (UOX) or mixed oxide (MOX) fuel in stainless steel claddings will serve as a driving core for six experimental fuel assemblies containing the advanced carbide (ceramic) fuel. The second core will consist solely of the ceramic fuel, enabling operation of ALLEGRO at the high target temperature.

Central European members of the European Union – the Czech Republic, Hungary and the Slovak Republic – are traditionally prominent users of

nuclear energy. They intend to use nuclear energy over the long term. In addition to lifetime extensions of their nuclear units, each country has decided to build new units in the future.

Four nuclear research institutes and companies in the Visegrad-Four region (ÚJV Řež, a.s., Czech Republic, MTA EK, Hungary, NCBJ, Poland, VUJE, a.s., Slovak Republic) have decided to start joint preparations aiming at the construction and operation of the ALLEGRO demonstrator for the Gen-IV gas-cooled fast reactor (GFR) concept, based on a memorandum of understanding signed in 2010. The French Alternative Energies and Atomic Energy Commission (CEA), as the promoter of the GFR concept since 2000, supports these joint preparations, bringing its knowledge and its experience to building and operating experimental reactors, and in particular fast reactors.

In order to study safety and design issues, as well as medium- and long-term governance and financial issues, in July 2013 the four aforementioned organizations created a legal entity, the V4G4 Centre of Excellence, which performed the preparatory work needed to launch the ALLEGRO Project. The V4G4 Centre of Excellence is also in charge of international representation for this project. As a result of the preparatory work, it was revealed that during earlier work certain safety and design issues remained unsolved and for several aspects a new ALLEGRO design had to be elaborated. In 2015, therefore, when the ALLEGRO Project was launched, a detailed technical program was established with a new time schedule.

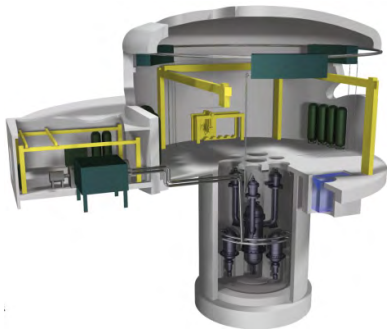
### Fuel cycle and fuel

Fuel development efforts must be conducted in close relation with reactor design efforts so that both the fuel meets core design requirements and the core operates within fuel limits.

Technology breakthroughs are needed to develop innovative fuel forms, which:

- preserve the most desirable properties of thermal gas-cooled reactors, particularly to withstand temperatures in accidental situations (for the high-temperature reactor [HTR] up to 1 600°C, and to be confirmed through design and safety studies for the GFR);
- resist fast neutron-induced damage, to provide excellent confinement of the fission products;
- accommodate increased heavy metal content.

Figure GFR-1. ALLEGRO Systems



Alternative geometries of the fuel and innovative claddings should be investigated. The path to GFR fuel development is intricately bound to the ALLEGRO project, and an iterative approach will be necessary. The ALLEGRO start-up core will consider MOX or UOX fuel pellets deployed in conventional, steel-clad tubing, necessitating its own design and licensing program. An iterative step to a full ceramic demonstration core in ALLEGRO is an essential part of the RD&D required for the GFR.

The candidate fuel types already identified are:

- UOX and MOX pellets in 15-15 titanium (Ti) tubular steel cladding for the ALLEGRO start-up core;
- pin/pellet type fuels characterized by solid solution fuel pellets in a ceramic cladding material, whereby such pins, and eventually assemblies, would be introduced into the ALLEGRO start-up core and eventually into the demonstration.

A significant amount of knowledge is available on MOX fuel, but more needs to be available to establish the ALLEGRO start-up core.

Data on potential ceramic (particularly, SiCf/SiC) and refractory alloys for cladding materials are inconsistent. These materials need to be adapted in order to cope with the different loads (e.g. thermal gradients, interaction fuel-barrier, dynamic loads), which means that their composition and microstructure need specific developments. The main goal of high-temperature experiments is to investigate the behavior of 15-15 titanium (15-15Ti) alloy in high-temperature helium. Beyond the testing of small tube samples, ballooning and burst experiments will be performed at high temperature. Mechanical testing will be carried out to investigate the change of the load-bearing capacity of cladding after high-temperature treatments. The cladding microstructure will be examined by scanning electron microscopy (SEM) and metallography.

The development of a qualification procedure for start-up fuel will include specification of the steps for MOX/UOX fuel with 15-15Ti cladding, including irradiation in reactors with fast spectrum and post-irradiation examination of irradiated fuel samples.

Numerical model development for the start-up core will focus on the extension of FUROM code with fast reactor fuel properties and models in order to simulate fuel behavior for the ALLEGRO start-up core. Validation of the code should be based on sodium-cooled fast reactor fuel histories.

Testing of SiC claddings in high-temperature helium will be carried out to track potential changes. Mechanical testing and the examination of the microstructure with SEM and metallography is planned with the samples after high-temperature treatment.

The ion-irradiation effect on SiC composites will be investigated in order to evaluate the importance of the significant volume change observed for hydrogen (Hi)-Nicalon type-S fibre and C fibre coating. High-dose ion irradiation will be carried out with various temperature ranges, including

GFR operating temperatures for SiC composites. The high-dose irradiation effect on SiC composites will be examined.

The investigation of high-temperature oxidation behavior of SiC composites is important for severe accident studies. Various kinds of silicon carbide composites and monolithic SiC ceramics will be oxidized up to 1 500°C. Surface modification of SiC will be carried out based on the understanding of oxidation behavior.

The following topics will be analyzed in the short term:

Design of the ALLEGRO reactor core:

- UOX core feasibility study using ERANOS, MCNP, SERPENT;
- determination of total reactor power and power density to satisfy both safety limits and irradiation capabilities;
- formulation of selection criteria to choose an optimal core.

Development of fuel behavior codes for ALLEGRO fuel:

- collection of material data for fast reactor materials;
- derivation of the reactor's physical parameters needed for the FUROM code;
- implementation of fast reactor material data in the FUROM code.

Tasks related to ALLEGRO fuel qualification and specification:

- ALLEGRO fuel-related acceptance criteria;
- review of fuel candidates for the first core of ALLEGRO;
- selection of the components of optimal ceramic fuel for ALLEGRO;
- development of the ceramic fuel qualification procedure.

Tasks related to research on fuel materials:

- review of SiCf/SiC cladding materials;
- testing UOX/MOX fuel cladding in high-temperature He;
- mechanical testing of UOX/MOX fuel cladding.

## The SafeG project

The SafeG project has received funding from the Euratom Horizon 2020 program NFRP-2019-2020-06, under grant agreement no. 945041. The global objective of the SafeG project is to further develop GFR technology and strengthen its safety. The project will support the development of nuclear, low-$CO_2$ electricity and the industrial process heat generation technology through the following main objectives:

- to strengthen the safety of the GFR demonstrator ALLEGRO;
- to review the GFR reference options in materials and technologies;
- to adapt GFR safety to changing needs in electricity production worldwide, with increased and decentralized portions of nuclear electricity, by studying various fuel cycles and their suitability from safety and proliferation resistance points of view;
- to bring in students and young professionals, boosting interest in GFR research;
- to deepen the collaboration with international, non-EU research teams, and relevant European and international bodies.

The main task of the project is to respond to the safety issues of the GFR concept and to introduce the key safety systems of the ALLEGRO reactor. An important part of the design is to acquire new experimental data using recent research from experimental devices and special computational programs to carry out safety analyzes and the study of relevant physical phenomena. The SafeG project takes into account the most urgent questions and open issues concerning the GFR technology and the ALLEGRO demonstrator. To answer these questions, the SafeG project is divided into six technical work packages and one co-ordination work package.

The ambitions of the SafeG project can be divided into four tasks:

1) Completing the ALLEGRO demonstrator safety concept:

- core optimization from the neutronic, thermo-hydraulic and thermo-mechanic points of view;
- design of diversified reactor control and reactor shutdown systems;
- passive decay heat removal strategy completed with the design of fully passive systems for the decay heat removal tested on the experimental helium loop.

2) Upgrading the ALLEGRO demonstrator design and GFR concept through innovative materials and technologies, such as fuel cladding based on SiC composition, and construction materials capable of withstanding the extreme temperatures used for the primary system and safety-related systems.

3) Linking national research activities and creating an integrated platform that aims to share knowledge, and results achieved, as well as to co-ordinate activities, and spread new ideas and findings throughout the scientific society worldwide.

4) Expanding cooperation between Europe and Japan on GFR research through the sharing of knowledge about advanced high-temperature resistant materials for fuel rod claddings and other primary system components.

**Branislav Hatala**
Chair of the GFR SSC, with contributions from GFR members

## Lead-cooled fast reactor

The Generation IV International Forum (GIF) has identified the lead-cooled fast reactor (LFR) as a technology with great potential to meet the needs of both remote sites and central power stations, fulfilling the four main goals of GIF. In the technology evaluations of the *Generation IV Technology Roadmap* (2002), and its update in 2014, the LFR system was ranked at the top in terms of sustainability (i.e. a closed fuel cycle can be easily achieved), and in proliferation resistance and physical protection. It was also assessed as good in relation to safety and economics. Safety was considered to be enhanced by the choice of a relatively inert coolant. This section highlights the main collaborative achievements of the LFR-provisional System Steering Committee (pSSC) to date. It also presents the status of the development of LFRs in GIF member countries and entities.

### Main characteristics of the system

LFR concepts include three reference systems: 1) a large system rated at 600 MWe (e.g. the European lead fast reactor [ELFR EU]), intended for central station power generation; 2) a 300 MWe system of intermediate size (e.g. BREST-OD-300, Russia); and 3) a small, transportable system of 10-100 MWe size (e.g. the small secure transportable autonomous reactor [SSTAR], United States) that features a very long core life (see Figure LFR-1). The expected secondary cycle efficiency of each LFR system is at or above 42%. GIF-LFR systems thus cover the full range of power levels: small, intermediate and large sizes. Important synergies exist among the different reference systems, with the co-ordination of the efforts carried out by participating countries one of the key elements of LFR development. The typical design parameters of GIF-LFR systems are briefly summarized in Table LFR-1.

### R&D objectives

The LFR System Research Plan (SRP) developed within GIF is based on the use of molten lead as the reference coolant and lead-bismuth eutectic (LBE) as the back-up option. Given the R&D needs for fuel, materials and corrosion-erosion control, the LFR system is expected to require a two-step industrial deployment: in a first step, reactors operating at relatively modest primary coolant temperatures and power densities would be deployed by 2030; and higher performance reactors by 2040. Following the reformulation of the GIF-LFR-pSSC in 2012, the SRP has been completely revised. The report is presently intended for internal use by the LFR-pSSC, but it will ultimately be used as a guideline for the definition of project arrangements once the decision of a transition from the present memorandum of understanding (MoU) status to a system arrangement organization is engaged.

**Table LFR-1: Key design parameters of the GIF-LFR concepts**

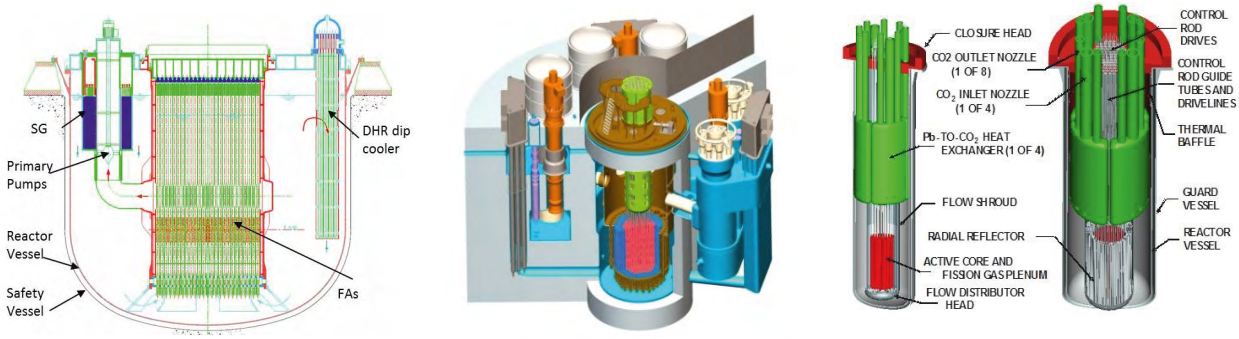
| Parameters | ELFR | BREST | SSTAR |
|---|---|---|---|
| Core power (MWt) | 1 500 | 700 | 45 |
| Electrical power (MWe) | 600 | 300 | 20 |
| Primary system type | Pool | Pool | Pool |
| Core inlet T (°C) | 400 | 420 | 420 |
| Core outlet T (°C) | 480 | 540 | 567 |
| Secondary cycle | Superheated steam | Superheated steam | Supercritical $CO_2$ |
| Net efficiency (%) | 42 | 42 | 44 |
| Turbine inlet pressure (bar) | 180 | 180 | 200 |
| Feed temperature (ºC) | 335 | 340 | 402 |
| Turbine inlet temperature (ºC) | 450 | 505 | 553 |

### Main activities and outcomes

The collaborative activities of the LFR-pSSC over the last eight years were centred on top level reports for GIF. After the issuance of the "LFR White Paper on Safety" in collaboration with the GIF Risk and Safety Working Group (RSWG) in 2014, the pSSC has been very active on the following main lines of work:

- *LFR system safety assessment* – The RSWG asked SSC chairs to develop a report on their systems in order to analyze them systematically, assess the safety level and identify further safety-related R&D needs. The LFR assessment report was prepared in collaboration with the RSWG and was published in June 2020. It is presently available.[1]

**Figure LFR-1. GIF-LFR reference systems: ELFR, BREST and SSTAR**




Alemberti, A. et al. (2018).

1. www.gen-4.org/gif/upload/docs/application/pdf/2020-06/gif_lfr_ssa_june_2020_2020-06-09_17-26-41_202.pdf.


Figure LFR-2(a). Experimental investigation of the dynamics of fuel assembly mock-ups


Figure LFR-2(b). Testing of loading and unloading of fuel assembly mock-ups

- "LFR Proliferation Resistance and Physical Protection (PRPP) White Paper" – In 2018, the PRPP Working Group realized the need for a substantial revision of the PRPP white paper for the six GIF systems. The modifications related to the LFR paper were mainly related to the addition of the BREST system (Russia) and refinements to information available on the SSTAR (US) and ELFR (Euratom) systems. The paper has been developed with the PRPPWG, and the final version was sent to the Experts Group for final approval at the end of 2020. A public issue on the GIF website is expected in 2021.
- *LFR safety design criteria (SDC)* – Development of the LFR-SDC was based on the previously-developed SFR SDC report. It was later realized, however, that the IAEA SSR-2/1 (the reference document for SFR SDC development) did not require many of the features identified for the SFR to be adapted for the LFR (note that the IAEA SSR-2/1 refers substantially to LWR technology). After a first set of comments, received at the end of 2016, the LFR-pSSC updated the report following the IAEA revision of SSR-2/1 and the document was re-circulated for comments within the RSWG. The final comments from the RSWG were received in December 2020, and the report is presently expected to receive Experts Group approval 1st semester 2021.

The LFR-pSSC has also been working actively with the GIF Task Force on R&D Infrastructure and has contributed to the questionnaire provided by the Advanced Manufacturing and Materials Engineering (AMME) Task Force. These activities led to participation in the February 2020 workshop organized at the NEA in Paris.

Interaction between LFR-pSSC and the Working Group on Safety of Advanced Reactors (WGSAR) started through the participation of LFR representatives in the October 2020 meeting of the WGSAR. LFR-SDC and LFR-pSSC activities were presented, and it was agreed to transmit the LFR-SDC report to the WGSAR.

## Main activities in Russia

The innovative fast reactor with lead coolant, BREST-OD-300, is being developed as a pilot demonstration prototype of basic commercial reactors with a closed nuclear fuel cycle for the future nuclear power industry.

The lead coolant was chosen on the basis of the favourable characteristics of its properties, namely: 1) in combination with dense (U-Pu)N fuel, it allows for complete breeding of fissile materials in core, maintaining a constant small reactivity margin and thus preventing any prompt-neutron excursion with an uncontrolled power increase (equipment failures or personnel errors); 2) it enables the possibility to avoid the void reactivity effect due to the high boiling point and high density of lead; 3) it prevents coolant losses from the circuit in the postulated event of vessel damage because of the high melting/solidification points of the coolant and the use of an integral layout of the reactor; 4) it provides high heat capacity of the coolant circuit, which decreases the probability of fuel damage; 5) it capitalizes on its high density and albedo properties for flattening the fuel assembly (FA) power distribution; and 6) it facilitates larger time lags of the transient processes in the circuit, which makes it possible to lower the requirements for the safety systems' rate of response.

Mixed uranium-plutonium nitride fuel is used in the core design, and low-swelling ferritic-martensitic steel is used as the fuel cladding. Fuel elements are placed in shroud-less hexagonal fuel assemblies. Currently, the technology of dense nitride fuel is implemented on pilot production lines. These technological processes are being improved, and industrial fuel production is being created for the fabrication of fuel for the BREST-OD-300 reactor. For the initial stage of BREST-OD-300 operation, a reduced value of the maximum fuel burn-up is planned – 6% heavy atoms (h.a.); then, a gradual justified transition to the design target values of burn-up up to 9-10% h.a. is envisaged. The performance of the nitride fuel is confirmed by the results of radiation tests in the BN-600 power reactor and BOR-60 research reactor. In total, more than 1 000 fuel elements were irradiated. For one experimental FA with fuel elements of the BREST type, burn-up of more than 9% h.a. and a damage dose of more than 100 dpa were achieved. All semi-finished products from EP823 steel were put into production, and all properties have been obtained that ensure the operability of the fuel elements up to 6% h.a. burn-up: short-term, long-term, under irradiation, and in

a lead coolant medium. FA mock-ups (all types) and reflector blocks were produced in industrial conditions, and manufacturing technology is fully developed. All the necessary experimental studies were carried out for these mock-ups: spills in water and lead, vibration tests, and tests for bending stiffness and strength. The loading and unloading of FA mock-ups from the core were experimentally tested, as shown in Figure LFR-2.

The main objective of the reactor vessel, when performing safety functions, is to exclude the loss of coolant. Estimated probability of coolant leakage from the reactor circuit is about 9·10-10 1/year. With this event, only a partial loss of the coolant is possible (non-critical, acceptable); at the same time, the primary circuit does not break, and the possibility of natural circulation of the coolant in the circuit remains. A wide range of experimental work was carried out on the metal-concrete vessel – on various concrete samples, mock-ups of the vessel itself and on its elements. Properties of high-temperature concrete were obtained experimentally at temperatures of 400-700°C, and under irradiation. The chemical inertness of the lead coolant in relation to concrete was shown. Sufficient knowledge has been collected to start manufacturing the reactor vessel of the BREST-OD-300 reactor.

The steam generator consists of monometallic tubes, corrosion resistant in water and lead, with no welds along the entire length. The steam generator has twisted heat exchange parts. To date, a comprehensive justification of the elements and processes occurring in the steam generator has been carried out. It can be noted in particular that the absence of induced failure in case of one tube rupture was experimentally demonstrated. Experiments have shown that neighbouring tubes are not damaged, which is a very important achievement for safety. Another important point that has been confirmed through calculations alone, but will be tested in an experiment at the Fast Critical Facility (BFS), is that with the postulated passage of steam bubbles through the core (i.e. when the tubes of the steam generator break) there is no burst of positive reactivity. The value of the void-vapour reactivity effect is close to zero. It should be noted that because of the presence of a free surface level in the reactor vessel, the probability of steam entering the core during depressurization of the steam generator tubes is extremely low.

To justify the pumps, a wide range of studies were carried out. At the initial stages, the flow parts were optimized, and the shapes of the impeller blades were selected. By means of calculations and experiments on scale models, the head characteristics of the main circulation pump (MCP) were obtained. Positive results were obtained on the life tests of the bearing (justification of the life of 100%), and full-scale bench MCP testing in lead is being created, see Figure LFR-3. As for other equipment and systems, the prototype of the control and protection system (CPS) actuator


Figure LFR-3: Testing of the friction pair of the MCP lower radial bearing as part of the model block

passed acceptance and life tests; endurance testing of coolant quality system components is being conducted; and automated monitoring and control system has been developed. Prototypes of equipment are undergoing final testing, and thus they are ready for implementation in the reactor during construction. A large set of experimental studies were carried out concerning the assessment of the yield of fission and activation products from a lead coolant. This knowledge is important when performing a radiation safety analysis for various temperature levels, typical of normal operation (500°C) and accidents with significant lead heating (680°C). Based on the data obtained in the experiment, the requirements are determined for the composition of the initial lead for the primary coolant. In the course of the optimization performed, the composition of impurities was minimized, while maintaining an acceptable cost of lead.

The safety analysis has shown that under the most conservative scenario of inserting the full reactivity margin, the maximum fuel temperature will reach 1 640°C, and fuel cladding 1 260°C (for a few seconds). There is no fuel melting, and the lead coolant does not boil. The implementation of such a scenario is feasible with a probability of 2.9·10-9 1/year. For another conservative scenario, complete blackout of the power unit with failure of mechanical shutdown systems (ATWS), the level of the attainable fuel-element cladding temperature is lower than in the first scenario and does not exceed 903°C. Long-term cooling is carried out using a passive emergency cooling system of the reactor, with natural circulation of lead in the primary circuit. For both scenarios, the main requirement has been met - there is no need to protect the population.

The BREST-OD-300 reactor is being created as one of the most important components of the pilot demonstration power complex operating in a closed fuel cycle, together with modules for fabrication, re-fabrication and reprocessing of spent fuel. In

addition to operation (power generation), the most important task is the implementation of the R&D program at the reactor. Various studies and life tests are planned to be carried out on components, equipment, and irradiation experiments in a lead coolant and in a fast neutron spectrum. This will form an essential scientific basis for research. The BREST-OD-300-unit design received a positive conclusion from the Glavgosexpertiza, and licensing by Rostekhnadzor is being completed. In 2020, an examination was undertaken by the Russian Academy of Sciences, which gave a positive conclusion and recommended the construction of the power unit, confirming that the design corresponds to the modern level of science and technology, as well as to scientific ideas about the problems of existing nuclear energy and ways to solve them.

## Main activities in Japan

Theoretical studies of fast reactors using lead-bismuth eutectic as a coolant have been performed in Japan since the beginning of LFR activities. One of the advantages of lead or lead-bismuth coolant is the better neutron economy in the core due to the hard neutron spectrum and the small neutron leakage. These features make it easy to realize the once-through fuel cycle, fast reactor concepts. The concepts of the breed-and-burn reactors and CANDLE burning reactors with lead-bismuth coolant have been studied at the Tokyo Institute of Technology (TIT). One of the important issues related to these concepts is maintaining the integrity of fuel elements in very high burn-up conditions. Research has confirmed the possibility to solve the problem through the introduction of the melt-refining process, based on metallic fuel. The study also considered the use of plutonium from LWR spent fuel for the start-up core to achieve effective use of plutonium. A new fuel shuffling scheme was proposed as the output of the studies. It has proven that it is possible to achieve a stationary wave equilibrium condition by implementing a fuel shuffling scheme concept.

Chemical compatibility of lead (Pb) and Pb alloys with various materials in different situations is being studied at the Tokyo Institute of Technology (TIT). Figure LFR-4 shows these different situations where chemical compatibility presents important issues to be addressed. The structural materials that exhibit corrosion resistance are essential to expand the operating life and to improve the reliability of Pb coolant systems. Excellent corrosion resistance of ferritic iron-chromium-aluminium (FeCrAl) alloys (Kanthal® APMT and FeCrAlZr-ODS) in liquid Pb and Pb alloys was confirmed. $\alpha$-Al2O3 formed on the FeCrAl alloys from the pre-oxidation treatment in air atmosphere at 1 273K for 10 hours.

This oxide layer functions as a protective layer, which can significantly improve the thermo-dynamic stability and the chemical compatibility of the alloys. The metallurgical analysis with scanning transmission electron microscope (STEM) on the

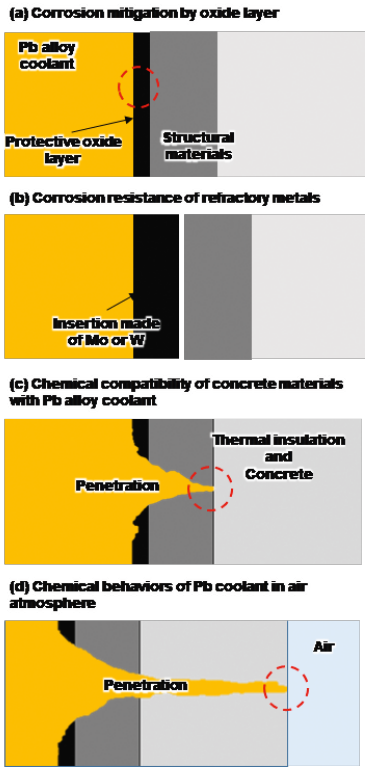



Figure LFR-4: Corrosion issues in various situations

protective oxide layer after the immersion in liquid Pb alloy was performed in the collaborative project for the development of FeCrAl zirconium (Zr)-oxide dispersion-strengthened alloys. Experimental studies on the mass transfer of metal and non-metal impurities in a lead-bismuth coolant system have been performed. The diffusion behaviors of metal impurities such as Fe and nickel (Ni) in lead-bismuth were investigated by means of long capillary experiment and molecular dynamic (MD) simulation. The diffusion coefficients of these elements were newly obtained for various temperatures. Refractory metals such as molybdenum (Mo) and tungsten (W) are also corrosion resistant in liquid Pb alloys. Therefore, the insertion and the lamination of plates made of the refractory metals are proposed to suppress the corrosion of structural materials as shown in Figure LFR-4 (b).

Concrete materials must work as an important barrier, which suppresses the Pb coolant leakage and the loss-of-coolant accident, especially for the pool-type Pb-based reactors, as shown in Figure LFR-4 (c). The chemical compatibility of some cement and concrete materials having various water/cement (W/C) ratios is being investigated by means of their immersion in liquid Pb alloys. Corrosion-resistant concrete materials are also going to be developed.

The thermo-dynamic behavior of liquid Pb alloys in the air atmosphere, shown in Figure LFR-4 (d) were investigated by means of the static oxidation experiments for Pb alloys with various chemical compositions. The results of the static oxidation tests for Pb-bismuth (Bi) alloys indicate that the chemical reactivity of Pb and Pb alloys in air at high temperatures was quite mild. In the oxidation procedure of the Pb alloys, Pb was depleted from

the alloys due to the preferential formation of lead oxide (PbO) in air at 773K. Bi was not involved in this oxidation procedure. Pb-Bi oxide and Bi2O3 were formed only after the enrichment of Bi in the alloys due to Pb depletion.

The chemical control of liquid Pb alloy coolant was improved with high-performance solid electrolyte oxygen sensors, which can provide a better response in high-temperature conditions. The excellent performance of the sensor with shorter stabilization time is achieved by reducing the gas volume in the reference compartment of the oxygen sensor.

### Main activities in Euratom

The main activities in Europe related to liquid metal technologies are centred on two main projects: 1) the development of the Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA) research infrastructure, which is being carried out by SCK-CEN in Mol (Belgium) and is aiming at the demonstration of an accelerator-driven system (ADS) technology and supporting the development of Gen-IV systems; and 2) preliminary activities for the construction of an LFR demonstrator in Romania, or the ALFRED project. These two projects are supported through dedicated Euratom initiatives.

Concerning the development of MYRRHA, the project roadmap for the implementation of Lead Bismuth Eutectic (LBE) technology for an Accelerator Driven System (ADS) was defined at the end of 2018. In September 2018, the Belgium federal government also decided to allocate EUR 558 million to the implementation of MYRRHA during the period 2019-2038 as follows:

- EUR 287 million for phase 1: building of MINERVA (linear accelerator up to 100 MeV, 4 mA + Proton Target Facility [PTF]) during the period of 2019-2026;
- EUR 115 million for phases 2 and 3: phase 2 involves the design and R&D of the second section of accelerator up to 600 MeV, while phase 3 involves further design and licensing activities related to the LBE-cooled sub-critical reactor, both to be carried out in the period of 2019-2026;
- EUR 156 million for the operating expenses of MINERVA for the period of 2027-2038.

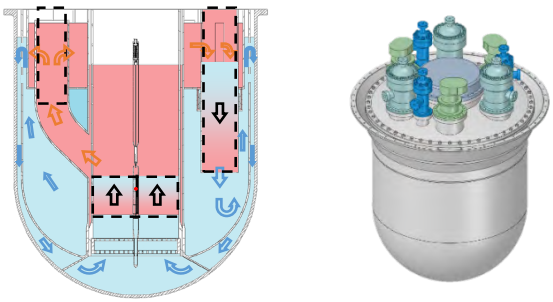
The MYRRHA project is currently being implemented, and is also supported by numerous Euratom-funded collaborative projects.

Regarding the ALFRED project, the main development activities are conducted by Ansaldo Energia (Italy), the National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA, Italy) and the Institute for Nuclear Research (RATEN ICN, Romania), which are the signatories of the Fostering ALfred CONstruction (FALCON) Consortium Agreement. The FALCON Consortium Agreement was renewed at the end of 2018 for an additional phase of activities. One of the main aims of the consortium is to involve a number of additional European partners in the ALFRED project, through the signature of memoranda of agreement (MOA) expanding throughout Europe as much as possible the interest in the development of lead technology. By the end of 2020, the FALCON Consortium enlarged the community and extended the ALFRED project, with the signature of several MOAs with partners willing to provide in-kind support to technical activities related to ALFRED development.

An important event took place in June 2019 in Pitesti (Romania), where the European Commission (EC) co-organized the Fission Safety (FISA) 2019 and EURADWASTE'19 conferences with the Ministry of Research and Innovation of Romania and RATEN ICN, under the auspices of the Romanian presidency of the EU and in collaboration with the IAEA. The conference gathered 500 stakeholders, presenting progress and key achievements of around 90 projects, which are or have been carried out as part of the 7th and Horizon 2020 Euratom Research and Training Framework Programmes (FPs). In that framework, a side workshop organized by the FALCON Consortium on ALFRED infrastructure attracted a very large number of participants, stimulating discussions on the status of heavy liquid metal technology R&D activities and the roadmap for the LFR demonstrator in Europe.

The FALCON Consortium took important steps during the period of 2018-2020. First, a main step of the design review was completed, and a new system configuration was defined, consisting of three steam generators (SGs) (using benefits from the new configuration it was designed single-wall bayonet tubes), three dedicated dip coolers for the second decay heat removal (DHR) system, and three primary pumps (PP). The definition of the placement of other dedicated systems and components on the reactor roof is presently under way. Additional design changes have been carried out in the primary system configuration, involving an improved definition of hot and cold pools and a special arrangement of the primary flow path to completely eliminate the thermal stratification on the vessel for both forced and natural circulation conditions. The new configuration and its main characteristics are presented in Figure LFR-5.

**Figure LFR-5: ALFRED primary system flow-path configuration (left) and external view (right)**



Frignani, M., Alemberti, A., and Tarantino, M. (2019).

Alemberti, A., et al. (2020).

The DHR-1 system consists of isolation condensers connected to steam generators (three units) and equipped with an anti-freezing system, which is being investigated in the PIACE Euratom collaborative project (cf. below). A similar system is being used for the DHR-2 system connected to a dip cooler, which uses double wall bayonet tubes.

In 2019, the FALCON Consortium also took an important decision regarding ALFRED operation and licensing. Namely, it was decided to approach both operation and licensing using a stepwise approach to better face the known limits concerning materials corrosion and consequent qualification in a representative environment.

The idea is to follow a staged approach, characterized in principle by a constant primary mass flow and increasing power levels, which results in an increase of the maximum lead coolant temperature as follows:

- 1st stage: low temperature
  - proven technology, proven materials, oxygen control, low temperatures;
  - hot FA for in-core qualification of dedicated coating for cladding;
- 2nd stage: medium temperature
  - need for FA replacement, same SGs and PPs;
  - hot FA for in-core qualification at higher temperatures;
- 3rd stage: high temperature
  - replacement of main components for improved performances;
  - representative of first-of-a-kind (FOAK) conditions for the LFR deployment.

In this way, each stage is consequently used to qualify (through the hot fuel assembly conditions) the operation that will be carried out in the following stage. Each stage of the operation will need to be separately licensed, but, using the confidence gained in the previous stage(s), the licensing process is expected to be a continuous process able to bring the technological solutions to the higher temperatures needed for industrial deployment. Table LFR-3 below provides the main parameters of the envisaged staged approach.

**Table LFR-3: ALFRED staged approach – Main parameters**

| Normal operation – full power | Stage 1 | Stage 2 | Stage 3 |
|---|---|---|---|
| Thermal power (MW) | 100 | 200 | 300 |
| Core inlet temperature (°C) | 390 | 400 | 400 |
| Core outlet temperature (°C) | 430 | 480 | 520 |
| Pump head (MPa) | 0.15 | 0.15 | 0.15 |

For the interested reader, further information on the ALFRED design and staged approach can be found in the papers presented at the FISA 2019 Conference.

During the year 2019, the Romanian government also awarded RATEN ICN (Romanian research laboratory) a funding of EUR 2.5 million for a project dedicated to "Preparatory activities for ALFRED infrastructure development in Romania". The project ended successfully in November 2020, with a final workshop organized by RATEN ICN.

RATEN ICN also responded to a call for proposals from the Romanian government with a project called "ALFRED - Step 1, experimental research support infrastructure: ATHENA (lead pool-type experimental facility) and ChemLab (lead chemistry laboratory)". The project proposal was awarded funding in June 2020, and the competitive bids for design and construction of related facilities are to be published in 2021 for a total budget of about EUR 20 million.

Finally, with regard to Euratom R&D projects, the main collaborative projects already in place related to LFR technology and Gen-IV fuels are: 1) GEMMA, dedicated to material R&D and qualification for Gen-IV LFRs; 2) M4F, covering material R&D for Gen-IV and fusion applications; 3) INSPYRE, dedicated to fuel R&D for fast reactors; and 4) the LFR SMR INERI project, which involves the European Commission Joint Research Centre (JRC) and the US Department of Energy. This Euratom project portfolio has recently been complemented by three new projects: PIACE (started in 2019), as well as PATRICIA and PASCAL (both of which commenced in 2020). The PIACE project is dedicated to demonstrating the prevention of lead freezing in LFRs through passive safety provisions. The project had its kick-off meeting at the ENEA research laboratory (Brasimone) and is presently underway, with some experimental results expected to be available in 2021. The PATRICIA project provides further supporting R&D for the implementation of MYRRHA and related pre-licensing efforts, while the PASCAL project involves R&D on selected safety aspects for heavy liquid metal systems, specifically focusing on the extension of experimental evidence to demonstrate the increased resilience of MYRRHA and ALFRED to severe accidents. Lastly, the SESAME Euratom collaborative project was concluded in 2019 with the final workshop and release of a book dedicated to the thermal-hydraulic aspects of liquid metals.

## Main activities in Korea

The Korean government joined the GIF-LFR pSSC by signing the MoU at NEA in November 2015. LFR R&D progress has been made mainly through university programs during the past 20 years, since the first study in 1996 at Seoul National University (SNU). Since 2019, the primary momentum of LFR development has been transferred to the Ulsan National Institute of Science and Technology (UNIST). The Korean LFR Programme, however, remains unchanged with two main objectives:

- a new electricity generation and hydrogen production unit development requirement to match the needs of economically competitive

distributed power and hydrogen sources for both developed countries and developing nations that need massive and inexpensive electric power with an adequate margin against worst case scenarios encompassing internal and external events;

- a technology development requirement for sustainable power generation using energy produced during nuclear waste transmutation.

To meet the first goal, the Korean government has been funding international collaborative R&D to further upgrade the ubiquitous, rugged, accident-forgiving, non-proliferating, and ultra-lasting sustainer (URANUS) design into a micro-reactor design called MicroURANUS, which can be applied to maritime applications and has a 40-year lifespan without refuelling. A pre-conceptual design has been completed with all the top-tier design requirements met, by following GIF-LFR methodologies including LFR-SDC. Results using PIRT analysis (Phenomena Identification and Ranking Table) were reviewed by the LFR-pSSC members through a video conference.

For the second goal, since 1996 the Korean first LFR-based burner, the proliferation-resistant environment-friendly accident-tolerant continual-energy economical reactor (PEACER), has been transmuting long-lived waste in spent nuclear fuel into short-lived low-intermediate level waste. In 2008, the Korean Ministry of Science and Technology selected the sodium-cooled fast reactor (SFR) as the technology for long-lived waste transmutation. Since then, LFR R&D for transmutation in Korea has turned its direction towards an accelerator-driven Th-based transmutation system designated as the thorium optimized radioisotope incineration arena (TORIA), with the leadership of Sungkyunkwan University and Seoul National University, as well as UNIST.

### Main activities in the United States

Work on LFR concepts and technology in the United States has been carried out since 1997. In addition to reactor design efforts, these activities have included work on lead corrosion/material compatibility and thermal-hydraulic testing at a number of organizations and laboratories, and the development and testing of advanced materials suitable for use in lead or LBE environments. While current LFR activities in the United States are limited, past and ongoing efforts at national laboratories, universities and the industrial sector demonstrate continued interest in LFR technology.

With regard to design concepts, of particular relevance is the past development of the small, secure transportable autonomous reactor (SSTAR), carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other organizations over an extended period of time. SSTAR is a SMR that can supply 20 MWe/45 MWth with a reactor system that is transportable. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life (15-30 years) with cassette refuelling; and an innovative supercritical $CO_2$ (S-$CO_2$) Brayton cycle power conversion system. Although work on SSTAR is no longer active, SSTAR continues to be represented as one of the reference designs of the GIF-LFR pSSC.

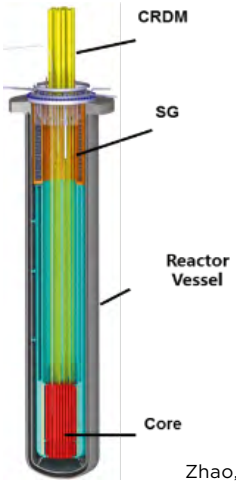
Additional university-related design activities include past work at the University of California on the Encapsulated Nuclear Heat Source (ENHS) and more recently in several projects sponsored by the US Department of Energy under Nuclear Energy University Project (NEUP) funding. These include the following ongoing efforts:

- An effort led by the Massachusetts Institute of Technology (MIT) in the area of corrosion/irradiation testing in lead and lead-bismuth eutectic. The project seeks to investigate the "Radiation Decelerated Corrosion Hypothesis", relying on simultaneous exposure tests (rather than separate long-term corrosion and neutron irradiation), followed by microstructural characterization, mechanical testing and comparison to enable rapid down selection of potential alloy candidates and directly assess how irradiation affects corrosion.
- An effort at the University of Pittsburgh to develop a versatile liquid, lead testing facility and test material corrosion behavior and ultrasound imaging technology in liquid lead.

In the industrial sector, ongoing LFR reactor initiatives include the continuing initiative of the Westinghouse Corporation to develop a new advanced LFR system (Westinghouse-LFR) and the efforts of Hydromine, Inc. to continue development of the 200 MWe LFR identified as LFR-AS-200 (i.e. amphora shaped), as well as several micro-reactor spin-off concepts identified as the LFR-TX series (where T refers to transportable, and X is a variable identifying power options ranging from 5 to 60 MWe). It should be noted that Westinghouse is engaged with several universities and national laboratories to pursue technology developments related to the LFR, including an experimental investigation of radioisotope retention capability of liquid lead, as well as efforts to use the versatile test reactor for LFR-related investigations. Additionally, Westinghouse is currently engaged in the phase 2 effort of the UK Government's Department for Business, Energy and Industrial Strategy's (BEIS) Advanced Modular Reactor (AMR) Feasibility and Development project to demonstrate LFR components and accelerate the development of HT materials, advanced manufacturing technologies and modular construction strategies for the LFR.

### Main activities in China

The Chinese government has provided continuous national support to develop lead-based reactor technology since 1986, by the Chinese Academy of Sciences (CAS), the Ministry of Science




Zhao, Z. et al. (2019).

Figure LFR-6. Overall view of the CLEAR-M reactor

IAEA LMFNS database

Figure LFR-7. Lead-based engineering validation reactor, CLEAR-S

and Technology (MOST), the National Science Foundation (NSF), the 13th Five-Year plan, etc. After more than 30 years of research on lead-based reactors, the China LEAd-based reactor (CLEAR), proposed by the Institute of Nuclear Energy Safety Technology (INEST)/FDS team, was selected as the reference reactor for the ADS project, as well as for the technology development of the Gen-IV lead-cooled fast reactor. Activities related to the CLEAR reactor design, reactor safety assessment, design and analysis software development and the lead-bismuth experimental loop, as well as R&D on key technologies and components, are being carried out.

The CLEAR-M project, with a typical concept of a 10 MW-grade CLEAR-M10, aiming at the construction of a small modular energy supply system, has been launched (see Figure LFR-6). The main purpose of the project is to provide electricity as a flexible power system for wide application, such as island, remote districts or industrial parks. In addition, two small LFR projects have been supported by MOST to explore innovative LFR concept designs.

For the ADS system, several concepts and related technologies are under assessment. For example, the detailed conceptual design of CLEAR-I with the final goal of minor-actinide (MA) transmutation, which has a dual operation capability of subcritical and critical modes, has been completed. An innovative ADS concept system, such as the advanced external neutron source-driven, travelling-wave reactor, CLEAR-A, was proposed for energy production. The CiADS project, conducted in collaboration with the CAS and other industrial organizations, to build a 10 MWth subcritical experimental LBE-cooled reactor coupled with an accelerator was approved, and the preliminary engineering design is underway.

In order to support the CLEAR projects, as well as to validate and test the key components and integrated operating technology of the lead-based reactor, a multi-functional lead-bismuth experiment loop platform (i.e. KYLIN-II) was built and has operated for more than 30 000 h. Various tests have been conducted, including corrosion tests, LBE thermal-hydraulic experiments and components prototype proof tests. In addition, three integrated test facilities have been built and have started commissioning since 2017, including the lead-based engineering validation reactor CLEAR-S (See Figure LFR-6), the lead-based zero power critical/subcritical reactor CLEAR-0, coupled with the HINEG neutron generator for reactor nuclear design validation, as well as the lead-based virtual reactor, CLEAR-V. A loss-of-flow benchmarking test, based on the pool-type CLEAR-S facility, is being prepared.

In recent years, other organizations have started paying more attention to LFR development. For example, the China General Nuclear Power Group (CGN), China National Nuclear Corporation (CNNC), State Power Investment Corporation (SPIC) and several universities such as Xi'an Jiaotong University (XJUT), and the University of Sciences and Technology of China (USTC) have been carrying out LFR conceptual design and related R&D, including materials tests, thermal-hydraulic analysis and safety analysis. INEST was appointed by MOST as the leading organization to co-ordinate the participation of domestic organizations in GIF activities. A domestic LFR joint working group will therefore be established.

To promote the engineering and commercial application of China lead-based reactor projects, the China Industry Innovation Alliance of Lead-based Reactor (CIIALER) and the International Co-operative Alliance for Small Lead-based Fast Reactors (CASLER), both led by the INEST/FDS team, were established and supported by over 100 companies, and the construction of a related industrial park has begun.

**Alessandro Alemberti**
Chair of the LFR SSC, with contributions from LFR members

# Molten salt reactor

## Main characteristics of the system

Molten salt reactor (MSR) concepts have been studied since the early 1950s, but with only one test reactor operated at the Oak Ridge National Laboratory (ORNL, United States) in the 1960s. For the past 15 years, there has been a renewal of interest in this reactor technology, in particular for its acknowledged inherent reactor safety and its flexibility.

MSR uses molten salts as fuel and/or coolant. When a fluoride salt is the coolant alone, the concept is called a fluoride salt-cooled high-temperature reactor (FHR). Today, in the GIF pSSC MSR, most, if not to say all, the studied concepts are actual MSRs with liquid fuel.

The MSR is a concept and not a technology. Indeed, the MSR generic name covers thermal and fast reactors, operated with a U/Pu or a Th/233U fuel cycle, or as trans-uranium (TRU) burners, with a fluoride or a fluoride carrier salt. An illustration of the most studied concept is provided in Figure MSR-1 below.

Depending on the fuel cycle, MSRs can re-use fissile and fertile materials from LWRs, or they can use uranium, or burn plutonium or minor actinides. They have an increased power conversion efficiency (the fission directly occurs in the carrier salt, which transfers its heat to the coolant salt in the heat exchangers). MSRs are operated under low pressure, slightly above atmospheric pressure. They can be deployed as large power reactors or as small modular reactors (SMRs). Their deployment is today limited by technological challenges, such as high temperatures, structural materials, and corrosion.

The MSR pSSC today includes seven full members (Australia, Canada, Euratom, France, Russia, Switzerland and the United States) and three observers (China, Japan and Korea) and is moving towards a system arrangement. The mission of the MSR pSSC is to support the development of future nuclear energy concepts that have the potential to provide significant safety and economic improvements over existing reactor concepts.

## R&D objectives

The common objective of MSR projects is to propose a conceptual design with the best system configuration – resulting from physical, chemical and material studies – for the reactor core, the reprocessing unit and wastes conditioning. Mastering of the technically challenging MSR technology will require concerted, long-term international R&D efforts, namely:

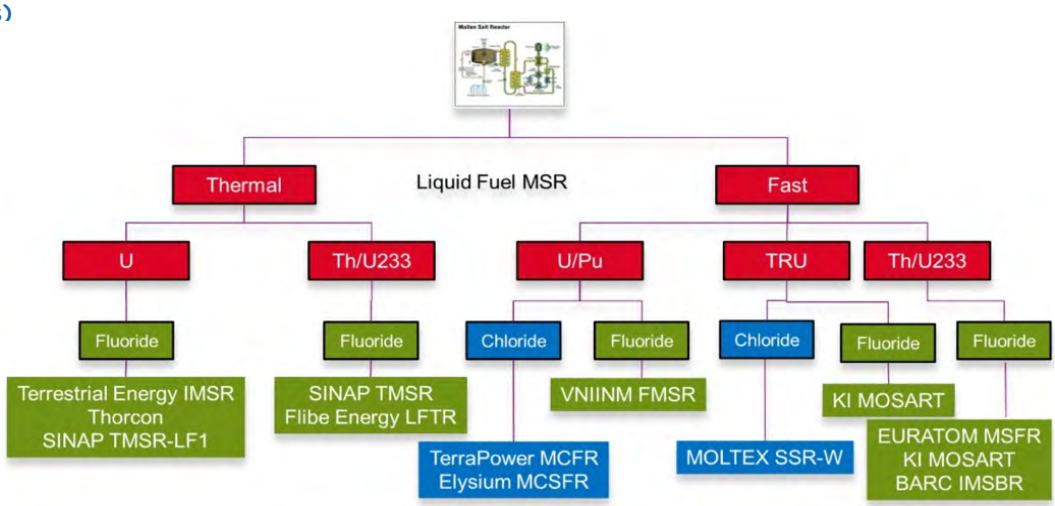
- the study of salt chemical and thermo-dynamic properties;
- for the system design, development of advanced neutronic and thermal-hydraulic coupling models;
- the study of materials compatibility with molten salt;
- salt Redox control technologies to master corrosion of the primary fuel circuit and other components;
- development of efficient techniques for the extraction of gaseous fission products from the coolant through He bubbling;
- for salt reprocessing, reductive extraction tests (actinide-lanthanide separation);
- development of a safety approach dedicated to liquid-fuelled reactors.

## Main activities and outcomes

### *MSR pSSC activity*

In 2019, the key activity was the preparation of the system arrangements (SAs) with the definition of three potential projects arrangements, which would allow the community to contribute broadly. These PAs are therefore quite transversal, and not concept-dependent, but they can support the development of any concept (see Figure MSR-2).

**Figure MSR-1. The most studied MSR concepts, with key players (research & technology organization or vendors)**



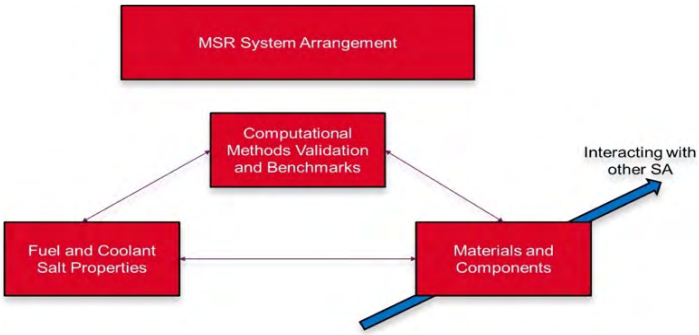



Figure MSR-2. Foreseen structure of the MSR SA, including three PAs

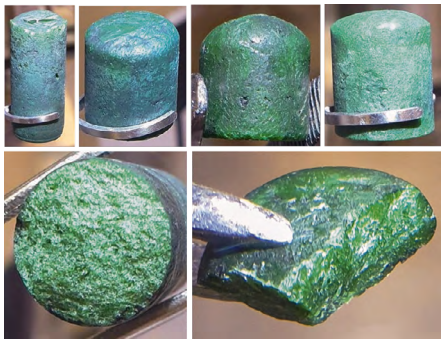


Figure MSR-3. Fuel salt ingots fabricated at JRC Karlsruhe for the SALIENT-03 irradiation experiment

They address the salt behavior, materials properties and system integration. The SA should enter into force in 2021.

## Euratom

### *European SAMOSAFER project:*

On 1 October 2019, the new Severe Accident Modeling and Safety Assessment for Fluid-fuel Energy Reactors (SAMOSAFER) project started with the aim of developing new simulation models and tools, and designing new safety barriers for the MSR. The goal of this new project is to develop and demonstrate new safety barriers for more controlled behavior of MSRs in severe accidents, based on new simulation models and tools validated with experiments. The overall objective is to ensure that the MSR can comply with all expected regulations in 30 years' time. After successful completion of this project, the simulation models and tools can be used by the nuclear industry, and the innovative safety barriers can be implemented in new MSR designs. This will lead to increased safety margins in future Gen-IV MSRs to ensure that they comply with the latest and future safety standards. SAMOSAFER is co-ordinated by TU Delft and will run until 2023.

In 2020, SAMOSAFER partners focused, inter alia, on the continuation of the design of the MSR, the distribution of radionuclides in the fuel treatment unit, the risk identification (list of post-irradiation examinations [PIEs]) in the Fuel Treatment µUnit (FTU) as input for the safety analysis using the failure modes and effects analysis (FMEA) method, the development of new algorithms and the design and construction of experimental setups for validation of these, as well as the generation of physico-chemical data of various molten salts to extend the JRC Karlsruhe database.

In the Netherlands, the Salient-01 irradiations in the Petten high-flux reactor were finalized. The samples are currently being investigated in the framework of SAMOSAFER in the NRG and JRC Karlsruhe laboratories. Follow-up irradiations (Salient-03), containing five fuel salt samples encapsulated in nickel-based alloys, are under preparation.

### *JRC Karlsruhe:*

Among experimental studies on basic thermo-chemical properties, JRC is extensively involved in the synthesis and fabrication of the fluoride fuel salt for the planned irradiation experiment, SALIENT-03, in the high-flux reactor (HFR) at Petten via this major collaboration with NRG. Four batches of fuel salts were synthesized, having the following compositions (mol. %): 757LiF-18.7ThF4-6.0UF4-0.3PuF3 (28.02 g), 757LiF-18.7ThF4-5.7UF4-0.3UF3-0.3PuF3 (15.02 g), 757LiF-18.6ThF4-6.0UF4-0.4CrF3-0.3PuF3 (7.01 g) and 75LiF-23.0ThF4-2.0UF4-0.1UF3 (50.12 g). The first three salts will be irradiated, while the latter will serve for out-of-pile electro-chemical tests at NRG. The end members, 7LiF, ThF4, UF4 and PuF3, were synthesized and their purity verified using methods developed and published previously by JRC Karlsruhe. The method for synthesis of UF3, based on reduction of UF4 through gaseous hydrogen at 800°C, was developed specifically for the project. The fuels for irradiation were prepared in a form of solid ingots of the quenched salts exactly fitting into the irradiation capsules. The salt mixtures were melted in liners made of the same materials as the irradiation capsules (Hastelloy-N and GH3535) under a flow of pure HF gas, which was found necessary for further purification. All obtained ingots have the required shape and mass and an excellent purity, as proven by the combination of XRD, DSC, ICP-MS and oxygen analysis through Knudsen effusion mass spectrometry (KEMS) methods. The selected ingots are shown in Figure MSR-3, including a cross-section demonstrating sufficient homogeneity and purity.

With the increasing demand for reliable measurements of density for MSR fuels, significant effort has been expended in 2020 to design and test the novel densitometer at JRC Karlsruhe. The selected method of measurement is based on the Archimedean buoyancy effect, and entire set up was designed such that it fits the current glove boxes that are kept under protective argon atmosphere, and which are licensed to handle nuclear materials. Furthermore, the spherical bob that is immersed into the molten salt during measurement is made of nickel to avoid corrosions at high temperatures. The first high-temperature

measurements were undertaken using the LiCl-KCl eutectic salt, and were successfully tested up to 650°C. The results obtained are shown in Figure MSR-4 below, which indicates high reliability of the method. The experimental set up is designed such that it provides data for both fluoride and chloride based MSR fuels. The figure also shows the roadmap of the densitometer development, indicating very rapid development. In early 2021, further tests are planned with a few more fluoride and chloride inactive salt mixtures, with successive installation of the final design in a hot glove box for measurement of actinide containing fuels.

Further development of the Joint Research Centre Molten Salt Database (JRCMSD) continued with the addition of BeF2-ZrF4, KF-ThF4 binary systems. With increasing worldwide demand for the use of chloride salts as fuels for certain MSR designs, the database is being extended to relevant chloride systems.

***Research Centre Řež:***

In addition to the activities carried out within the SAMOSAFER project, the Research Centre Řež, together with other Czech companies, has also continued the national MSR technology development program. The program is focused on the theoretical and experimental development of selected areas of MSR technology. In 2020, the main task of the program was to fully finalize all preparatory phases and stages for the measurement of neutronic characteristics of the molten fluoride salt, FLIBE, in the working temperature range of MSRs using the method of "hot inserted FLIBE zone" in the LR-0 experimental reactor. For this purpose, the core of the LR-0 reactor was completely reconstructed so that a set of the "hot inserted FLIBE zones" could be placed at its centre. Before the end of 2020, a set of inserted zones was completed, the location in the LR-0 reactor was verified and all so-called "cold" non-active experiments were successfully completed. The inserted zone occupies the space of seven fuel assemblies in the middle of the LR-0 reactor core. Hot experiments-measurements of the neutronic characteristics of the FLIBE melt in the range of MSR operating temperatures (550-750°C) will start in the first half of 2021.

In addition to hot inserted zone experiments, the R&D program in the area of materials research, development and verification of components and equipment for fluoride melt media, and the study of electro-chemical separation methods, along with methods of fused salt volatilization suitable for online MSR liquid fuel reprocessing technology, also continued.

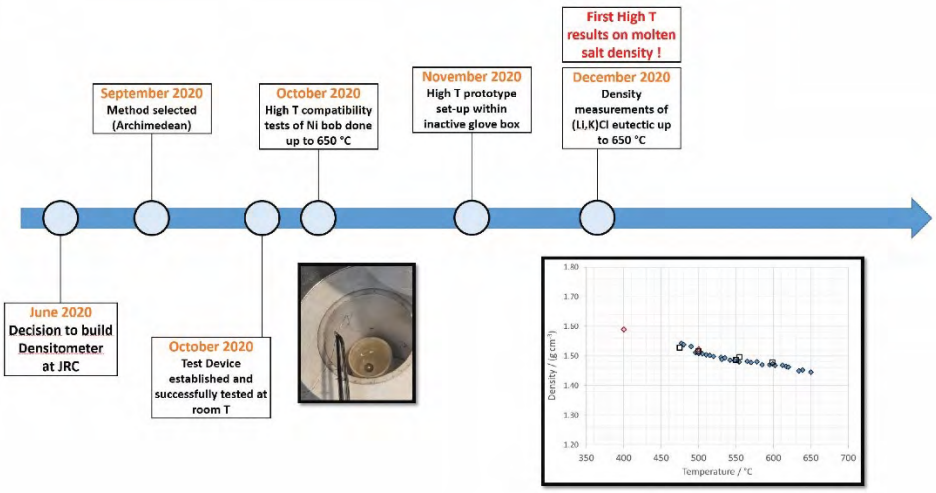
## France

Since the beginning of 2020, the CEA has been carrying out its a research program oriented around defining a sketch of a molten salt reactor. Three options are being considered, all in a fast spectrum and in molten chloride: isogenerator, Pu burner, and minor-actinide transmutor.

Studies cover the reactor system (i.e. neutronics, materials, components) and the associated fuel cycle (i.e. salt behavior, corrosion, salt polishing). Multi-physics and chemistry modelling and simulation are also part of the scope. This program, involving the three research institutes of the CEA (IRESNE at Cadarache, ISEC at Marcoule and ISAS at Saclay[1]), is carried out in collaboration with the CNRS (Grenoble, Orsay), with the support of Orano. JRC Karlsruhe is also contributing.

In 2020, work focused on the definition of the plutonium burner option with the study of different

**Figure MSR-4. Roadmap of densitometer development at JRC for measurement of MSR fuels**




Note: the graph first shows the successful results on the (li,K)Cl eutectic salt – blue: data obtained during heating; red: data obtained during successive cooling; black: data obtained during repeated measurement.

1. IRESNE : Institut de recherche sur les systèmes nucléaires pour la production d'énergie bas carbone; ISEC : Institut pour les Sciences et Technologies pour une Économie Circulaire des Énergies bas carbone, ISAS : Institut des Sciences Appliquées et de la Simulation pour les énergies bas carbone.

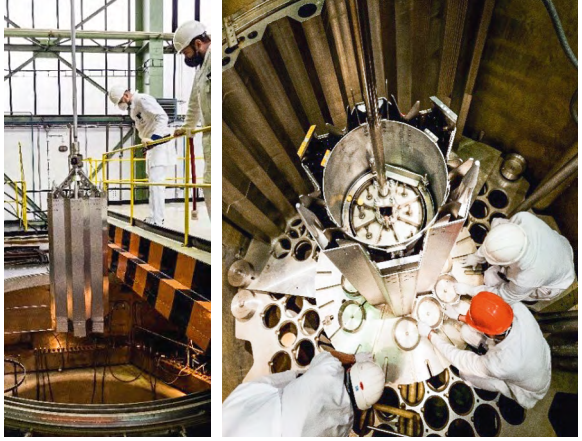
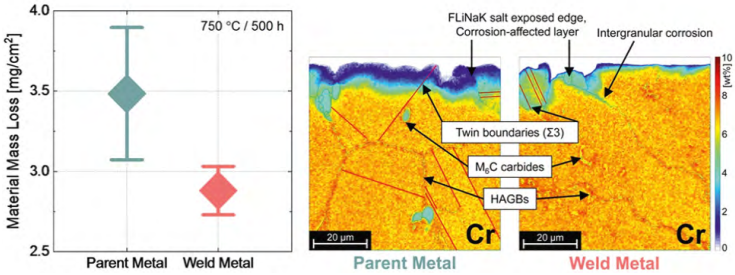


Figure MSR-5. Test insertion of the complete FLIBE insertion zone assembly into LR-0 and verification its location in the centre of the reactor core




Source: Danon, A. E. et al. (2020)

Figure MSR-6. Corrosion rate and microstructural differences in a GH3535 weldment a corrosion tested in FLiNaK molten salt at 750°C

concepts for the reactor, and for different salt composition. In parallel, a tool calculating the evolution of the composition of the salt under irradiation was built (MOSARELA). This tool will contribute to the definition of the reactor operation conditions and the salt treatment strategy.

## Australia

The molten salt technology is becoming increasingly important in a wide range of low-carbon energy production and storage systems. Successful deployment requires the development and qualification of materials and components capable of withstanding the challenging operation conditions. Australia's Nuclear Science and Technology Organisation (ANSTO) continues to collaborate with GIF partners to understand corrosion in FLiNaK of candidate stainless steels and nickel-based alloys, as well as how advanced manufacturing techniques may be used to decrease their time to deployment in advanced reactors.

Highlights include a recent investigation of the corrosion performance of the welded Ni-Mo-Cr (GH3535) alloy, where it was shown that the weld, while having a very similar composition, had a superior corrosion resistance to the parent metal. The difference is attributed to the differences in microstructure and in particular, a significantly lower density of high-angle grain boundaries (HAGBs) in the weld metal and the large M6C carbides present in the parent metal.

## Russia

During the year 2020, Rosatom continued to provide support to preliminary design development for: 1) test 10-megawatt thermal (MWt) Lithium, Beryllium, Actinides/Fluorine MSR with homogeneous core; and 2) its fuel salt clean-up unit at the site of the Mining and Chemical Combine (Zheleznogorsk) in order to demonstrate the control of the reactor and fuel salt management with different long-lived actinide loadings, drain-out, shut down, etc.

Two main objectives of the MSR project for the period up to year 2024 include:

- development and demonstration of key technological solutions for a MSR with circulating fuel for the transmutation of long-lived actinides;
- development of a preliminary design for the test MSR and required materials to obtain a license for its placement.

During the year 2020, the main R&D efforts were focused on the following issues:

- optimization of neutron and the thermal-hydraulic characteristics of the core and fuel circuit;
- development of analytical methods to measure the impurities in the fuel salt and intermediate coolant;
- development of an advanced high nickel alloy with enhanced corrosion and radiation resistance properties for the fuel circuit;
- construction of experimental units for materials tests with fuel and coolant salts in laboratory and reactor conditions.

## United States

The US government continues to foster US MSR industry development through a number of cost-sharing R&D programs. The US Department of Energy (DOE) in particular is supporting both university and national laboratory activities at a limited scale to overcome the remaining technical hurdles to MSR deployment.

In 2020, MIT received a DOE award to build an in-reactor molten salt test loop.[2] This facility will provide researchers with an understanding of by-products in an MSR and test instrumentation. It will also serve as a prototype for other university loop studies and DOE test reactors. The research will use both a non-irradiated and irradiated flowing salt loop to examine the behavior of fission by-products, especially ones that do not stay dissolved in the salt. These by-products in particular will deposit themselves on surfaces

2. https://nrl.mit.edu/announcements/2020/nrl-receives-four-doe-project-awards.

within the loop, or separate from the liquid salt as gaseous by-products. Understanding how these by-products affect the loops will give valuable insights into the MSR design. Natura Resources LLC also granted Abilene Christian University (ACU) USD 21.5 million[3] over the next three years as part of a USD 30.5 million effort to design and license a research reactor in collaboration with three major universities: Georgia Institute of Technology, Texas A&M University, and The University of Texas at Austin. Launched in spring 2019, the consortium's goal is to design, license and commission the first university-based molten salt research reactor, which ACU will host and own. The deal represents the largest sponsored research agreement in the university's history. Recent national laboratory activities related to MSRs include developing a molten salt thermal properties database based on molten salt thermo-physical and thermo-chemical property measurement and engineering evaluation of off-gas system technology.

In parallel, the Nuclear Regulatory Commission (NRC) is making progress on the process of modernizing its licensing requirements to better reflect the safety characteristics of advanced reactors. MSR features and phenomena are being incorporated into accident progression evaluation tools. Progress in being made in developing methodologies for qualifying liquid fuel salts, as well as non-power reactor review guidance.

Vendors such as Kairos Power and Terrestrial Energy USA have also filed multiple topical reports and white papers to the NRC in 2020 in order to support the licensing process of their MSR concepts.

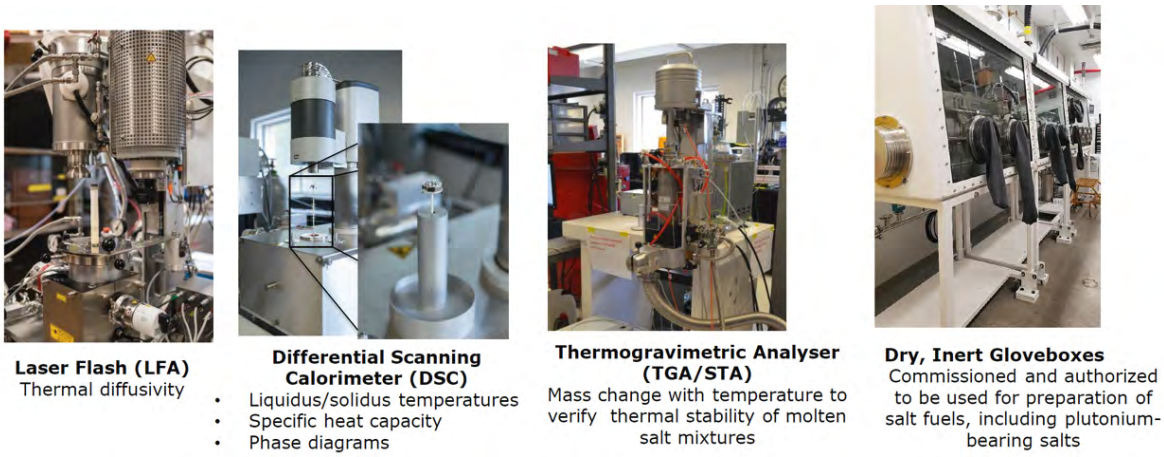
### Canada

In 2020, Canadian Nuclear Laboratories (CNL) continued to develop expertise and capabilities in support of SMRs. The CNL executed multiple projects for SMR vendors under a new cost-sharing R&D program called the Canadian Nuclear Research Initiative (CNRI). The CNRI program was established by the CNL to accelerate the deployment of SMRs in Canada, enabling research and development and connecting the SMR industry with facilities and expertise within Canada's national nuclear laboratories. Among the many benefits of the program, participants are able to optimize resources, share technical knowledge and gain access to CNL expertise so as to help advance the commercialization of SMR technologies. Among the first to take part in this new program, three MSR vendors worked with the CNL on a diverse program of work, including electro-chemical separation methods, tritium management, reactor physics, thermal-hydraulics and safeguards studies.

Under the auspices of the Canadian Federal Nuclear Science and Technology Programme, the CNL continued to develop molten salt capabilities across a wide range of areas including:

- development of actinide molten salt fuel synthesis using no gaseous reagents;
- fission product retention in molten salt experiments; evaluation of passive cooling during a station blackout with experiments on coupled natural circulation heat transfer between water and molten salt loops, and evaluation of molten salt plug melting in accident conditions;
- corrosion loop development for measuring the corrosion of structural materials;
- modelling and simulation of MSR designs, including evaluation of the codes for an advanced reactor coupled transient simulation toolset against ORNL MSRE: Physics (SERPENT, Rattlesnake); TH (RELAP5-3D, ARIANT); CFD (STAR-CCM+) and atomistic simulations to predict molten salt properties.

**Figure MSR-7. Sample encapsulation and measurement technique development at CNL**




3. https://techtransfercentral.com/2020/08/04/abilene-christian-u-with-21-million-from-natura-resources-to-build-research-reactor-as-part-of-consortium/.

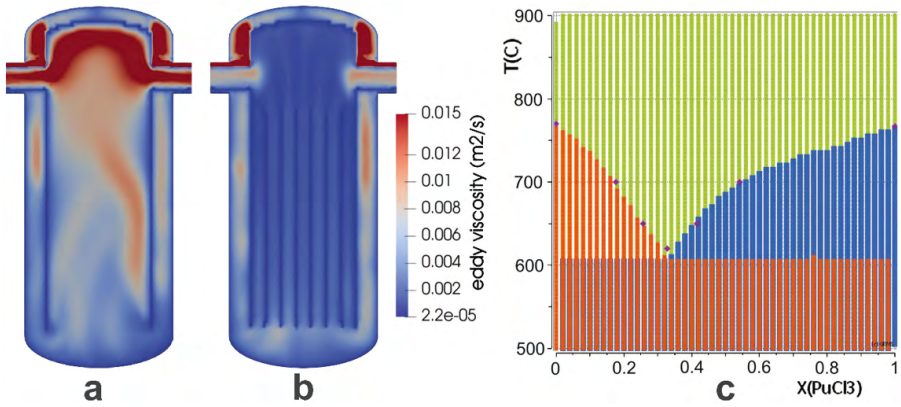



**Figure MSR-8. Conceptual layout of the MCFR core vessel a) without and b) with baffles. Phase diagram of the ThCl-PuCl3 system c).**

Finally, significant efforts have continued in further developing nuclear qualified measurement techniques for the thermo-physical properties of molten salts.

Terrestrial Energy: Terrestrial Energy Inc. (TEI) is a privately funded reactor vendor developing the integral MSR (IMSR). A 195 MWe, graphite-moderated design employing standard assay low-enriched uranium (LEU) for near- term deployment in the late 2020's. TEI is sourcing R&D services worldwide in order to develop its IMSR concept. In 2020, NRG was contracted for a major multi-year irradiation program in the Petten reactor of multiple commercial graphite grades suitable for MSR use (sub-micron pore diameter). The first specimens were introduced in the reactors in October 2020 with similar protocols to the INNOGRAPH study for gas-cooled graphite candidates. Argonne National Laboratory will provide services for various fuel salt property verifications with the high standards needed for the regulator process.

In terms of commercialization milestones achieved in 2020, the IMSR concept has been short-listed by Ontario Power Generation as one of three technologies developed for potential commercialization, after extensive commercial due diligence. TEI has also been awarded CAD 20 million from the Canadian federal government's Strategic Innovation Fund.

## Switzerland

Swiss MSR research continued in 2020 at the Paul Scherrer Institute (PSI) with a major aim of monitoring the technology, education of new experts, and development of knowledge and simulation capabilities in: fuel cycle, system behavior, and thermo-dynamics areas of molten salt research. A big part of PSI activities represent contributions to the EU Horizon 2020 project, SAMOSAFER, and therefore these belong to the EU progress report.

In the area of fuel cycle assessment, the PSI continued to develop a dedicated benchmark for the respective simulation tools with partners from the SAMOSAFER project. The breed-and-burn fuel cycle in the molten chloride fast reactor (MCFR) was further assessed, and the Serpent 2-based procedure, EQL0D, was applied to several additional fuel cycle configurations. The in-depth knowledge of this reactor and fuel cycle type, together with past Swiss research in this area, was used for preparation of an MCFR chapter for Elsevier's *Encyclopedia of Nuclear Energy* (to be published in 2021). For the same encyclopedia, a series of three chapters dedicated to self-sustained breeding in advanced reactors was prepared based on the extensive knowledge of actinide behavior during irradiation in numerous MSR concepts and other advanced reactors.

The system behavior study with Open-FOAM based solver, ATARI, continued in 2020 covering three different aspects: 1) simulation of the SAMOSAFER project reference concept MSFR; 2) assessment of freezing phenomena in printed circuit heat exchangers; and 3) conceptual design of an MCFR core with tube-in-tube and baffles options. The impact of baffles on eddy viscosity is illustrated in Figure MSR-8 a and b.

The thermo-dynamics simulation of molten salts continued with the GEMS TM code, focusing on further refinement of the database and major fluoride and chloride salt components. The adjusted database was applied to phase diagram calculations (see Figure MSR-8 c) and to the cGEMS code for the estimation of evaporation behavior.

## China

In 2020, the Shanghai Institute of Applied Physics and the Chinese Academy of Sciences (SINAP-CAS) have been steadily promoting the related work of the thorium molten salt reactor (TMSR). This 2 MWth molten salt test reactor (TMSR-LF1) was approved with a construction license. Construction of the plant structure for the experimental reactor was started and completed in 2020, and the equipment was delivered for installation. The application for an operation license (including the final safety analysis report [FSAR] and other relevant attachments) was submitted, and the first stage review was completed. Key equipment has entered the final

stage of manufacturing and will be delivered to the project site in succession. At present, installation of the main equipment has started.

A number of experiments have been completed on the scaled experimental device (TMSR-SF0), including: the thermal-hydraulic performance experiment of key equipment and the steady-state and transient characteristic experiment on the salt system. The experimental program will continue in 2021.

The conceptual design of the flowsheet for TMSR fuels has been finished, and validation of the flowsheet for thorium fuel reprocessing is in progress. Fundamental studies on the structure and reaction of actinide and fission product fluorides in molten salt have been carried out.

Significant progress has been achieved on MSR material research. It was proven that GH3535 alloys maintain good creep properties in FLiNaK molten salts. Experiments on molten salt erosion of nuclear graphite at elevated temperatures were carried out, which provided data support for the further application of nuclear graphite in MSRs. In addition, the neutron-radiation-induced defect evolution of nickel-based alloy has been studied using the newly developed rate theory method.

### Japan

In Japan, the International Thorium Molten Salt Forum (ITMSF) was established in 2008 for the basic study of MSR technology, such as conceptual designs and safety analysis for MSR-FUJI. The ITMSF has been an observer in the GIF-MSR System Steering Committee from the beginning of the committee. In 2010, Thorium Tech Solution Inc. (TTS) was established for the business application of the MSR-FUJI. In addition to these activities, several universities have been carrying out basic studies in the recent decade.

The Japanese government began supporting the development of MSR technology in 2019, and continued to do so until the beginning of 2021. Three MSR companies were selected: two (the Thermal Transient Test Facility for Structures [TTS] and MOSTECH) are promoting MSRs with fluoride salt moderated by graphite, and one is promoting a fast spectrum MSR with chloride salt, on which universities (Tokyo Institute of Technology [TIT], Fukui, Doshisha) and the Central Research Institute of Electric Power Industry (CRIEPI) are working together.


Figure MSR-9. FLiNaK molten salt loop at the National Institute for Fusion Science

MOSTECH is planning to construct a molten salt loop at Kyushu University and is also preparing freeze valve tests for a fusion blanket loop system of molten salt, together with Kyushu University and the University of Electro Communications (UEC), as shown below. This loop system (Oroshhi-2: described by A. Sagara et al. in Fusion Science and Technology in 2015), was built in the National Institute for Fusion Science (NIFS) using FLiNaK.

**Stéphane Bourg**
Chair of the MSR SSC, with contributions from MSR members

## Supercritical water reactor

### Main characteristics of the system

The supercritical water-cooled reactor (SCWR) is a high-temperature, high-pressure, water-cooled reactor that operates above the thermo-dynamic critical point (374°C, 22.1 megapascals [MPa]) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: pressure-vessel concepts proposed first by Japan, and more recently by a Euratom partnership with China, and a pressure-tube concept proposed by Canada. Apart from the specifics of the core design, these concepts have many similar features. The R&D needs for each reactor type are therefore common, enabling collaborative research to be pursued.

The main advantage of the SCWR is improved economics because of the high thermo-dynamic efficiency and the potential for plant simplification. Improvements in the areas of safety, sustainability, and proliferation resistance and physical protection are also possible, and are being pursued by considering several design options using thermal and fast spectra, including the use of advanced fuel cycles.

There are currently three Project Management Boards (PMBs) within the SCWR system: for system integration and assessment (provisional), materials and chemistry (MC), and thermal-hydraulics and safety (THS). The extension of the project arrangements to thermal-hydraulics and safety, as well as to MC, with project plans covering 2021-2025, are in progress and have been discussed during the steering committee and Project Management Board meetings held in early December 2020.

### R&D objectives

The following critical-path R&D projects have been identified in the SCWR system research plan (SRP):

- System integration and assessment: definition of a reference design, based on the pressure tube and pressure-vessel concepts, which meets Gen-IV requirements (sustainability, economics, safe and reliable performance, proliferation resistance).
- Thermal-hydraulics and safety: gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed to validate thermal-hydraulic codes. The design-basis accidents for a SCWR have some similarities with conventional water reactors, but the difference in thermal-hydraulic behavior and large changes in fluid properties around the critical point compared to water at lower temperatures and pressures need to be better understood.
- MC: qualification of key materials for use in in-core and out-core components of both pressure tube and pressure-vessel designs. Selection of a reference water chemistry will be sought to minimize materials degradation and corrosion product transport, and will be based on materials compatibility and on an understanding of water radiolysis.

### Main activities and outcomes

#### *System integration and assessment*

Because of the COVID-19 pandemic, and priority work on the THS and MC PMBs, no GIF SCWR activities were undertaken in this field in 2020.

The Joint European Canadian Chinese Development of Small Modular Reactor Technology (ECC-SMART) project was launched in September 2020. ECC-SMART is a collaborative project covering most GIF SCWR research fields. ECC-SMART is oriented towards assessing the feasibility and identification of safety features of an intrinsically and passively safe SMR cooled by supercritical water (SCW-SMR). The project takes into account specific knowledge gaps related to the future licensing process and the implementation of this technology. The main objectives of the project are to define the design requirements for the future SCW-SMR technology, to develop a pre-licensing study and guidelines for demonstration of safety in the further development stages of the SCW-SMR concept, including the methodologies and tools to be used, and to identify key obstacles for future SMR licensing and a strategy for this process.

To reach these objectives, specific technical knowledge gaps were defined and will be assessed to better achieve future licensing and implementation of the SCW-SMR technology, particularly in terms of the behavior and irradiation of materials in the SCW environment, and validation of the codes and design of the reactor core, evaluated through simulations and experimentally validated.

The ECC-SMART project consortium consists of the EU, and Canadian and Chinese partners, who are making use of trans-continental synergy and knowledge developed separately by each partner, as well as under the GIF umbrella. The project consortium and scope were created according to joint research activities at the IAEA and at GIF, and as much data as possible will be taken from projects already performed. ECC-SMART brings together the best scientific teams working in the field of SCWRs, using the best facilities and methods worldwide, to fulfil the common vision of building an SCW-SMR in the near future.

For China, two projects supported by the Ministry of Science and Technology of China (MOST) were started in 2020. One is the GIF SCWR THS and the other is the GIF SCWR MC. The main goals of the two projects are to improve the China CSR1000 design and finish the international review before the end of 2022 so as to compile an expanded database based on previous research results. A kick-off meeting was held in 2020 in the Nuclear

power Institute of China (NPIC), Chengdu, China. Five Chinese institutes participated in the two projects, including the NPIC, Shanghai Jiaotong University (SJTU), Xi'an Jiaotong University (XJTU), the China Institute of Atomic Energy (CIAE) and the University of Science and Technology Beijing (USTB). Two virtual meetings were held online in March 2020 and September 2020 as a result of the COVID-19 situation.

### *Thermal-hydraulics and safety*

*Euratom activities*

The ECC-SMART project was launched in September 2020, and it comprises several different work packages (WP). WP 3 in particular focuses on thermo-hydraulics and safety analyses, and the current task is to define a common design that is to be analyzed.

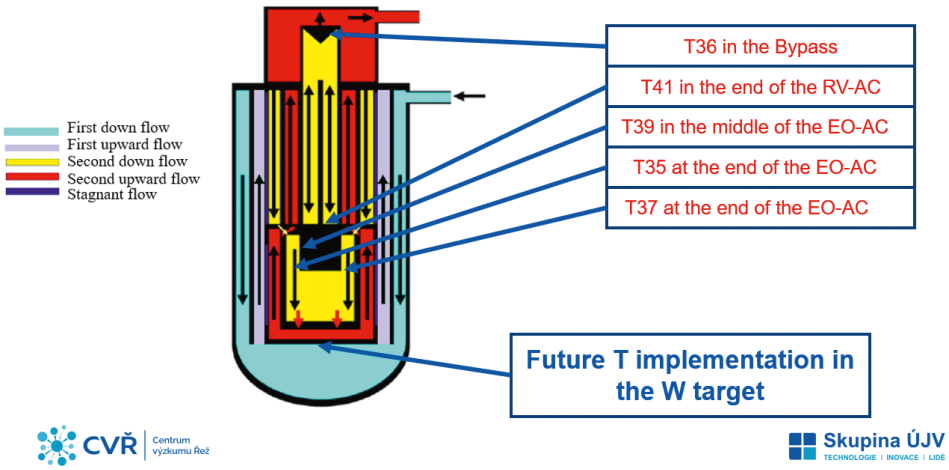
In Hungary, two institutions are working in close collaboration on THS research for SCWRs: the Centre for Energy Research (EK) and the Budapest University of Technology and Economics (BME). The EK continued its experimental activities (not only on heat transfer) in collaboration with the Department of Energy Engineering (DEE), the Department of Chemical and Environmental Process Engineering (DCEPE) and the Institute of Nuclear Techniques (NTI) of the BME. Two experimental works have been proposed in 2020 and may be elaborated further in the near future. The BME DEE has been working on theoretical research of water chemistry and thermal-hydraulic issues related to SCWs during the 2020 year. The BME NTI continued its numerical and theoretical research on the thermal-hydraulics of SCWs.

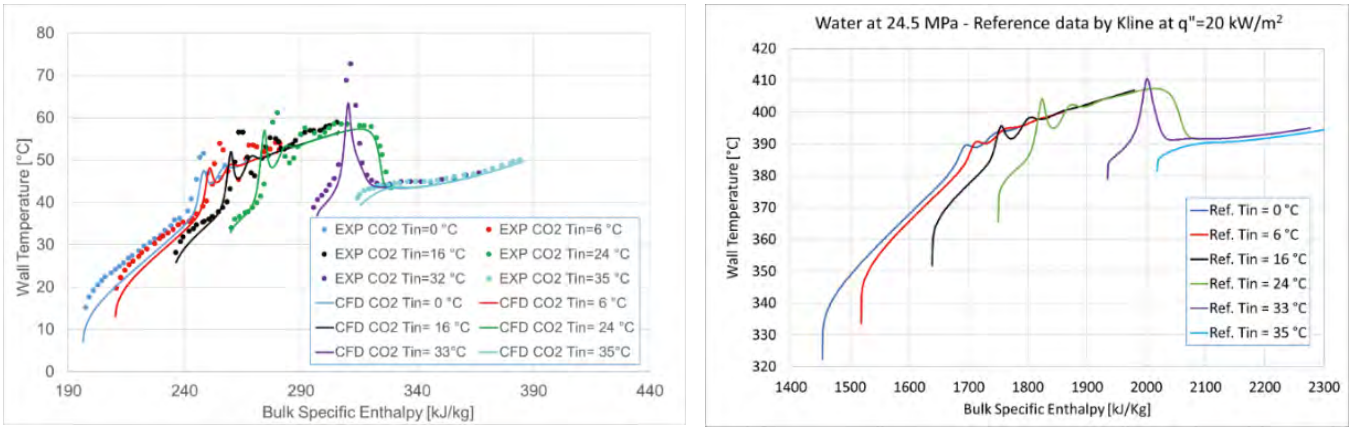
The main activity of the Research Centre Řež focused on the second licensing phase in order to insert the supercritical water loop (SCWL) in the LVR-15 reactor (Musa et al., 2020). This advanced facility is monitored by the Czech Republic State Office for Nuclear Safety (SONS). New analyses were performed with the goal of providing boundary conditions for the assessment of stress and strain calculations. The analyses were performed in ATHLET 3.1 code. After the initial revision of the flow regimes, all scenarios were reconsidered as a result of the lowering of the SCWL operational pressure from 25 MPa to 24 MPa. In addition, a new criterion for a new SCRAM signal will be implemented. The design will be improved by inserting a thermocouple in one of the two tungsten targets (T SCRAM around 550°C). This thermocouple could provide additional information on the heat transfer during transient phase.

Additional activities focused on simulating the SCWR thermo-hydraulic behavior. ATHLET code was benchmarked with the experimental data from the out-of-pile configuration. Among the postulated scenarios, an abnormal sequence (labelled A2 – loss of power in the loop) was analyzed. This scenario is similar to the postulated in-pile A2. The analyzed correlation in this phase were performed by Gupta et al. and Mokry et al.

In the past years, the University of Pisa in Italy has been addressing by a RANS model $CO_2$ data, water data and other data produced by several researchers, making use of an algebraic heat flux model (AHFM) developed in the STAR-CCM+ code, on the basis of the Lien et al. (1996) model. The RANS model was assessed and improved on a variety of experimental data, obtaining good results in comparison with experimental data. Based on these results and on data provided through direct numerical simulation (DNS) studies by a group at the University of Sheffield, in 2020 the subject of a fluid-to-fluid similarity theory for heat transfer at supercritical pressure (already proposed in past years) was further developed. Very good results were obtained as a result of these new steps, making it possible to confirm a sound rationale for assessing the scalability of results obtained with different fluids. The results of this work, performed at a distance during the current pandemic, have been published in three papers produced in 2020 and 2021 (Pucciarelli et al., 2020; Pucciarelli and Ambrosini, 2020; Kassem et al., 2021). Figure SCWR-2 provides comparisons between the original experimental data by Kline

**Figure SCWR-1. Active channel thermocouples map**






Note: as predicted by the similarity theory (data by Kline: $CO_2$, 8.35 MPa, 4.6 mm ID, q"=20 kW/m2, G=300 kg/m2s, upward flow, different inlet temperatures).

**Figure SCWR-2. Comparison between experimental and data (right) and corresponding similar water trends (left)**

obtained with $CO_2$, with the corresponding trends predicted in fluid similarity in the case in which water would be used instead, providing a clear idea of corresponding trends that can guide experimentalists in planning their experiments with simulant fluids.

SCWR research at the University of Sheffield (USFD) focuses on high-fidelity numerical simulations using DNS to produce high-quality reliable data in order to complement physical experiments. These tend to be for lower Reynolds numbers, but are able to produce detailed information and data to help understand the physics and the development of practical engineering models. The USFD has developed a versatile DNS code, CHAPSim, which has now been selected by the UK Collaborative Computational Project for Nuclear Thermal Hydraulics – supporting next generation civil nuclear reactors (CCP NTH) (sponsored by EPSRC EP/T026685/1 2020-2025) to be developed as a UK NTH community code. Over 2020, USFD work has included: 1) implementing conjugate heat transfer in CHPASim and carrying out some preliminary simulations; 2) carrying out simulations of flows in a horizontal orientation; and 3) investigating the development of a unified approach to explain the mechanisms of flow laminarization and heat transfer deterioration in a heated vertical flow of supercritical fluid. The USFD has also started working on simulations of SuperCritical fluid flows over rough/corroded surfaces in the context of the ECC-SMART project.

*Activities in China*

Two supercritical water thermal-hydraulics benchmarks were released in November 2020 in China. One is on the 2X2 bundle SC-water tests performed by the NPIC and the XJTU several years ago. The other is on parallel channel instability experiments of supercritical water. The structure layouts are shown in Figure SCWR-3. The 2X2 rod bundle tests are used to validate the computational fluid dynamics (CFD) tools (e.g. CFX, Fluent, Star CCM+, Open-FOAM) and subchannel tools (e.g. ATHAS, SC-COBRA). The parallel channel instability experiments were used to validate the system analysis tools, such as SC-TRAN, RELAP5 and APROS. The first-round comparisons are planned for 2021.

Experiments on heat transfer of supercritical water in a single subchannel with grid spacer were successfully carried out. More reference characteristics about the influence mechanism of the grid spacer can be observed through this experiment in Figure SCWR-4. The section structure of the test part is clearly shown in in Figure SCWR-3. The test section is a triangular-shaped sub-channel with a standard grid spacer of an 8 mm core rod diameter, 1.4 pitch ratio, 990 mm total length and 2.5 mm thickness. The grid spacer was located 550 mm from the inlet. The influence of pressure, flow rate, and heat flux on the standard grid spacer within the framework of sub-channel flow heat

**Figure SCWR-3. The structure layout of 2X2 rod bundles (left: from NPIC; right: from XJTU)**

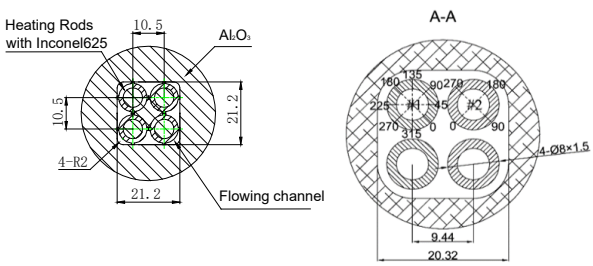
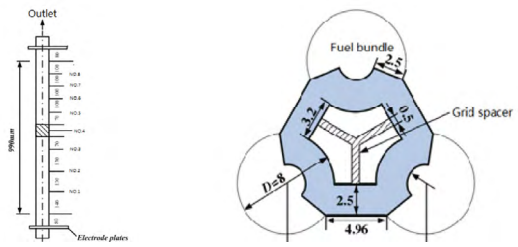



**Figure SCWR-4. The structure layout of the sub-channel test section**



transfer characteristics of supercritical water were studied. Figure SCWR-3 shows the distribution curve of the heat transfer coefficient, changed with the enthalpy in several working conditions. With the comparison of the five curves, the grid spacer greatly influenced the heat transfer characteristic of the entire flow, especially downstream. For the downstream flow direction, the fluid disturbance was reduced because of the decreasing blockage area. The circulation area abruptly widened in comparison to the grid spacer.

A new thermal amplification system was constructed to improve the accuracy of the calculated heat transfer coefficient of a supercritical fluid near its pseudo-critical point with the influence of the parameters of supercritical water cooling via downward flow inside a tube accompanied by pool boiling outside of the tube (McLellan et al., 2021). The results of this study will be helpful in understanding the effects of pressure, fluid temperature, mass flux and heat flux on the characteristics of supercritical downward cooling heat transfer. The test section is shown in Figure SCWR-6. A type 316 stainless steel circular tube was used as the test section. Its inner diameter and wall thickness were 20 mm and 2.5 mm, respectively. Five-sixths of the test section was directly immersed in the water of test pool. Figure SCWR-6 presents the variations in the wall temperature, heat flux and heat transfer coefficient with respect to the fluid temperature. The wall temperature gradually increases with fluid temperature below and above the pseudo-critical point. At a pseudo-critical point (384.9°C), a sharp jump appears in the wall temperature, in which it increases from 253.5°C to 317.4°C with an increase in fluid temperature from 375.7°C to 383°C. The heat flux increases with fluid temperature below the pseudo-critical point, and is almost maintained at a constant value above the pseudo-critical point, but decreases sharply near the pseudo-critical point itself. The heat transfer coefficient increases and then decreases with fluid temperature, reaching its maximum at 14 kW·m−2·K at the pseudo-critical point.

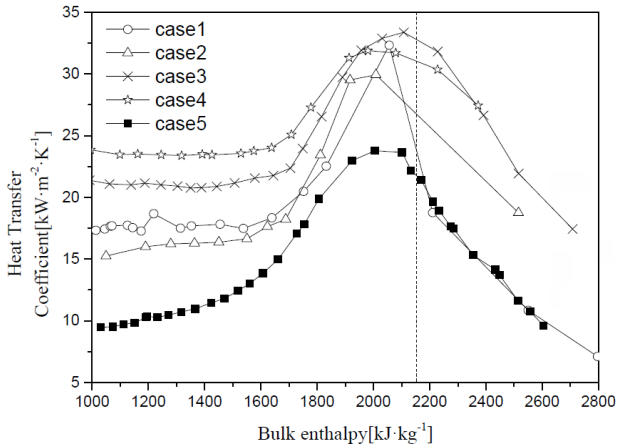
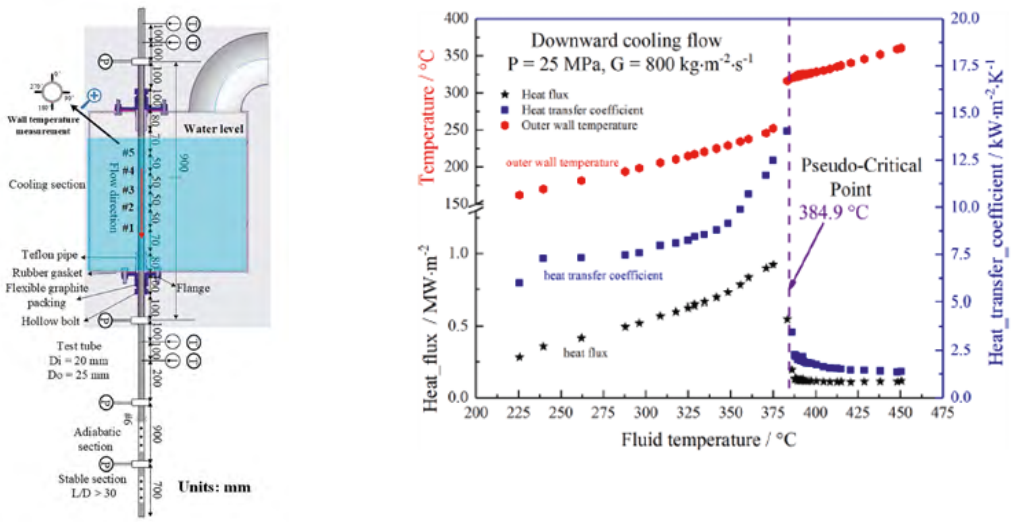



Figure SCWR-5. The distribution curve of heat transfer coefficient under supercritical conditions

*Activities in Canada*

In Canada, a preliminary SCW-SMR concept has been established from the reference Canadian SCWR concept. The overall core diameter is about 3 meters, which has resulted in a slender reactor core, inefficient from both neutronics and thermal hydraulics points of view. The selection and optimization of the SCW-SMR fuel assembly uses knowledge gained from the development of Canadian fuel bundles. It considers geometrical features (such as heated perimeter, flow area, sub-channel size) to minimize the maximum cladding temperature (MCT). From a thermal-hydraulics and reactor physics coupled analysis, the study focused on computing the reactor power distribution and the maximum channel power. Based on these results, a thermal-hydraulics analysis using the subchannel code ASSERT-PV was performed. The analysis focused on the maximum cladding temperature. Several concepts have been proposed, however, based on the results, and two concepts are being further investigated, namely: 1) THE CANFLEX-20 fuel bundle; and 2) the 64-element concept used for the Canadian SCWR concept. The assessment of these fuel bundles is ongoing and the results are still preliminary. Moreover, a proper and complete development requires linking to operational and

Figure SCWR-6. The downward flow test section (left) and the general behaviors (right)



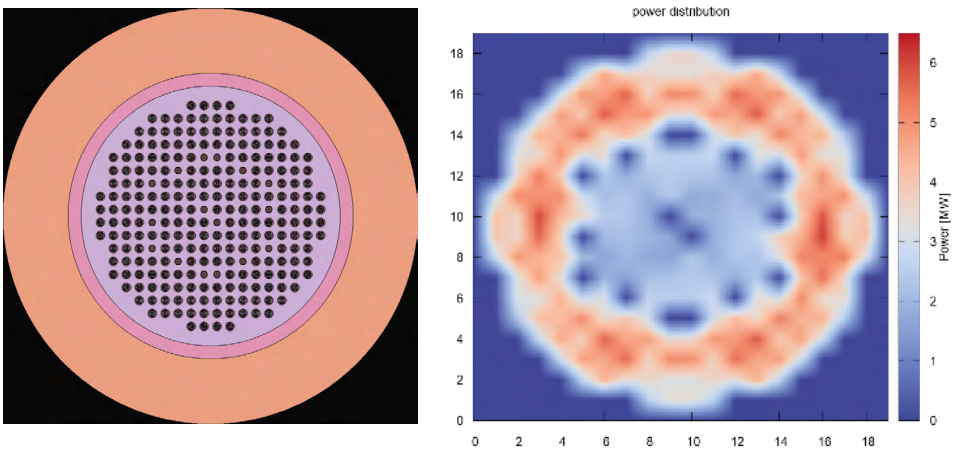



**Figure SCWR-7. (a) Preliminary Canadian supercritical water-cooled SMR (core layout), (b) channel power heat map (in MW) of the preliminary Canadian supercritical water-cooled SMR**

safety constraints and requirements, such as higher burn-up, maximum linear power, and higher-reliability or reduced operational and maintenance costs. These analyses are ongoing.

The CNL gravity assisted loop uses a heat pipe to remove heat from a pool of water. The experimental loop was originally designed to be used to remove heat from the spent-fuel pools. A study to examine the detailed behavior of this gravity assisted loop was recently undertaken. The loop was verified and tested to ensure it is fully operational, and a series of tests were conducted to provide experimental data that can be compared to a computer model. The objective was to further analyze the loop and to dimension components that could be used to design and assess a passive cooling system that could be included in the Canadian SCW-SMR. The system was modelled using CATHENA and RELAP5-3D codes. The investigation revealed unexpected effects in the steam piping between the evaporator and condenser due to the use of a Coriolis mass flow metre, which indirectly limited heat transfer. Additionally, the presence of non-condensable gases in the system further reduced heat transfer. The mathematical models were refined to include these factors, resulting in good agreement with experimental data. Further, it was demonstrated that changes to the loop, such as increasing the diameter of the condensate return line and relocating the mass flow metre are needed. Simulation results also provide guidance for the next phase of investigation, including the addition of a steam-to-air heat exchanger.

### *Materials and chemistry*

*Euratom activities*

A European Union funded project called Mitigating Environmentally-Assisted Cracking Through Optimisation of Surface Condition (MEACTOS) is ongoing to study the effect of surface finishing on the corrosion resistance of selected alloys (A182 and stainless steel 316 L). In this project, one task involves SCW being used as an aggressive environment because of the higher test temperatures. Several laboratories from the EU are involved in these SCW activities (e.g. JRC, Valtion Teknillinen Tutkimuskeskus [VTT] and CIEMAT).

In the context of the ECC-SMART project, work package 2 is focusing on materials testing with more than 400 person months and the participation of 11 laboratories, among which 8 European laboratories/companies[1], one from Canada (CNL) and two from China (University of Science and Technology Beijing [USTB] and Shanghai Jiao Tong University [SJTU]). In this work package, the corrosion behavior in SCW of two of the most promising materials for cladding applications (i.e. 800 H and 310 S) will be studied.

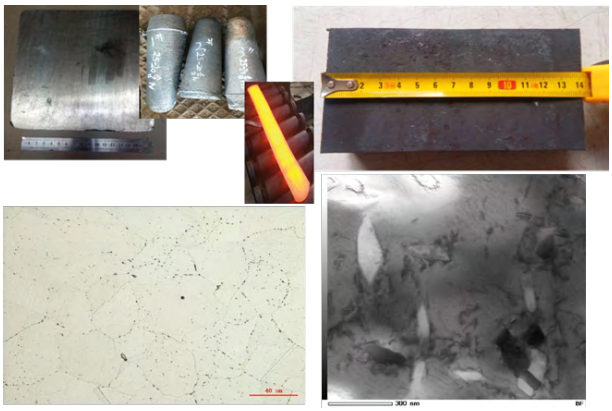
JRC Petten, in collaboration with VSCHT, has published a paper in the Corrosion Science Journal entitled: "In-situ electro-chemical impedance measurement of corroding stainless steel in high subcritical and supercritical water". This work summarized the main findings from a study on changes in the physico-chemical properties of SCW with pressure and temperature. This work could be considered the starting point of the tests to be carried out in the ECC-SMART project on this topic.

CIEMAT finished a preliminary work in collaboration with the Research Centre Řež. Selected, in-situ tensile tests with a nickel-based alloy 690 pretested in SCW were performed in this work. Both laboratories plan to continue these tests throughout 2021, and for this reason CIEMAT has designed and machined new specimens that fit into the scanning electron microscope available at the Research Centre Řež Pilsen, where the previous tests were performed. It is expected that this new configuration of specimens will allow a more in-depth study of the role of microstructural defects in the corrosion behavior of Ni base alloys. Preliminary results from these tests were presented at the Electric Power Research Institute (EPRI) annual meeting on alloy 690 (Tampa, December 2019).

1. CIEMAT, Research Centre Řež, JRC, Regia Autonoma Tehnologii Pentru Energia Nucleara [RATEN], Slovak University of Technology in Bratislava (STU), VTT, the University of Prague (VSCHT), ENEN (European Nuclear Education Network)

Research activities at the Research Centre Řež are linked to previous years, focusing on two more exposures in the SCWL, commissioning and test operation of the ultracritical water loop (UCWL) and publishing of data from exposures in 2018-2020. Effects of supercritical water on the corrosion behavior was studied on perspective materials for nuclear power plants, such as alloy 800H, AISI 310S, AISI 321SS (08Cr18Ni10Ti), T505 (T91), Inconel 718, NIMONIC 91 and NITRONIC 60. Two exposures in the SCWL were carried out at 395°C, 25 MPa, each for 1 000 hours. The supercritical medium was deoxygenated water with pH 5.5-6.8, conductivity 0.77-1.88 µS/cm, Fe <92 µg/l and TOC 886 to 251 µg/l. Commissioning of the UCWL took place during the year 2020 and the test operation at 600°C, 25 MPa, with 500 hours running since November 2020. The effects of SCW were evaluated by weight changes, scanning electron microscopy (SEM) with chemical analysis detectors (BSE, EDS, EBSD) in combination with X-ray diffraction. In addition, the special analysis to detect the very thin surface layer by focused ion beam (FIB) was used on alloy 800H. In this case, no thin surface layer was observed. Only randomly distributed spinels were observed: trevorite ($NiFe_2O_4$, a = 8.347 Å), magnetite ($FeFe_2O_4$, a = 8.397 Å) and chromite ($FeCr_2O_4$, a = 8.376 Å) crystals with dimensions up to 1 µm on 800H and only magnetite crystals on 08Cr18Ni10Ti after three exposures – 1 700 hours. The density of all crystals increased slightly after the second and third exposure. Simultaneously, the surface of Inconel was irregularly covered by oxide crystals about 1 µm, identified as hercynite and chromite $Fe(Al,Cr)_2O_4$ after one exposure – 1 000 hours. Other materials will be analyzed in 2021. Selected materials, such as 800H, AISI 321SS and T91, were exposed several times, for 2 700 hours (four exposures) in total. The commissioning of new facilities such as in-pile and out-of-pile autoclaves (volume 137 ml, 600°C/25 MPa and volume 850 ml, 700°C/30 MPa) are in progress. The effect of radiation on microstructure stability and corrosion resistance of candidate materials exposed to SCW will thus be possible to assess.

**Figure SCWR-8. AFA steels developed at the University of Science and Technology Beijing (China)**



*Activities in China*

The SJTU is studying the corrosion behavior of alloy 800H, austenitic stainless steel 310S and alumina forming austenitic (AFA) alloys in SCW. Moreover, particular interest is being taken in the effects of variables such as temperature, plastic deformation, water chemistry and surface finishing in the corrosion behavior of these alloys, and for this reason the SJTU is performing tests to study the effects of these variables on general corrosion and stress corrosion cracking processes in SCW. In addition, the SJTU is leading the third international round robin on corrosion behavior of candidate alloys for the SCWR. In this case, the international group will focus their efforts on the study of stress corrosion cracking processes in the SCW. Part of these activities will be complementary to the ECC-SMART project.

The USTB is designing and fabricating new grade materials suitable for fuel-cladding application under a high-temperature SCWR environment. These materials will be co-evaluated by colleagues at SJTU and the NPIC to determine the candidate materials for further round robin tests. The composition design of new grade materials is mainly based on 310SS, including oxide dispersion-strengthened (ODS) steel type fabricated through the powder metallurgy technique and AFA steel type through a traditional melting and casting method. Both types of new grade materials show promising high-temperature strength and SCW corrosion resistance. ODS steels show much better microstructure stability at high temperature, while AFA steels are attractive in terms of engineering and manufacturing. AFA steels show good performances in the aspect of high-temperature strength as compared with similar traditional steels because of the formation of strengthening phases of NbC, Laves and B2-NiAl, which are more stable than M23C6 in traditional steels. It can be expected that the AFA steels will also show better corrosion resistance in SCW environments as a result of the formation of alumina surface oxide, which is dense, thin and stable, as approved by exposure tests in SCW. Figure SCWR-8 shows the pictures of ingots, microstructure and precipitates of a fabricated AFA steel.

*Activities in Canada*

The CNL has studied the corrosion behavior in SCW of alloy 625 and alloy 800H. Both have showed an excellent strength and ductility after welding. In addition, Cr-coated Zr-2.5Nb, Zr-1.2Cr-0.1Fe, Ti and Ti-6Al-4V met performance criteria in short-term tests. Moreover, they have developed a schedule for proton and heavy ion (Cr3+) irradiation up to 5-15 dpa in top 20-30 µm, followed by micro-mechanical testing. It is expected that the irradiation will start during the year 2021.

Engineering, structural and core metallic nuclear components of NPPs must handle thermal loading; otherwise, localized hotspots resulting from changes in geometry or heat transfer fouling

could develop within the column and degrade the performance of components during operation and over time, such as that of fuel-cladding tubes. To further progress in this area, a project was established at the CNL with the objective of determining the thermal properties data (thermal conductivity) of candidate cladding tube material of small modular SCWRs, and to assist thermal-hydraulics calculations.

Chromium coated Zr-2.5% Nb material, such as a cladding material, showed a better resistance to corrosion in supercritical water conditions. In the open literature, empirical or semi-empirical models are available to assess the thermal conductivity of zirconium based fuel-cladding materials, but there is limited or no data on thermal conductivity of Cr-coated and/or oxided zirconium based fuel-cladding materials is lacking or not available to support thermal hydraulics modelling. Thermal conductivity measurements using the laser flash method were performed on the as-received material (Zr-2.5% Nb) at different test temperatures. The specimens are disc shaped with 12.16 mm in diameter, and an average thickness of 1.364 mm was cut from Zr-2.5% Nb. At each temperature, measurements were repeated five times to obtain an average. In the current study, thermal diffusivity and conductivity of baseline material is determined.

The CNL has an ongoing R&D program to support the development of a scaled-down 300 MWe version of the Canadian supercritical water reactor (SCWR) concept. The 300 MWe and 170-channel reactor core concept uses LEU fuel and features a maximum cladding temperature of 500°C (McLellan et al., 2021). There are challenges to using zirconium alloys at temperatures exceeding 400°C. Zirconium alloys such as Zr-2 and Zr-4 typically experience high corrosion rates, and they are known to experience hydrogen embrittlement from aggressive hydrogen pickup during corrosion.

Two materials from the previous experimental campaign – Zr-1.2Cr-0.1Fe (R60804) and Zr-2.5Nb (R60901) –were also used in the campaign described here. A nominal thickness of 5 to 10 μm of chromium coating was tested for about 150 hours of exposure time in oxygenated SCW. The oxidizing environment was chosen to simulate water radiolysis in the SCWR core. In addition, the corrosion behavior of candidate materials in an alkaline environment using LiOH solution was also evaluated.

Microstructural analyses, including scanning electron microscopy (SEM) and energy-dispersive X-ray spectroscopy (EDX) were performed to observe the effects of the microstructure of the base alloys on the observed chromium coating. The results from short-term autoclave oxidation at supercritical water conditions show that when the coating reaches approximately 10 μm thickness, the grain orientation of base Zr- and Ti-based alloys does not affect the morphology of the chromium coating. Moreover, weight gain measurements indicate a significant improvement in corrosion resistance of coated coupons compared to the as-received alloys, for Zr-2.5Nb and Zr-1.2Cr-0.1Fe. Long exposure experiments are ongoing.

**Yanping Huang**
Chair of the SCWR SSC, with contributions from SCWR members

# Sodium-cooled fast reactor

## Main characteristics of the system

The primary mission of the sodium-cooled fast reactor (SFR) is the effective management of high-level waste and uranium resources. If innovations to reduce capital cost and improve efficiency can be realized, the Gen-IV SFR is an attractive option for electricity production. The *Generation IV Technology Roadmap* ranked the SFR highly for advances it offers towards sustainability goals. The closed fuel cycle significantly improves the use of natural uranium, as compared to ~1% energy recovery in the current once-through fuel cycle. By recycling the plutonium and minor-actinide spent-fuel components, decay heat and the radiotoxicity of the waste are minimized. The SFR is also highly rated for safety performance.

The SFR system uses liquid sodium as the reactor coolant, allowing high-power density with low coolant volume fraction. Because of advantageous thermo-physical properties of sodium (high boiling point, heat of vaporization, heat capacity and thermal conductivity) there is a significant thermal inertia in the primary coolant. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water, and requires a sealed coolant system. The primary system operates at near-atmospheric pressure with typical outlet temperatures of 500-550°C; at these conditions, austenitic and ferritic steel structural materials can be used, and a large margin to coolant boiling at low pressure can be maintained. The reactor unit can be arranged in a pool layout or a compact loop layout. The typical design parameters of the SFR concept being developed in the framework of the Gen-IV system arrangement (SA) are summarized in Table SFR-1. Plant sizes ranging from small modular systems to large monolithic reactors are being considered.

**Table SFR-1. Typical design parameters for the Gen-IV SFR**

| Reactor parameters | Reference value |
|---|---|
| Outlet temperature | 500-550°C |
| Pressure | ~1 atmosphere |
| Power rating | 30-5 000 MWt (10-2 000 MWe) |
| Fuel | Oxide, metal alloy, and others |
| Cladding | Ferritic-martensitic, ODS, and others |
| Average burn-up | 150 GWD/MTHM |
| Breeding ratio | 0.5-1.30 |

There are many sodium-cooled fast reactor conceptual designs that have been developed worldwide in advanced reactor development programs. For example, the BN-800 reactor in Russia, the European fast reactor in the EU, the advanced liquid metal reactor (PRISM) and integral fast reactor programs in United States, as well as the demonstration fast breeder reactor in Japan, have been the basis for many SFR design studies. For Gen-IV SFR research collaboration, several system options that define the general classes of SFR design concepts have been identified: loop configuration, pool configuration and SMRs. Furthermore, within this structure several design tracks that vary in size, key features (e.g. fuel type) and safety approaches have been identified with pre-conceptual design contributions by Gen-IV SFR members: CFR1200 (China), the Japanese sodium-cooled fast reactor (JSFR, Japan), Korea advanced liquid metal reactor (KALIMER, Korea), ESFR (Euratom), BN-1200 (Russia) and AFR-100 (United States). Gen-IV SFR design tracks incorporate significant technology innovations to reduce SFR capital costs through a combination of configuration simplicity, advanced fuels and materials and refined safety systems. They are thus used to guide and assess Gen-IV SFR R&D collaborations.

## Status of cooperation

The system arrangement for Gen-IV international R&D collaboration on the SFR nuclear energy system became effective in 2006 and was extended for a period of ten years in 2016. Several new members were added to the original agreement and the United Kingdom was welcomed to the system arrangement in 2019. The present signatories are: the French Alternative Energies and Atomic Energy Commission (CEA), France; the Department of Energy, United States; the Joint Research Centre, Euratom; the Japan Atomic Energy Agency, Japan; the Ministry of Science and Information and Communication and Technology (ICT), Korea; the China National Nuclear Corporation, China; Rosatom, Russia; the Department for Business, Energy and Industrial Strategy, the United Kingdom.

Based on international R&D plans, Gen-IV SFR research activities are arranged by SFR signatories into four technical projects: system integration and assessment (SIA), safety and operations (SO), advanced fuel (AF) and component design and balance of plant (CD&BOP).

## R&D objectives

SFR designs rely heavily on technologies already developed and demonstrated for sodium-cooled reactors, and for the associated fuel cycle facilities that have successfully been built and operated in several countries. Overall, approximately 400 reactor-years of operating experience have been logged on SFRs: 300 years on smaller test reactors and 100 years on demonstration or prototype reactors. Significant SFR research and development programs have been conducted in

France, Japan, India,[1] Russia, the United States and the United Kingdom. The only SFR power reactors in operation are the BN-600 and the BN-800 (both in Russia). Currently operating test reactors include the BOR-60 (Russia) and CEFR (China). The Joyo (Japan) test reactor is in the licensing process for restart. New SFR test reactors, the MBIR (Russia) and VTR (United States), are expected in the next decade. In addition, SFR technology R&D programs are being pursued by all SFR GIF members.

A major benefit of the maturity of the SFR technology is that the majority of the remaining R&D needs are related to performance rather than the viability of the system. Accordingly, Gen-IV collaborative R&D focuses on a variety of design innovations for actinide management, improved SFR economics, the development of recycle fuels, in-service inspection and repair (ISI&R) and verification of favourable safety performance.

The system integration and assessment (SIA) project: through a systematic review of the technical projects and relevant contributions on design options and performance, the SIA project will help define and refine requirements for Gen-IV SFR concept R&D. The SFR system options are assessed with respect to Gen-IV goals and objectives. Results from the R&D projects will be evaluated and integrated to ensure consistency.

The safety and operation (SO) project: the SO project is arranged into three work packages (WPs): 1) WP SO1 "methods, models and codes" for safety technology and evaluation; 2) WP $SO_2$ "experimental programs and operational experience", including the operation, maintenance and testing experience in facilities and SFRs (e.g. Monju, Joyo, Phenix, BN-600, BN-800 and CEFR); and 3) WP SO3 "studies of innovative design and safety systems" related to safety technology, such as inherent safety features and passive systems.

The advanced fuel (AF) project: the AF project aims at developing and demonstrating minor-actinide-bearing (MA-bearing) high burn-up fuel for SFRs. The R&D activities of the AF project include fuel fabrication, fuel irradiation and core materials (cladding materials) development. The advanced fuel concepts include both non-MA-bearing driver fuels (reactor start-up) and MA-bearing fuels as driver fuels and targets (dedicated to transmutation). The fuels considered are oxide, metal, nitride and carbide. Currently, cladding/wrapper materials under consideration include austenitic and ferritic/martensitic steels, but the aim is to transition in the longer term to other advanced alloys, such as ODS steels.

The component design and balance-of-plant (CD&BOP) project: this project includes the development of advanced energy conversion systems (ECS) to improve thermal efficiency and reduce secondary system capital costs. The project also includes R&D on advances in sodium ISI&R technologies, small sodium leak consequences and new sodium testing capabilities. The main activities in ECS include: 1) development of advanced, high-reliability steam generators and related instrumentation; and 2) the development of advanced energy conversion systems (ECS) based on a Brayton cycle with supercritical carbon dioxide or nitrogen as the working fluid.

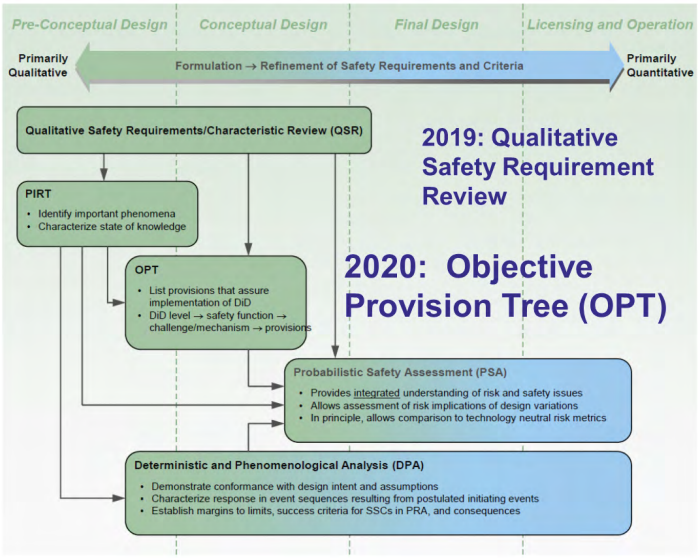



**Figure SFR-1. Assessment of safety requirements for the European sodium fast reactor**

## Main activities and outcomes

### *SIA project*

The China Institute of Atomic Energy (CIAE) contributed a study that evaluates the main heat transfer parameters of the CFR1200 design. Key factors that significantly influence the thermal performance were identified (e.g. primary/secondary circuit temperatures). They performed sensitivity analyses for these main factors and quantified the impacts on system efficiency and component design.

The CEA has developed the "CADOR" core concept, which reduces the volume power and adds some moderators in order to eliminate severe accidents scenarios induced by unprotected events. As a result of safety analysis studies, the CADOR core has demonstrated good natural behavior in the case of unprotected transients through improvements inDoppler feedback. However, the volume of the core becomes larger than the classical core design (ASTRID CFV core).

The JRC conducted on an assessment of safety requirements for the European sodium fast reactor (ESFR) using Integrated safety assessment methodology (ISAM) tools. The Horizon 2020 European Sodium Fast Reactor Safety Measures

1. India is not a member of GIF.

Assessment and Research Tools (ESFR-SMART) project, launched in 2017, aims at enhancing further the safety of Gen-IV SFRs, and in particular the commercial-size ESFR in accordance with the European Sustainable Nuclear Industrial Initiative (ESNII) roadmap. Within the project, Euratom applied the ISAM tools developed by GIF RSWG to the ESFR in order to assess safety requirements. This contribution is multi-annual, and 2019 deliverables dealt with the application of the ISAM qualitative safety feature review (QSR). In 2020, Euratom released deliverables on the application of the ISAM objective provision tree (OPT) to ESFR-SMART studies.

The JAEA has been reconsidering advantages gained from previous innovative technologies because significant changes were required in the design of the JSFR after the Fukushima Daiichi NPP accident. The R&D load, the risk of each innovative technology and updated lists of innovative technologies for loop-type reactors were reviewed. This review is based on development easiness and preparation of design standards. Additionally, the total mass and safety of the design are being considered.

KAERI is developing a steam generator concept to minimize sodium-water reaction. A copper bounded steam generator (CBSG) was selected as an alternative steam generator concept. Triple isolation wall structure with steel tube/copper matrix/steel tube layered geometry and eliminated welds were introduced to achieve a very low probability of sodium-water reaction, and the heat exchanger modules were designed by sizing and CFD analysis. Manufacturing tests of a small-scale module through hot isostatic pressing (HIP) diffusion bonding, tension and contact thermal resistance tests of HIP bonding materials, and thermal fluid visualization tests and structural analyses for CBSG were performed.

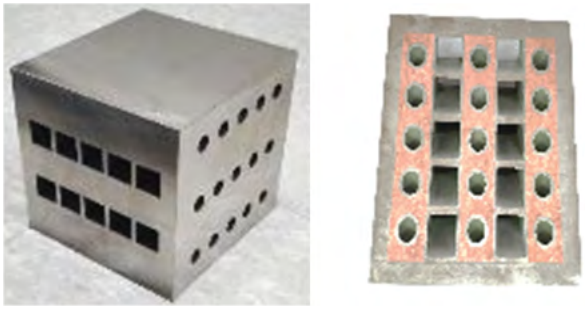


Figure SFR-2. Manufacture of a small-scale copper bonded steam generator module using the hot isostatic pressing process

## Safety and operations project

On the topic of the safety and operation (SO) project, the common project that consists of two benchmark analyses (the EBR-II test and Phenix dissymmetric tests) started from the last quarter of 2019. The first phase of the benchmark analysis is a "blind phase", which will take two years to complete. Argonne National Laboratory (ANL), the JAEA and KAERI completed the blind phase of the EBR-II test study at the end of 2020.

### *WP SO 1: Methods, models and codes*

The CEA has analyzed debris bed cooling after a severe accident that followed an unprotected loss of flow (ULOF). In this analysis, one in-vessel DHX was considered and one ex-vessel decay heat removal system. Two types of models have been used for CFD calculations: the laminar model and the K-Ω SST turbulence model. The CEA has demonstrated the calculation results on core catcher temperatures and hot pool temperatures. In terms of the core catcher, the temperature of the insulator (i.e. a few centimetres of $ZrO_2$) was less than 1 000°C. That value was verified at less than the fusion temperature of $ZrO_2$; 2 750°C during the simulations. The temperature of the debris, taking into account the conservative assumptions performed, were less than 1 500°C. This result provides a wide margin for a non-desired scenario disruption.

The CEA also presented detailed calculations for flow patterns in the lower plenum, which requires a full 3D-calculation approach. The heat removal trend up until 35 000 seconds was calculated through CFD, modified by the porous media

Figure SFR-3. Local debris and insulator temperatures

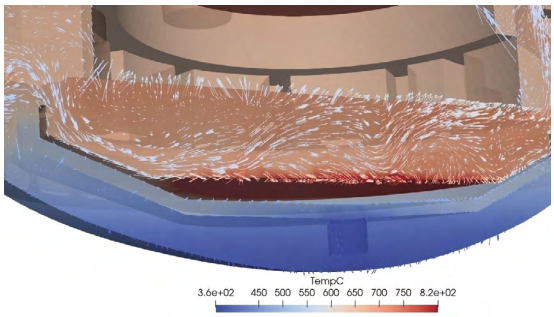
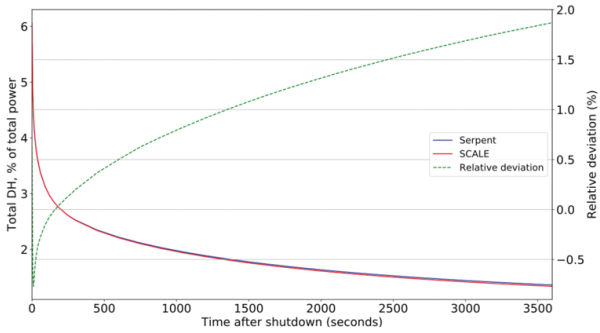



Figure SFR-4. Total decay heat normalized to nominal power



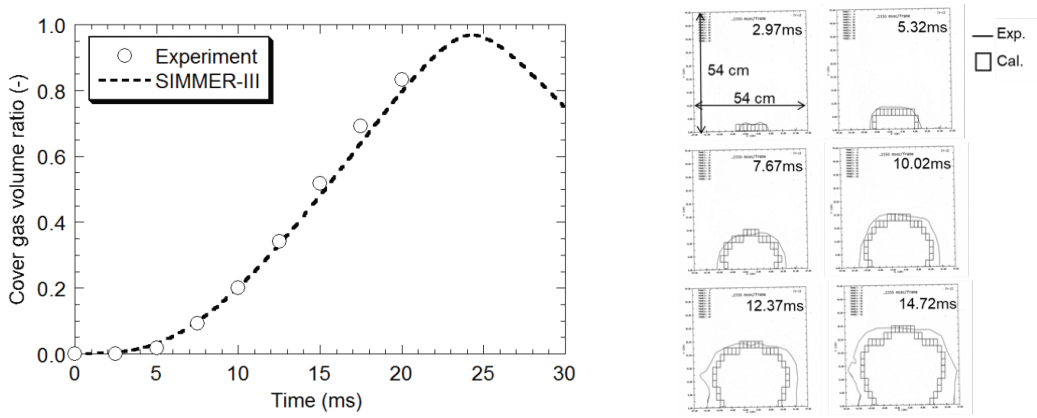


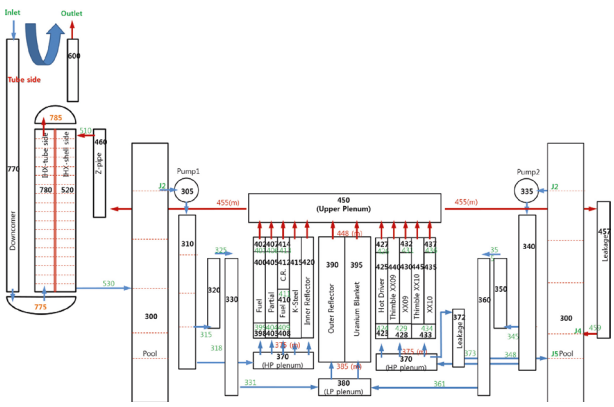

Figure SFR-5. Examples of SIMMER-III analysis results for the OMEGA test

model. After 35 000 seconds of transient, the power balance between heat decay and power removed had not yet been reached. The power gap decreases suggest a convergence after a few days.

Euratom has assessed safety parameters related to the end of cycle (EOC) loading of the ESFR-SMART core, including full core and local effect analysis. The EOC state of the core presents the most limiting case for safety analysis. The deliverable discusses estimation safety parameters, such as control rod insertion S-curve, sodium void reactivity, thermal expansion, Doppler constant and void worth. In addition, sensitivity and uncertainty analyses of different safety parameters related to nuclear data have been performed. The spatial and time-dependent decay heat characteristics normalized to nominal power were also estimated.

The JAEA has been developing an advanced computer code, SIMMER-III/IV, for the analysis of a core disruptive accident (CDA). For the validation of the SIMMER-III code, the JAEA presented a comparison of analysis results for material expansion dynamics, with experimental results. The JAEA selected two experiments for the validation study: the VECTORS test carried out by JAEA to focus on the phenomena of multi-phase flow in structure, and the OMEGA test undertaken by Purdue University to focus on the phenomena of huge vapour bubble expansion dynamics. SIMMER-III successfully simulated both the VECTORS test and the OMEGA test, and has proven to be practical and useful for SFR severe accident analysis.

Figure SFR-6. MARS-LMR modelling of the EBR-II BOP test




KAERI performed a preliminary benchmark analysis of EBR-II BOP-301/302R tests using MARS-LMR. Typical behaviors and significant increases in the core inlet temperature in unprotected loss of heat sink (ULOHS) were investigated in the EBR-II BOP-301/302R. Similar trends in the BOP301 and 302R results were observed and compared with ANL calculations.

The Institute of Physics and Power Engineering (IPPE, Rosatom) continued to develop the 3D severe accident analysis code, COREMELT3D. The 3D model of the reactor gas system (from the gas volume under the sodium level in the reactor through the expansion tank, and up to the ventilation system) has been developed and implemented into the code. This model has been integrated into the primary circuit 3D thermo-hydraulic model to simulate the transport of gaseous fission products from disintegrated fuel pins to the ventilation system, as well as potential releases into the environment.

The IPPE has performed integral analysis of the consequences of severe accidents in the BN-1200. The following codes have been used: COREMELT3D (core, primary and intermediate circulation loops, emergency system of heat removal, reactor gas system), KUPOL-BR (ventilation system), VYBROS-BN (transport of radioactive products in the environment under different meteorological conditions, doses). The IPPE has performed preliminary experiments with thermite compositions to obtain the melt of stainless steel with high temperatures. This technique will be used in a facility (which is currently being designed) to simulate transport of melted core in SFR conditions.

### *WP SO 2: Experimental programs and operational experiences*

The CIAE conducted experimental research and code development for a heat transfer analysis of the China experimental fast reactor (CEFR) damaged spent-fuel assemblies in a closed space. The experiment simulated the spent-fuel assemblies during transportation, and the heat transfer characteristics were investigated.

Euratom contributed to the standard procedure for the sodium loop operation and measurement treatment. The deliverable presents the results of the selection of review elements for sodium technology. Aspects considered include procedures used for testing prototypical components at large facilities, procedures for calibration of sensors and signal treatment for the measuring system, methodologies for treatments of the measurement for subsequent use as input data for codes and the conservation of a facility in appropriate conditions enable restart in safe conditions.

### *WP SO 3: Studies of innovative design and safety systems*

The JAEA attempted to identify accident sequences against severe accidents, referring to the rule of new regulations for LWRs. The JAEA carried out an internal event probabilistic risk assessment (PRA) in order to identify the accident sequences to be evaluated. Measures against the accident sequences (anticipated transient without scram [ATWS] and loss of heat removal system [LOHRS]) were studied and developed (see Figure SFR-7). Regarding external events, earthquakes and tsunamis were studied as the most prioritized initiating events. The accidents initiated with these events can be categorized into the accident category identified through internal event PRA (i.e. protected loss of heat sink [PLOHS]). By evaluating identified accident sequences, the JAEA can confirm that most of the SA (i.e. core damage and control valve failure) can be prevented.

The CEA has been studying the design of a small modular fast reactor. It has been considering constraints such as the plutonium content and maximal linear heat rate. The CEA has shown that criticality, which is an issue for small cores, could be achieved by adjusting the number and design of the assemblies. Further studies are planned (in 2021) to investigate other dimensioning parameters for a small modular sodium-cooled fast reactor (SMSFR).

## Advanced fuels project

The AF project consists of three work packages (WPs): WP2.1 "SFR non-minor-actinide (MA)-bearing driver fuel evaluation, optimization and demonstration"; WP2.2 "MA-bearing transmutation fuel evaluation, optimization and demonstration"; and WP2.3 "high-burn-up fuel evaluation, optimization and demonstration".

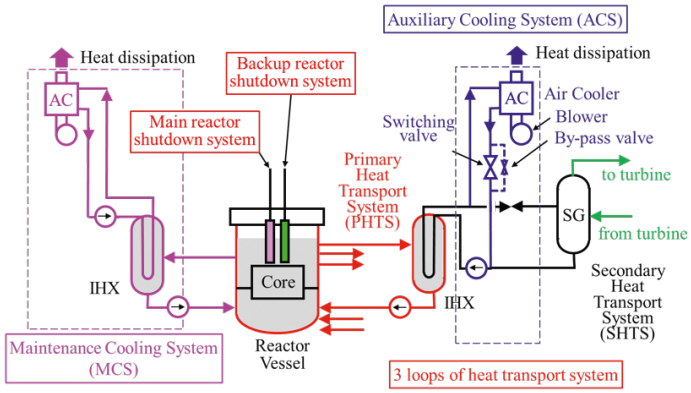



Figure SFR-7. MARS-LMR Modelling of the EBR-II BOP test

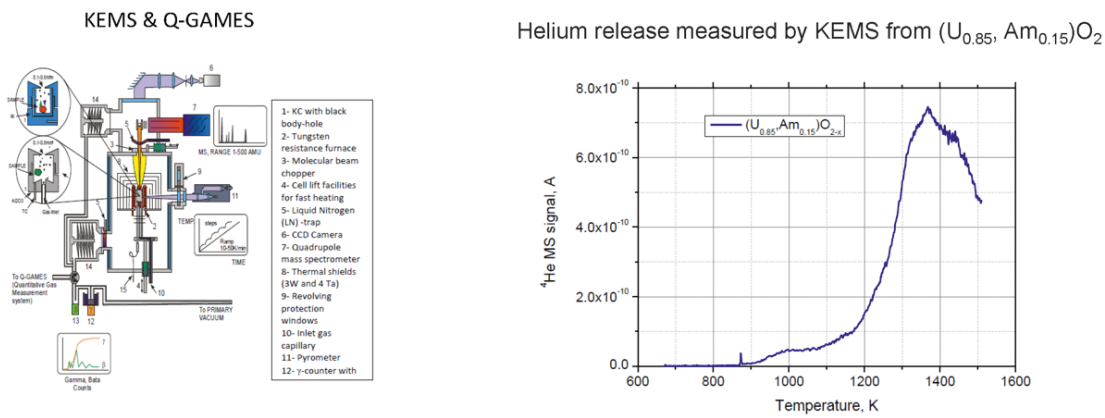
### *WP2.1: SFR non-MA-bearing driver fuel evaluation, optimization and demonstration*

The CIAE is preparing to undertake some irradiation tests. It has finished the fabrication of dummy irradiation assemblies and out-of-pile hydraulic tests (hydraulic characteristic experiments). In 2020, more out-of-pile hydraulic and mechanical tests were conducted to ensure the future safety of in-pile irradiation assemblies.

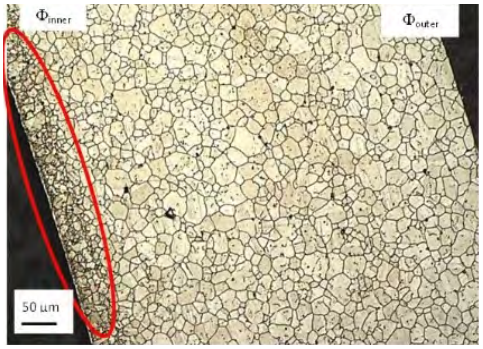
The CEA characterized a PAVIX-8 axially heterogeneous pin irradiated in the Phenix SFR at intermediate linear heat rate (LHR) to extend the validation basis of the GERMINAL V2 fuel performance code. Compared to high LHR irradiated fuels, major differences resulting from the lower fuel operating temperature have been observed. The GERMINAL V2 code underestimated the fuel swelling because this code does not consider gaseous swelling. The implementation of a new fuel swelling model in the GERMINAL code is underway.

The JAEA developed a plutonium and uranium mixed oxide powder adhesion prevention technology, applying nanoparticle coating on the acrylic panels

Figure SFR-8. Helium transport and release behavior in $(U,Am)O_2$



Cladding tubes made of 15-15Ti AIM1 steel




Grain size reduction on inner side of cladding tubes due to nitrogen contamination

**Figure SFR-9. Characterization of the 15-15Ti AIM1 cladding tubes**

of the glove box to minimize retention of nuclear fuel materials in glove box components and curtail the external exposure dose.

Rosatom manufactured three experimental nitride fuel assemblies for the BN-600 reactor.

### *WP2.2: MA-bearing transmutation fuel evaluation, optimization and demonstration*

Euratom has also contributed to the work of MA-bearing oxide fuel performance evaluation. (U,Am) $O_2$ mixed dioxides are promising candidate fuels for the transmutation of americium (Am) in fast reactors in the heterogeneous recycling concept. One of the major differences in the irradiation performance of these fuels, compared to conventional MOX or uranium dioxide ($UO_2$), is their large He production, which can have a significant impact on safety-related phenomena, such as fuel swelling and pressure build-up inside the fuel pin. However, knowledge of its behavior in fuel is limited. Therefore, separate effect tests were performed on He generated in situ by alpha decay in (U,Am) $O_2$, and He introduced by ion-beam bombardment in $(U,La)O_2$ simulant materials. He transport and release mechanisms were then investigated by Knudsen Cell effusion mass spectrometry for both sample types, with complementary experiments on their microstructure evolution by transmission electron microscopy (TEM).

The JAEA is developing a simplified pelletizing process for MA-bearing MOX fuel fabrication. As part of this project, the JAEA evaluated performance of the granulation system in actual scale with simulated powder, which is composed of modernized a wet granulator, sizing machine, dryer and other auxiliary equipment.

KAERI developed metal fuel for the prototype Gen-IV sodium-cooled fast reactor (PGSFR). The fuel assembly was designed to satisfy requirements for the core performance and safety. The PGSFR fuel assembly consists of the handling socket, upper/lower reflector, hexagonal duct, fuel rods and nose piece. The structural characteristics and design features of PGSFR fuel assembly and its components have been described.

KAERI also analyzed the interaction between casting parts and U-10 wt.% Zr alloy containing rare-earth (RE) elements through the sessile drop test. Candidates for alternative crucibles and moulds was demonstrated using a casting of the U-Zr-RE alloy. Interaction behaviors and defects of the casting parts were evaluated after casting.

Rosatom developed technological processes for the manufacture of americium-burning elements within the framework of the "heterogeneous" scenario of the nuclear fuel cycle closing (NFC). Rosatom manufactured an experimental batch of mixed nitride of uranium and plutonium (MNUP) fuel samples through the method of high-voltage electric pulse consolidation and control of their characteristics.

### *WP 2.3: High-burn-up fuel evaluation, optimization and demonstration*

The CIAE will conduct a program to do some CN-1515 and CN-FMS material irradiation tests in the CEFR in the coming years. The R&D and fabrication of CN-1515 and CN-FMS has been completed. The design of the irradiation rig and the fabrication of irradiation assemblies has also been finished. The mechanical properties of CN-1515 and CN-FMS have been tested and have been used to evaluate the irradiation assemblies.

The CEA has characterized the cladding tubes of 15-15Ti AIM1 from two different fabrication routes. The AIM1 is a titanium (Ti)-stabilized austenitic stainless steel treated as the reference cladding material for Phenix and ASTRID SFRs. In the results of the glow discharge mass spectrometry (GDMS), nitrogen contamination is observed on the inner surface of the cladding tubes of one of the two batches. The CEA carried out tensile property measurements on the two batches. Based on these results and past experience with AIM1 cladding irradiated in Phenix, the CEA concluded that a good behavior in pile can be foreseen for these AIM1 cladding tubes.

The JAEA carried out high- and ultra-high-temperature creep rupture tests, internally pressurized creep rupture and ring rupture tests, and temperature-transient-to-burst tests of 9Cr-ODS steel claddings. For comparison, the

transient burst strength of 11Cr-ferritic/martensitic steel (PNC-FMS) cladding was also evaluated. The obtained data was used to investigate the applicability of the life fraction rule to rupture life prediction of 9Cr-ODS steel and PNC-FMS claddings in various load-time temperature histories.

KAERI developed technology for a barrier cladding tube to suppress fuel-cladding chemical interaction (FCCI) for the use of MA-bearing metal fuel. Cr plating was applied at the inner surface of the cladding tube to achieve 20 μm thickness of Cr at the 500 mm length of HT9 cladding. Optimization of Cr plating to enhance layer property, such as pulse plating and surface treatment through nitriding process, has been reported.

Rosatom developed and manufactured BN-600 irradiation assemblies for testing fuel elements up to extreme parameters.

## Component design and balance-of-plant project

### *ISI&R technologies*

The CEA has studied the capability of the leaky Llmb waves in view of inspection from the outside of the main vessel. In 2020, the CEA conducted this experiment using devices that consisted of several austenitic steel plates immersed in water, an ultrasonic emitter and receiver. The experimental results with one plate were compared to simulation results.

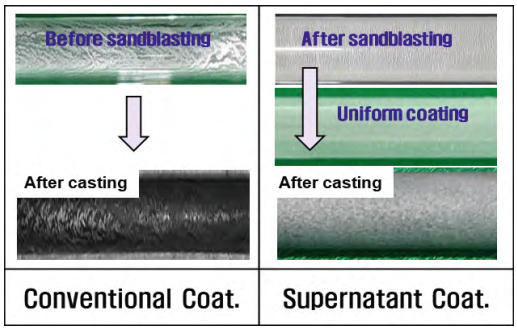



Figure SFR-10. Development of coating technology for the development of reusable mould

KAERI demonstrated the performance of the plate-type ultrasonic waveguide sensor in a sodium environment. KAERI fabricated under-sodium waveguide sensors and then conducted several under-sodium tests for viewing and ranging performance verification.

The JAEA performed the imaging test under-sodium viewer for medium distance in a sodium environment, and a performance test for long distance in actual plant configuration.

### *Supercritical $CO_2$ Brayton cycle*

The CEA investigated the development of sensors to examine this heat exchanger. For this, eddy current probes were developed after defining the

Figure SFR-11. Under-sodium performance demonstration of the ultrasonic waveguide sensor

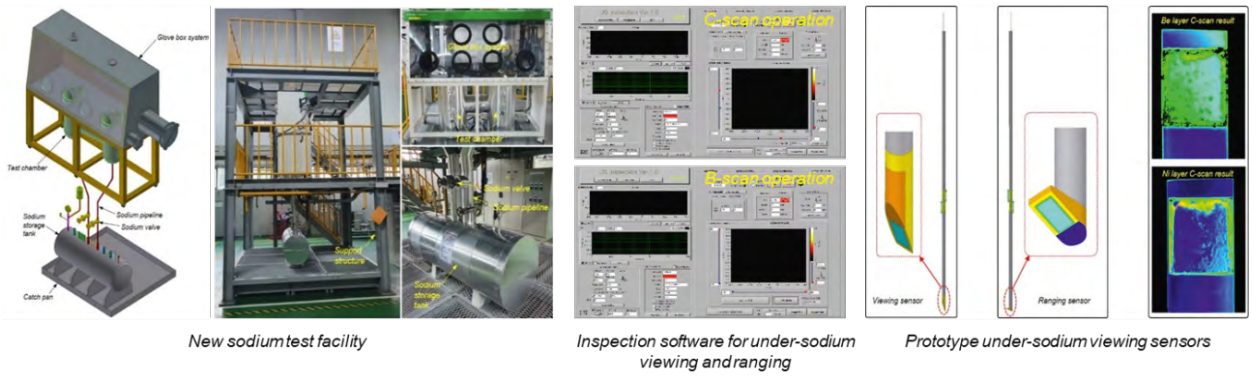



Figure SFR-12. Imaging experiments in sodium environment

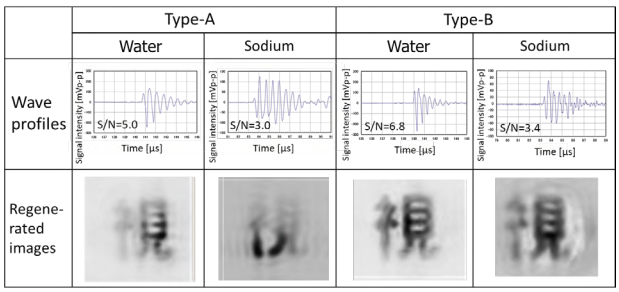



Figure SFR-13. Eddy current technique for NDT within small channels

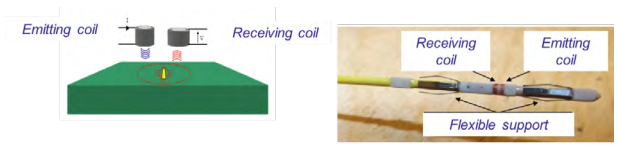


Figure SFR-14. Sodium-water reaction in a semi-open volume

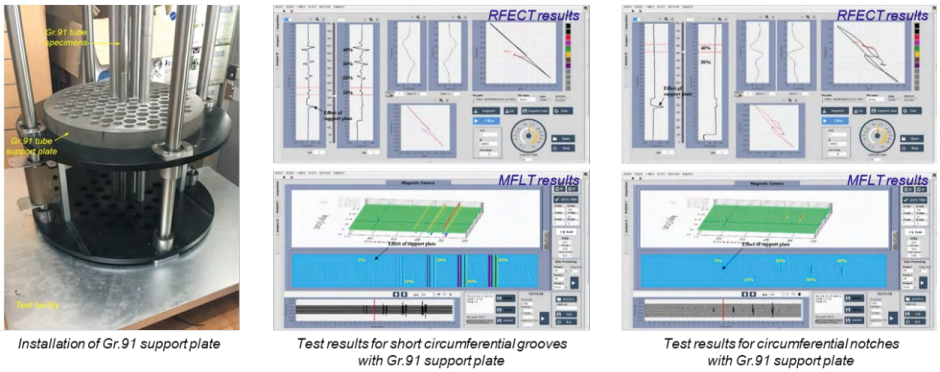



Figure SFR-15. Inspection system for a single-walled tube of a Rankine-type steam generator

specifications. A very small (3 mm by 3 mm, as an order of magnitude) probe was designed and manufactured.

The ANL prepared a report summarizing what has been learnt about sodium-$CO_2$ interactions and sodium-$CO_2$ reaction products based mainly upon data and results from the SNAKE sodium-$CO_2$ interaction experiments carried out at ANL.

### *Sodium leakages and consequences*

No specific activity was conducted this year in this work package.

### *Steam generators*

The CEA started a new activity in the field of the sodium-water reaction. It studied the sodium-water reaction (SWR) in specific conditions, such as open or semi-open volume (see Figure SFR-14). For this, after a review of the existing knowledge, the CEA defined and performed SWR in a dedicated facility (MININANET). The first experimental campaign results were then presented.

The JAEA performed the investigation on the applicability of mechanistic sodium-water reaction analysis code for steam generator performance and safety evaluation.

KAERI continued its performance demonstration of the upgraded prototype combined steam generator tube inspection system. More specifically, KAERI investigated the effects of the tube support plate and neighbouring tubes on measured remote field eddy current testing (RFECT) and magnetic flux leakage signals.

### *Sodium operation technology and new sodium testing facilities*

KAERI finished constructing the sodium thermal-hydraulic integral effect test facility (STELLA-2), and completed its shake-down test and start-up operation with liquid sodium. A couple of sets of sodium integral effect test databases were collected and were used for some computational verification and validation (V&V) codes. KAERI continued the study of specific sodium technologies to get some useful measurements of process variables,

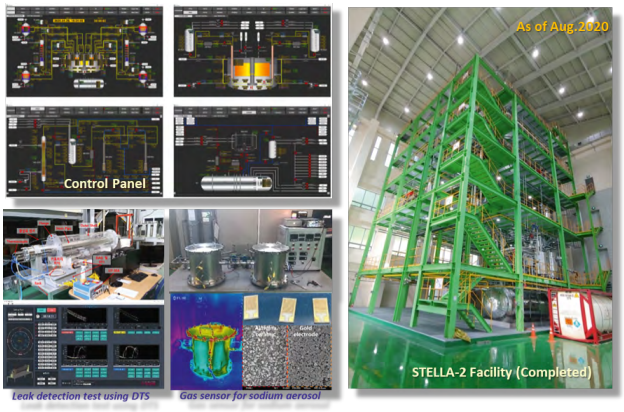



Figure SFR-16. Progress of the sodium integral effect test (STELLA-2) and development of advanced instrumentation techniques for liquid sodium applications

as well as for better operation of large-scale sodium facilities covering sodium feeding, draining, accident prevention, high-temperature operation and measurements.

The DOE conducted the design and construction of an intermediate-scale sodium test facility for the purpose of testing systems and components (mechanisms) in prototypical sodium environments. This facility consists of four experimental test vessels (of two sizes). Sodium has been fed to these test vessels from a main loop. The test vessels were designed to provide an independent testing environment, if isolated from the main loop. In addition, the test vessels were designed to allow for independent draining to the main dump tank without impacting the sodium environments in the other test vessels.

**Frédéric Serre**
Chair of the SFR SSC, with contributions from SFR members

## 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.

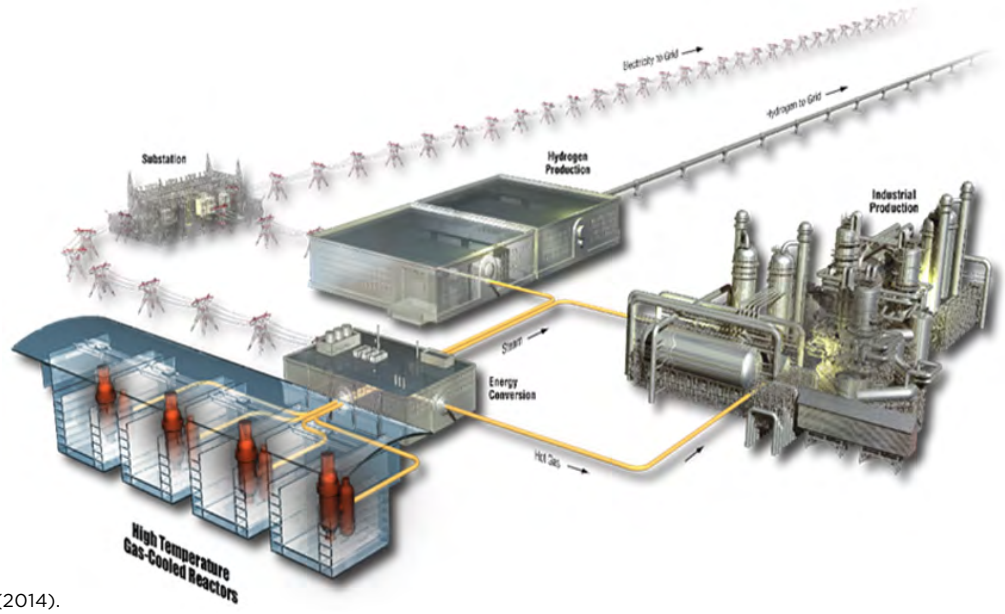
Cogeneration of heat and power makes HTRs and VHTRs attractive heat sources for big industrial complexes, such as chemical plants, to substitute large amounts of process heat at different temperatures, which are today produced by fossil fuels. Depending on the coolant outlet temperature, such reactors 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

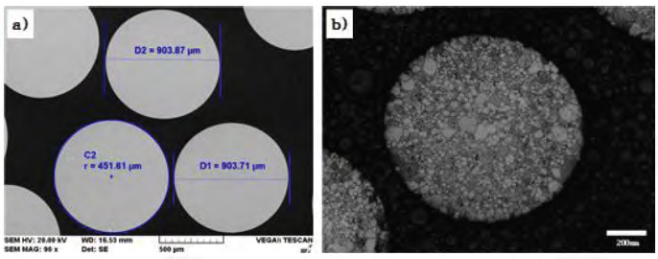
Work to develop advanced TRISO fuels, in some cases involving varying particle designs or new coating materials, is in progress in several member countries. In Korea, KAERI has been working to develop fabrication methods for $UO_2$ kernels with diameters significantly larger than the conventional 500 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.

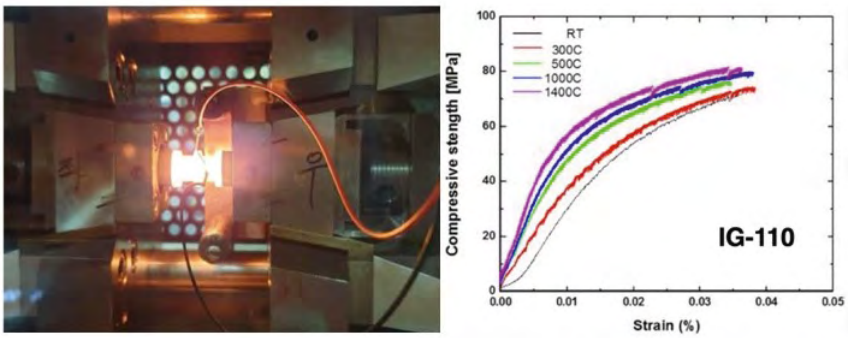
**Figure VHTR-2. KÜFA furnace installed in INET hot cells**

Liu, B., (2020).

**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).

**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.

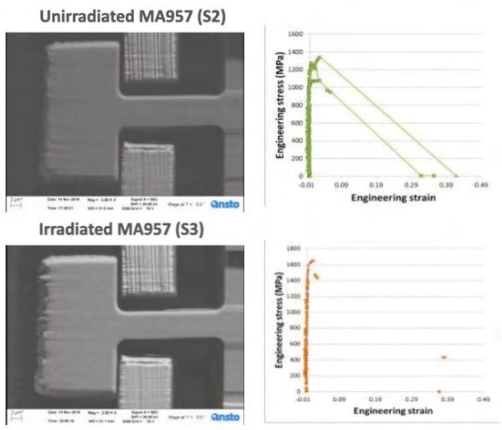
A detailed flowsheet analysis has also been carried out, and the ancillary components required for rendering the process a closed cycle were defined. Optimization of the process with respect to operating expenses (OPEX) and capital expenses (CAPEX) is being undertaken to fully understand their impact 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.

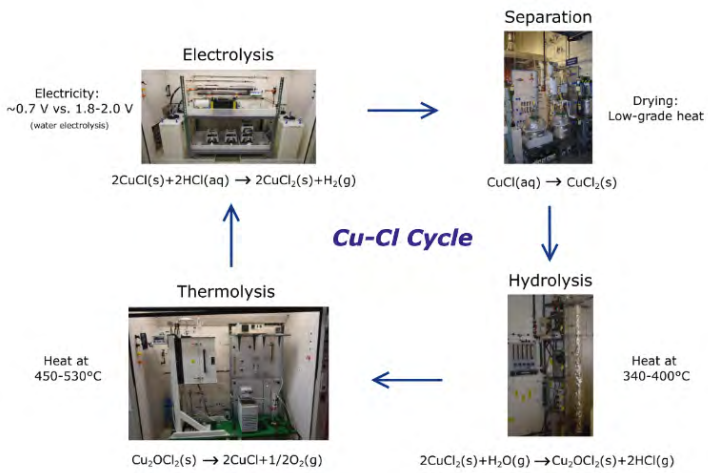
With the announcement of an ambitious recovery plan of EUR 7 billion over a ten-year period,

**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**



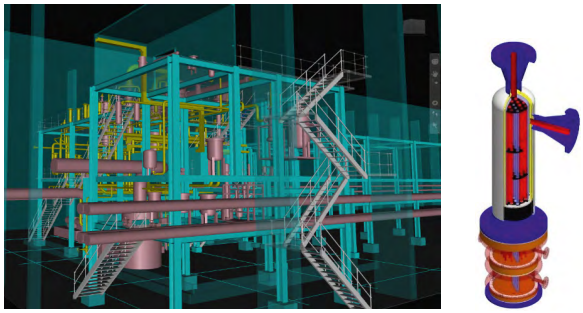
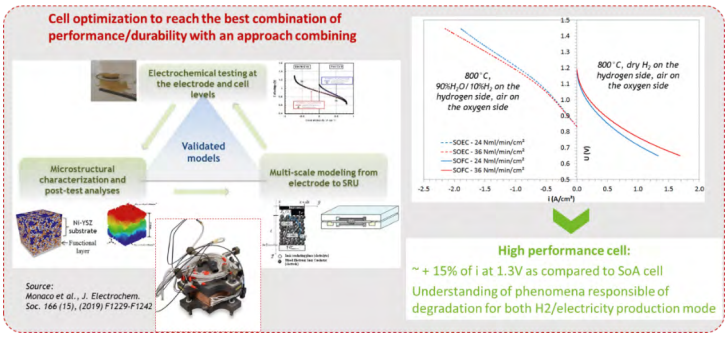


Figure VHTR-7. Sulphuric-acid and hydriodic-acid decomposition facilities and the high-temperature helium loop




Monaco, F. et al. (2019).

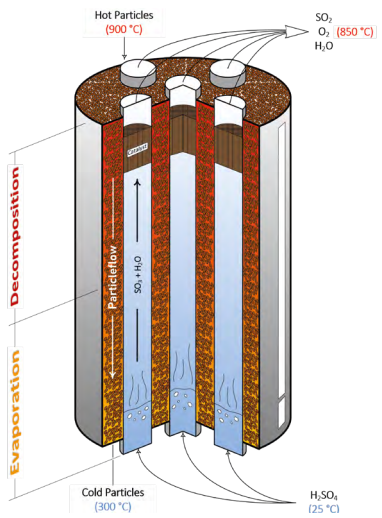
Figure VHTR-8. CEA activities to develop high-temperature steam electrolysis

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



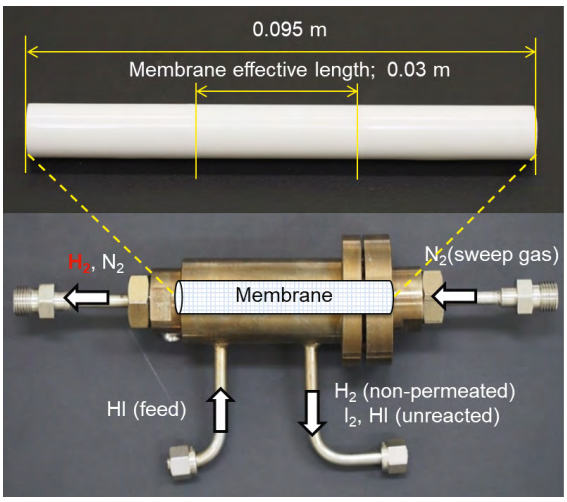

Thanda, V. K. et al. (2020).

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**

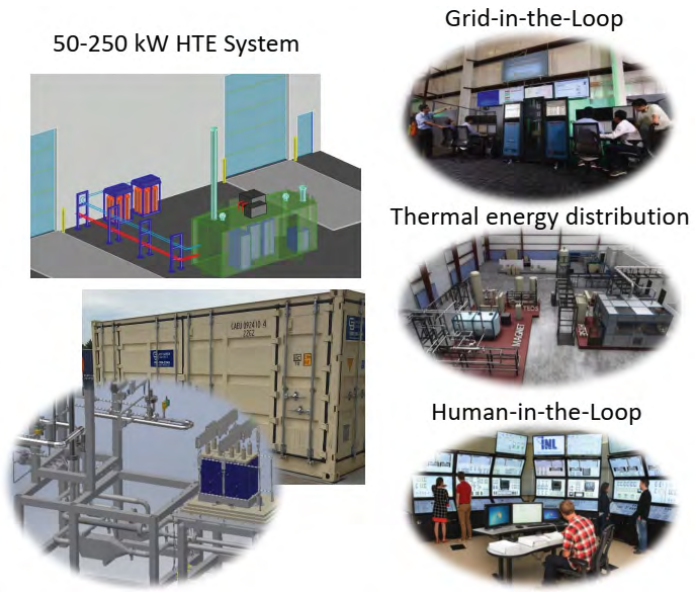



Myagmarjav, O. et al. (2019).

**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.

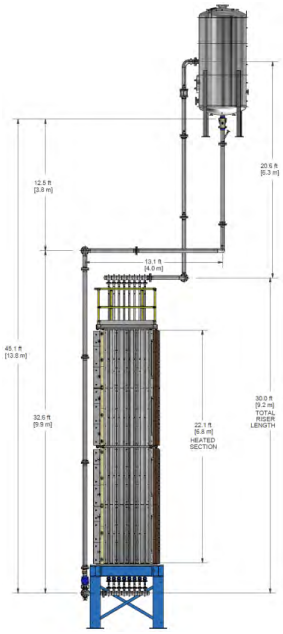
Table VHTR-1. Work Package titles of the VHTR CMVB PMB

| 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).

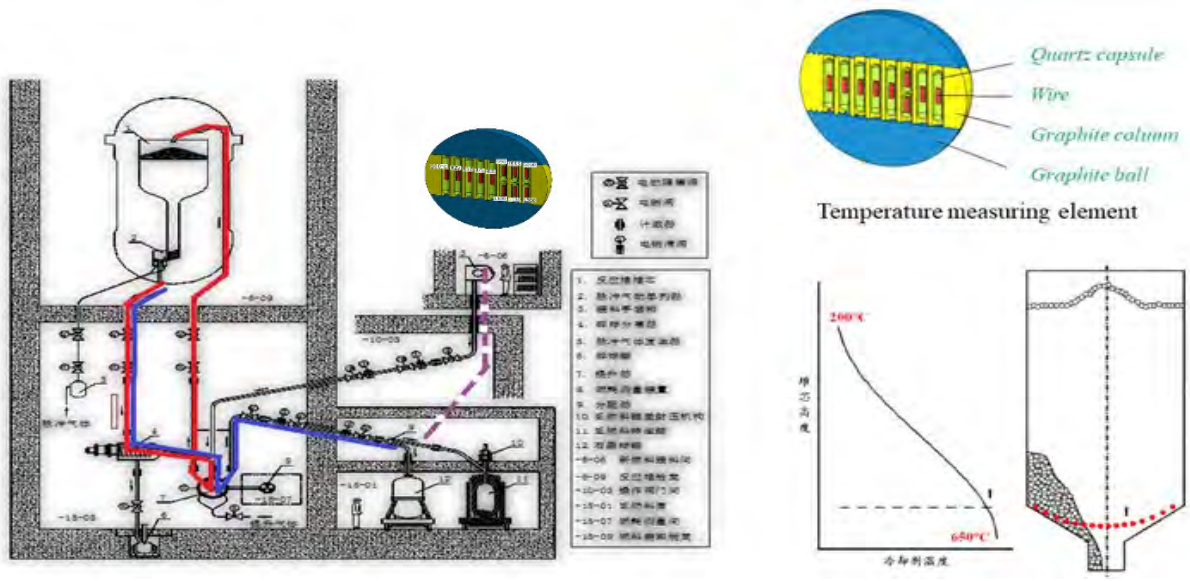


Zhang, Y. et al. (2021).

**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