3 System summaries

Gas-cooled fast reactor

Signatories of the System Arrangement for collaboration on gas-cooled fast reactor (GFR) research and development are the Generation-IV International Forum (GIF) members: Euratom, France and Japan. Two technical projects have been established for GIF collaborations:

- GFR conceptual design and safety, with the Joint Research Centre (JRC) and French Alternative Energies and Atomic Energy Commission (CEA) as members;
- GFR fuel, core materials and fuel cycle, with the JRC, CEA, and the Japan Atomic Energy Agency (JAEA) as members.

The second project is newly formed, and the Project Arrangement is expected to be signed in 2022.

Main characteristics of the system

The 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 being proposed as a longer-term alternative to liquid metal cooled fast reactors. The main advantages of GFRs, in addition to enabling the adoption of a closed fuel cycle, are:

- high operating temperature, allowing increased thermal efficiency and high-temperature heat for industrial applications similar to the VHTR;
- a chemically inert and non-corrosive coolant (helium);
- a single phase (no boiling) coolant (helium);
- relatively small (albeit positive) helium coolant void reactivity coefficient;
- the absence of dissociation or activation of helium,
- the transparency of helium, which facilitates in service inspection and repair, as well as fuel handling.

The reference concept for the GFR is a 2 400 megawatts thermal (MWt) 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 core is made up of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fueled pins contained within a ceramic hextube. The high 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 favored material for the pin clad and hextubes is currently silicon carbide fiber reinforced silicon carbide (SiCf/SiC). The entire three-loop primary circuit is contained within a secondary pressure boundary, the guard containment. The heat produced 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.

Technical highlights – conceptual design and safety project

The demonstration of the GFR technology assumes that the basic features of the GFR commercial reactor can be tested in the 75 MWth European Gas Fast Reactor Demonstrator Project (ALLEGRO). The objectives of the ALLEGRO GFR demonstration project are to illustrate the viability of, and 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.

Four nuclear research institutes and companies (i.e. ÚJV Řež, a.s., Czech Republic; the Centre for Energy Research, Hungarian Academy of Sciences, Hungary; the National Centre for Nuclear Research (NCBJ), Poland; and VUJE, a.s., Slovak Republic) in the Visegrád-Four region have decided to start joint preparations aiming at the construction and operation of the ALLEGRO demonstrator for the Gen-IV GFR concept, based on a memorandum of understanding (MoU) signed in 2010. The CEA, as the promoter of the GFR concept since 2000, supports these joint preparations, and it is bringing its knowledge and its experience to building and operating experimental reactors, in particular fast reactors.

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

ALLEGRO main characteristics	
Nominal Power (thermal)	75 MW
Driver core fuel/cladding	MOX(UO2) / 15-15ti Steel
Experimental fuel/cladding	UPuC / Sic-Sicf
Fuel enrichment	35% (MOX) / 19.5% (UO2)
Power density	100 MWth/m3
Primary coolant	He
Primary pressure	7 MPa
Driver core in/out temperature	260°C / 530°C
Experimental fuel in/out T	400°C / 850°C

Notes: MOX = mixed oxide (fuel); UPuC = mixed carbide (fuel); MPa = megapascal; UO_2 = uranium dioxide.

Source: 31st GRF SSC meeting, ALLEGRO overview, 2021.10.

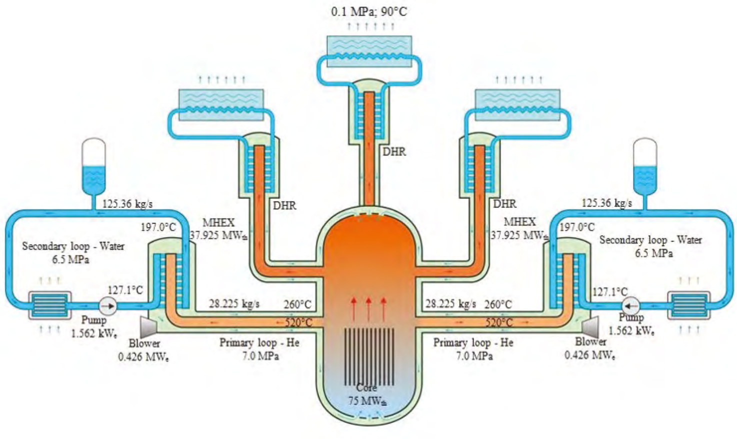


Figure GFR-1: ALLEGRO design overview

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 MOX fuel in stainless steel claddings will serve as a driving core for six experimental fuel assemblies containing the advanced carbide (ceramic) fuel, and the second core will consist solely of the ceramic fuel, enabling operation of ALLEGRO at the high target temperature.

Core design optimization requires a multidisciplinary approach. A core should be defined to satisfy the requirements of the performance (irradiation capabilities) and safety features (reactivity feedback coefficients), and safety analyses must prove the fulfillment of the acceptance criteria. For this, iterative neutronics, thermal hydraulics and coupled neutronics/thermal-hydraulics analyses are needed.

First, MOX and UOX cores were defined based on neutronics analyses, and performance and safety features were optimized. Limiting (enveloping) initiating events were selected, which were analyzed for all of the defined reactor cores.

Development of new calculation models and methodologies, and the extension and improvement of calculation tools were needed to achieve the above goals.

The SafeG project

The SafeG project has received funding from the Euratom Horizon 2020 Nuclear Fission and Radiation Protection Research program (NFRP-2019-2020-06). 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-carbon dioxide (CO_2) electricity and the industrial process heat generation technology through the following main objectives:

- strengthen the safety of the GFR demonstrator ALLEGRO;
- review the GFR reference options in materials and technologies;
- 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 the safety and proliferation resistance points of view;
- attract students and young professionals, boosting interest in GFR research;
- expand collaboration with international, non-European (i.e. EU) research teams and with relevant European 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 analyses 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.

Technical highlights – fuel, core materials and fuel cycle project

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 forms of fuel, 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.

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 made 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 the 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 fiber and C fiber 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 boron nitride (BN) particle dispersed SiC is candidate matrix material for the high-dose irradiation tolerant SiC composites. The BN particle dispersed SiC was irradiated with the reference chemical vapor deposition SiC at 800°C and up to 10 displacements per atom (dpa) under the SafeG project at the Dual-Beam Facility for Energy Science and Technology (DuET), Kyoto University. Significant differences in swelling were not observed. Regarding the behavior of the SiC brazing agent under irradiation, results of the CROCUS irradiation, performed in the OSIRIS MTR (CEA), have been shared. Several samples consisting of mini composites with brazed joint have been irradiated at dose levels ranging from 1 to 3 dpa. Some of them have been subjected to mechanical solicitation up to 150 MPa. Overall, the braze behaves well under irradiation. No creep has been observed on the samples subject to 150 MPa.

A key element of GFR developments is the introduction, testing and qualification of appropriate refractory fuel that can reliably withstand the high-temperature and high-dose conditions in the reactor for long periods and allow for the safe operation of the reactor.

The ALLEGRO concept as a European GFR demonstrator unit assumes that the “final” refractory (ceramic) fuel system design will finish its qualification process by irradiation in several experimental positions in the starting ALLEGRO core(s). The starting ALLEGRO core will consist of already well qualified fuel assemblies. The obvious shortcoming of such an idea is that there is no well-qualified fuel system for the GFR system because no GFR has ever been operated (and, as stated above, the fuel system qualification is product specific). The closest option is the reference French sodium-cooled fast reactor (SFR) design – a bundle of thin steel (15-15Ti) clad rods with UO_2/MOX fuel in the form of pellets spaced by a helically wound wire within a EM10 ferritic-martensitic steel hexagonal wrapper tube. From the fuel behavior point of view, the main differences between the SFR and GFR under normal operating conditions are:

- Higher desired outlet temperatures in the GFR, at about 850°C in the GFR compared to SFR temperatures less than 620°C. To overcome this difference, the outlet temperature of the steel-clad ALLEGRO core would be reduced in order to fit within the 15-15Ti qualification range. Note that three of the main phenomena of 15-15Ti behavior – swelling, creep and fuel/cladding chemical interaction – would have to be investigated beyond the SFR range to ensure reliable operation at higher temperatures.

- Higher system pressure, at about 7.5 MPa in the GFR compared to atmospheric pressure in the SFR, leading to inward cladding creep in the GFR as opposed to outward cladding creep in the SFR. Nonetheless, the creep behavior of 15-15Ti in the SFR temperature range is well-known, and taken into account in the fuel pin design.
- In ALLEGRO more specifically, the power density will be lower than in the reference SFRs, leading to much lower fuel center-line temperatures and hence reduced fission product migration and release, and fuel restructuring.
- Flow induced vibrations will be different in the GFR. The impact of this difference is nevertheless unknown.
- Coolant-cladding interactions will differ (in the case of the GFR, the main issue is impurities in the He coolant).
- The coolant volume in the core compared to the fuel volume is about 50% higher in the GFR compared to the SFR.

Considering the above information, the idea of using a fuel system based on the reference SFR fuel seems well founded, especially considering its proven manufacturing process, including the quality assurance on a semi-industrial scale.

A limited post-irradiation examination of the fuel would still be necessary to confirm the expected behavior (particularly the chemical interactions with the impurities in the coolant). The core power would have to be adjusted to keep the cladding and wrapper temperatures in the SFR range, but several fuel assemblies could be operated at reduced He inlet flow, and hence increased temperatures, in order to widen the experience base. The coolant flow rate would also be adjusted in the experimental positions containing the novel refractory fuel assemblies in order to reach the desired temperatures.

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

Lead-cooled fast reactor

Participants in the GIF MoU for collaboration on lead-cooled fast reactor (LFR) research and development are the GIF members: the People's Republic of China (hereafter "China"), Euratom, Japan, Korea, the Russian Federation (hereafter "Russia") and the United States. This section highlights the main collaborative achievements of the GIF LFR provisional System Steering Committee (pSSC) to date. In addition, highlights are summarized for the development of LFRs in GIF member countries and entities, as shared within the GIF collaboration.

Main characteristics of the system

Gen-IV LFR concepts include three reference systems: 1) a large system rated at 600 megawatts electric (MWe) – the European lead fast reactor (ELFR), intended for central station power generation; 2) a 300 MWe system of intermediate size – the Russian BREST-OD-300; and 3) a small, transportable system of 10-100 MWe in size – the US small secure transportable autonomous reactor (SSTAR), which 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 from small and intermediate to large sizes. Important synergies exist among the different reference systems, with one of the key elements of LFR development being the co-ordination of efforts carried out among participating countries.

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 would be deployed by 2040. Following the reformulation of the GIF-LFR pSSC in 2012, the SRP was 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 MoU status to a system arrangement organization is engaged.

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

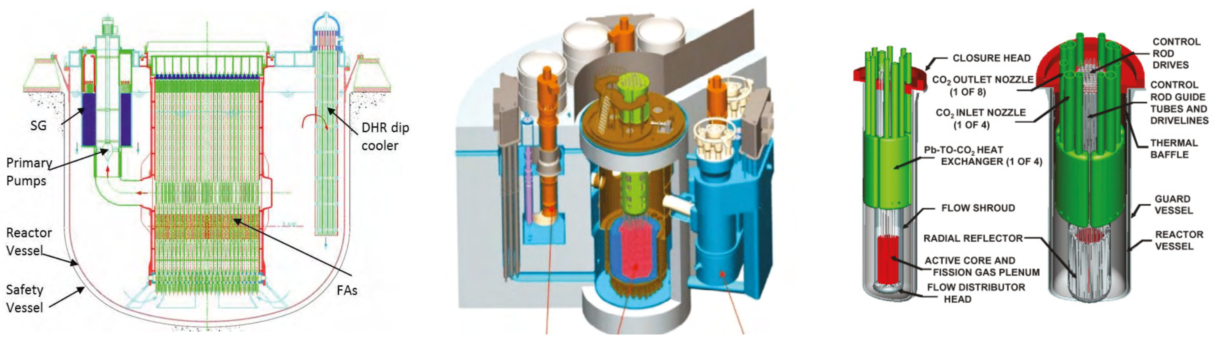
Parameters	ELFR	BREST	SSTAR
Core power (MWt)	1 500	700	45
Electrical power (MWe)	600	300	20
Primary system type	Pool	Pool	Pool
Core inlet T (°C)	400	420	420
Core outlet T (°C)	480	540	567
Secondary cycle	Superheated steam	Superheated steam	Supercritical CO_2
Net efficiency (%)	42	42	44
Turbine inlet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

Technical highlights – provisional System Steering Committee activities

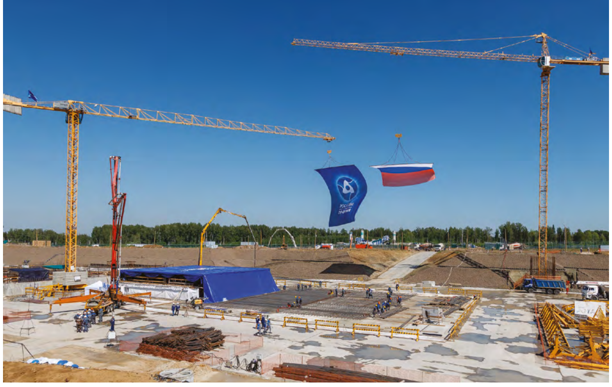
The pSSC has been very active on the following main lines of work in 2021:

- *GIF Lead-cooled Fast Reactor: Proliferation Resistance and Physical Protection White Paper* (GIF, 2021) – The report, prepared jointly by the GIF Proliferation Resistance and Physical Protection Working Group (PRPPWG) and the GIF-LFR-pSSC, follows the high-level paradigm of the GIF PRPP evaluation methodology to investigate the PRPP features of the three GIF LFR reference systems. The three systems share the following main technological features: a large reactor (i.e. ELFR) as a potential replacement of current large Generation-III reactors, a medium-sized reactor (i.e. BREST-OD-300) representative of the upper bound of what is accepted in the small modular reactor (SMR) power range, and a sealed-core (i.e. SSTAR), long-life microreactor expressly designed to provide enhanced nuclear proliferation resistance.

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

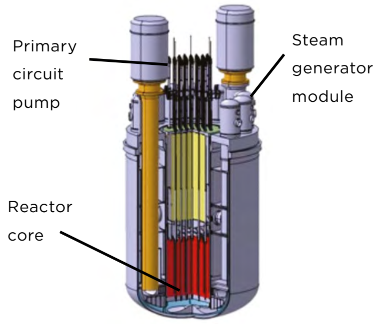


Source: Alemberti, A. et al. (2018).



Source: http://proryv2020.ru/en/photo/pervyj-beton-reaktora-brest-od-300-proekta-proryv.

Figure LFR-2. Ceremony to start pouring concrete for the BREST-OD-300 power unit



Source: www.akme-engineering.com/svbr.html.

Figure LFR-3. General view of the SVBR-100 reactor module

- *Safety Design Criteria for Generation IV Lead-Cooled Fast Reactor System* (GIF, 2021) – The GIF-LFR-pSSC decided to develop a set of reference criteria for the safety design of the structures, systems and components of LFR systems with the aim of achieving the safety goals of the Generation-IV reactor system. A set of eighty-two (82) reference criteria for LFRs are thus systematically and comprehensively explained in the safety design criteria (SDC). The report was prepared in collaboration with the GIF Risk and Safety Working Group (RSWG).

Interaction between the GIF-LFR-pSSC and the Working Group on Safety of Advanced Reactors (WGSAR) has continued in 2021 in an effort to define the benchmark specifications for LFR codes validation. During joint meetings between GIF-LFR-pSSC and WGSAR in 2021, benchmark tasks, including fuel assembly blockage and reactivity insertion scenarios, were discussed. The objective is to identify regulators' expectations for those codes and highlight possible knowledge gaps. The outcomes will be used as a basis to initiate further research on analytical codes and methods.

National LFR demonstration and development highlights

In Russia, the innovative lead-cooled fast reactor BREST-OD-300 is being developed as a pilot demonstration prototype of basic commercial reactor plants for future nuclear power with a closed nuclear fuel cycle (NFC). The target requirements for commercial reactors were defined as follows: 1) removal of restrictions on fuel resources given the efficient use of uranium raw materials during multiple recycling of nuclear fuel in the closed NFC; 2) exclusion of severe accidents at nuclear power plants with radiation consequences requiring evacuation and, especially, resettlement of the population; 3) technological enhancement of the nuclear non-proliferation regime; 4) closing of the NFC with the disposal of radioactive waste in a hazard-equivalent state in relation to the uranium raw material from which it was produced; and 5) economic competitiveness with other energy sources.

BREST-OD-300 received its construction license from Rostekhnadzor in February 2021. The construction of the nuclear power plant with the BREST-OD-300 lead-cooled fast reactor as a part of the Pilot Demonstration Energy Complex began on 8 June 2021 (see Figure LFR-2).

As part of the concept of small modular reactors, a project for the SVBR-100 reactor plant is being developed in Russia (see Figure LFR-3 and Table LFR-2). It is planned to build a pilot power unit with such a reactor at the site near Dimitrovgrad city. The choice of power level (100 MW) is to support its transportability by rail, which would ensure a fully factory manufactured reactor with high quality. Mastered parameters of the coolant ensure the use of traditional stainless steel (18% chromium [Cr], 9% nickel [Ni]) for the reactor vessel. Serial 4-6 unit (or more) nuclear power plants with modular reactors will be focused on a wide range of tasks from regional energy and heat supply to desalination of sea water or hydrogen production.

Table LFR-2. Main characteristics of the SVBR-100 reactor plant

Parameter	Value
Thermal power, MW	280
Generated steam pressure, MPa	7
Steam generating capacity, t/hour	580
Primary circuit coolant	44.5% Pb + 55.5% Bi
Primary coolant temperature, at core inlet/outlet, °C	335/477
Fuel: Medium enrichment on U-235, % Maximum enrichment on U-235, %	UO_2 16.7 19.5
Core lifetime, thousands effective hours (h)	50
Refueling after (once through), years	6-7
Reactor dimensions (diameter/height), m	4.40/6.53

In Japan, experimental studies on chemical compatibility in liquid lead alloys were performed at the Tokyo Institute of Technology. The corrosion resistance of iron chromium aluminum (FeCrAl) alloys in liquid metals was studied. Al-rich oxide layers functioned as an anti-corrosion barrier in the liquid metals. The self-healing behavior of the oxide layers formed on structural materials was also studied by means of in-situ electrochemical impedance spectroscopy. The corrosion studies with additive manufactured materials were performed in collaboration with a Japanese metal processing company.

The main activities in Europe related to liquid metal technologies are centered on three 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 technology, also supporting the development of fast-neutron spectrum Generation-IV systems; 2) R&D activities for the construction of an LFR demonstrator in Romania, i.e. the Advanced Lead Fast Reactor European Demonstrator (ALFRED) project (see Figure LFR-4); and 3) R&D activities carried out in the United Kingdom in collaboration with several EU organizations in the framework of the Advanced Modular Reactor (AMR) Program (Phase 2), which supports the development of the Westinghouse LFR concept.

In parallel, several ongoing European collaborative projects are ongoing (EURATOM co-funded initiatives, i.e. GEMMA, PIACE, PASCAL, PATRICIA, PUMMA) that are dedicated to heavy liquid metal technology, development and validation of numerical tools and safety assessments, as well as material and fuel development and qualification. These Euratom R&D projects are complemented by the R&D work conducted by the European Commission's JRC. In 2021, the latter included contributions to the PATRICIA project (scanning electron microscopy analysis of LBE samples with various concentrations of tellurium) and the GEMMA project (mechanical tests in liquid lead at the Liquid Lead Laboratory). In addition, a new European player promoting LFR development, Newcleo, is currently creating an international nuclear research center based in Turin (Italy). Newcleo took over the reactor design activities that had been previously performed by Hydromine. As such, Newcleo also owns a significant number of patents related to LFR technologies.

An important milestone achieved in 2021 was the beginning of construction of the ATHENA facility in Mioveni, Romania, by the Institute of Nuclear Research of the Technologies for Nuclear Energy State Owned Company (RATEN). ATHENA is a pure lead integral pool-type facility, with a 2.21 MW electrically heated core simulator and a main vessel of 3.2 m in diameter and 10 m in height.

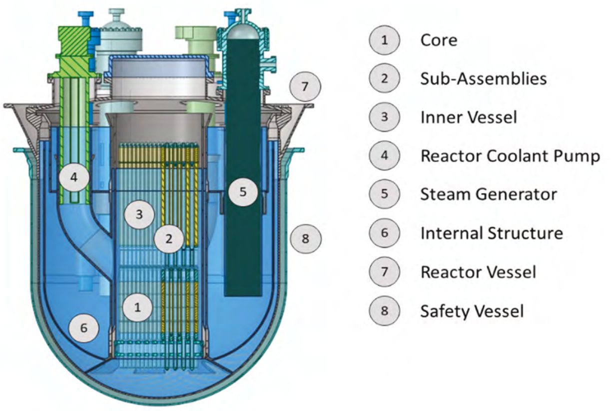
In the United Kingdom, the AMR Program is ongoing. It is funded by the UK Department of Business, Energy and Industrial Strategy and supports LFR development with GBP 10 million (British pounds). The program, led by Westinghouse, is supported by Ansaldo Nucleare (Italy), the Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA, Italy), Jacobs (United Kingdom), Ansaldo Nuclear (United Kingdom), the University of Manchester (United Kingdom), University of Bangor (United Kingdom), the Nuclear National Laboratory (United Kingdom), the Nuclear Advanced Manufacturing Research Centre (United Kingdom) and other UK partners. The program is devoted to material and coolant chemistry, technology and fuel development, code validation, innovative manufacturing and supply chain implementation. A number of new facilities are under construction to support the program.

In Korea, the primary momentum of LFR development has been the Ulsan National Institute of Science and Technology when the government funded a conceptual design project for a factory fueled non-refueling micro LFR for maritime applications. The Korean government has been funding international collaborative R&D to further upgrade its small LFR concept – the "ubiquitous, rugged, accident-forgiving, non-proliferating, and ultra-lasting sustainer" (URANUS) – into a microreactor design called MicroURANUS (see Figure LFR-5). The latter is being optimized for maritime applications with a 40-year lifespan without refueling.

In the United States, the US Department of Energy (DOE) has recently sponsored several LFR projects at universities. These include the following ongoing efforts:

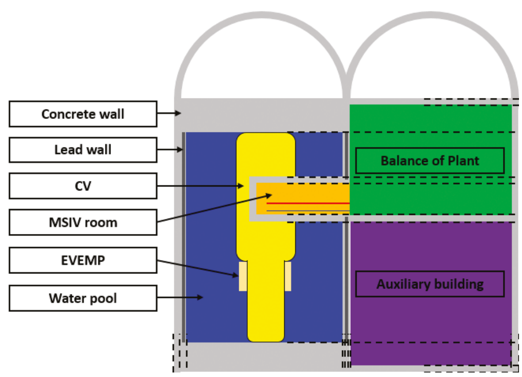
- A project led by the Massachusetts Institute of Technology (MIT) in the area of corrosion/irradiation testing in lead and LBE. The project seeks to investigate the "radiation decelerated corrosion hypothesis", relying on simultaneous exposure.

Figure LFR-4. ALFRED primary system configuration

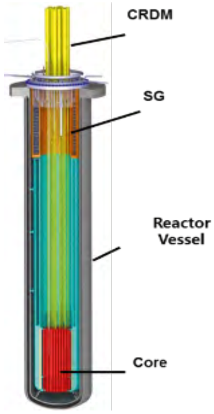


Source: Alemberti, A. et al. (2020); Frignani, M. et al. (2019).

Figure LFR-5. General layout of MicroURANUS 2.5



Source: Hwang, I.S. et al. (2021).



Source: Zhao, Z. et al. (2019).

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

Source: IAEA LMFNS database.

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

- A project 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.
- A project at the University of New Mexico to experimentally investigate the integral effects of radioisotope interactions with liquid lead, establishing a basis for liquid metal radioisotope retention.
- A project at the Rensselaer Polytechnic Institute to improve the accuracy of the neutronics simulation of lead-based systems by improving the nuclear data of lead isotopes.
- A project at the Rensselaer Polytechnic Institute to address critical issues with the compatibility and chemical interactions of uranium nitride fuel, alumina-forming austenitic alloys and lead coolants/sublayers.
- A project at the University of Wisconsin to design and model a renewable, nuclear integrated energy system for co-generation of cost competitive electricity and clean water, with LFR as one of the heat source options.

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 microreactor spin-off concepts identified as the LFR-TL-X series (where T refers to transportability, L refers to the long-lived core, and X is a variable identifying power options ranging from 5 to 60 MWe). During 2021, the Hydromine design activities were transferred to a newly incorporated European entity, Newcleo, with activities in Italy and the United Kingdom.

In China, the China lead-based mini reactor (CLEAR-M) project, with the 10 MW-grade CLEAR-M10 as a representative concept for 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, for example in the case of islands, remote districts or industrial parks. In order to support 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 prototype component proof tests. In addition, three integrated test facilities have been built and were commissioned in 2017, leading to the lead-based engineering validation reactor CLEAR-S, the lead-based zero power critical/subcritical reactor CLEAR-0, coupled with the High Intensity D-T fusion Neutron Generator (HINEG) for reactor nuclear design validation, as well as the lead-based virtual reactor, CLEAR-V.

In August 2021, the construction of an electric heated pool-type LBE-cooled integration facility, called CLEAR-M0, was completed and operations at IANS were initiated. These operations consist of reactor and power conversion system tests with power levels of greater than 5 MWth to complete integration verification experiments related to thermal-hydraulic characteristics and performance of full-scale prototype components of CLEAR-M10. Meanwhile, loops are being built or planned for high-temperature testing of materials and comprehensive thermal-hydraulic performance (see Figure LFR-7). Independent facilities for verification of prototype devices, such as control rod drive mechanism and hierarchical linear modeling measurements, are also being built or planned.

References

GIF (2021), *GIF Lead-cooled Fast Reactor: Proliferation Resistance and Physical Protection White Paper*, GIF, Paris.

GIF (2021), *Safety Design Criteria for Generation IV Lead-Cooled Fast Reactor System*, GIF, Paris.

Andrei Moiseev
Chair of the LFR SSC,
with contributions from LFR members

Molten salt reactor

Participants in the GIF MoU for collaboration on molten salt reactor (MSR) research and development are the GIF members: Australia, Canada, Euratom, France, Russia, Switzerland, and the United States. In 2021, two provisional System Steering Committee (pSSC) meetings were held online (the 30th and 31st meetings), and included observers from China, Japan and Korea. Based on a decision made by the Policy Group, GIF MSR collaboration will continue as a pSSC, with the situation to be re-evaluated in two years.

The mission of the MSR pSSC is to support international collaboration on the development of nuclear energy concepts that can help to meet the world's future energy needs. Gen IV designs will use fuel more efficiently, reduce waste production, be economically competitive, and meet stringent standards of safety and proliferation resistance. To this end, the pSSC is creating a new system research plan for the MSR pSSC around three main axes: salt behavior, materials properties and system integration.

Main characteristics of the system

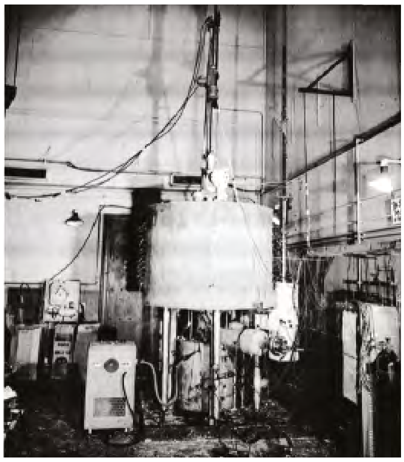
Liquid-fuel MSRs are a type of nuclear fission reactor in which a halide salt serves as the nuclear fuel and may also serve as the coolant. In solid-fuel molten salt reactors, the halide salt serves as the coolant for solid phase nuclear fuel. MSRs were originally conceived in the 1940s. The Oak Ridge National Laboratory (ORNL, United States) operated two MSRs and a number of supporting test facilities from the 1950s to the 1970s.

Both solid- and liquid-fueled MSRs have seen a resurgence in interest over the past two decades. Proposed designs with molten fluoride and chloride salt mixtures include both thermal and fast spectrum systems, as well as designs with time and spatially varying spectra. Nearly every form of fertile and fissile material is being considered for its potential use in an MSR fuel cycle.

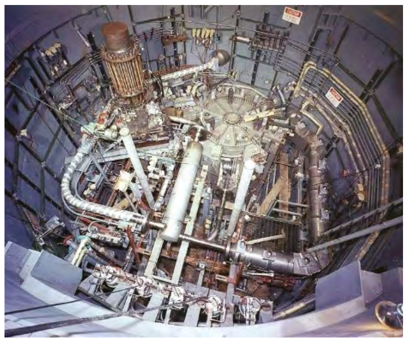
Figure MSR-1: Photos of MSR-related test facilities and reactor at ORNL

ORR in-pile fuel salt loop

ART 650°C critical assembly

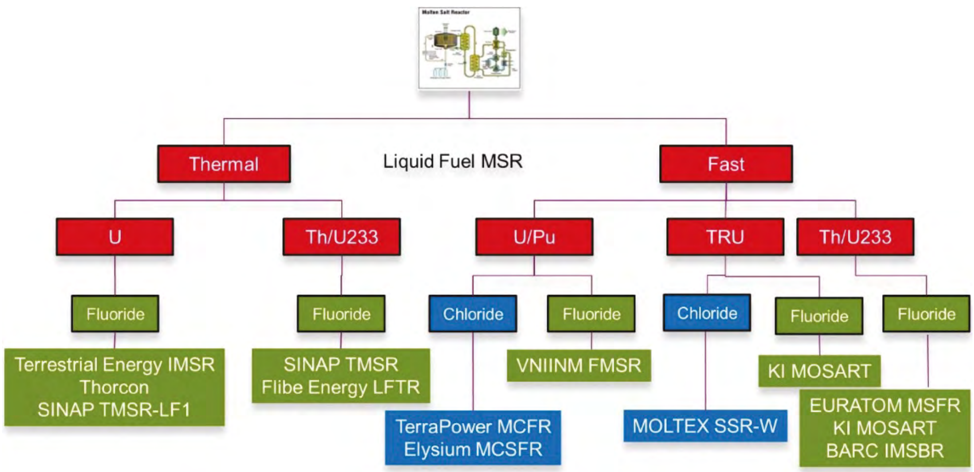


MSRE reactor

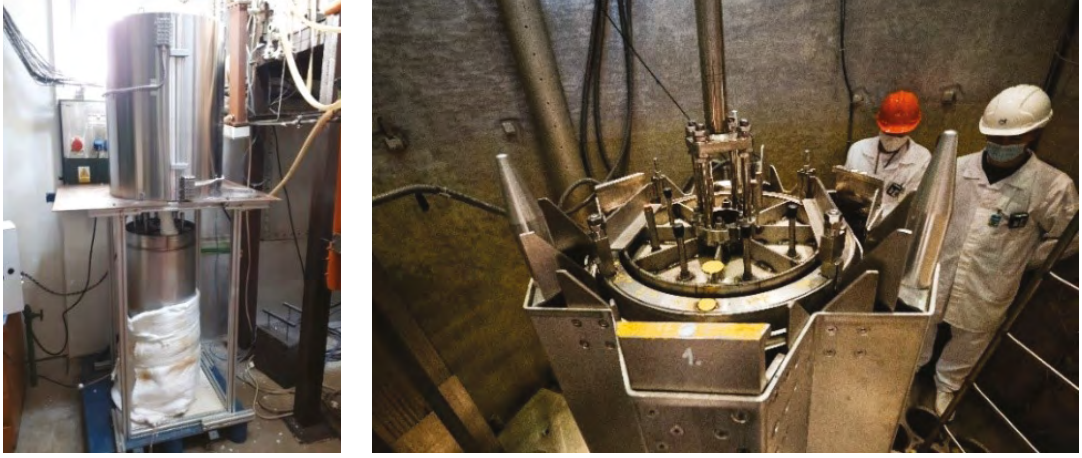


Source: ORNL.

Figure MSR-2: Recent MSR concepts, with key designers and concepts



Source: 39th GIF Experts Group Meeting.



Source: SALIENT-03 Project, JRC.

Figure MSR-3. Pouring the FLIBE melt into the central container of the zone and checking the complete zone in the LR-0 reactor vessel

MSRs have a number of advantageous characteristics ranging from high-temperature operation (and consequent increased thermodynamic efficiency) to low-pressure operation, reducing the driving force for radionuclide dispersal in the event of an accident. MSRs also tend to have strong negative reactivity feedback characteristics and effective passive decay heat rejection.

On the other hand, the extended distribution of radionuclides can necessitate fully remote maintenance. Molten salt can also become highly corrosive if exposed to oxidative impurities. Overall, MSRs have substantial technology differences from LWRs necessitating different approaches to safety assessment, safeguards, and operations.

R&D objectives

The common objective of MSR projects is to support collaboration on technology, data and analysis methods. While MSRs may be deployed in the near term (i.e. in the next few years) using adequate technology, their performance would be improved through the development of improved technologies and techniques, as well as through an increase in the amount of validated fuel salt thermophysical and thermochemical data. Potentially useful collaborative projects include:

- measurement of salt thermochemical and thermophysical properties;
- performance of integral and separate effect tests to validate safety performance;
- development of improved neutronic and thermal-hydraulic models and tools;
- study of materials issues associated with use at MSRs (e.g. erosion, corrosion, radiation damage, creep-fatigue);
- demonstration of tritium management technologies;
- salt redox control technologies to master corrosion of the primary fuel circuit and other components;
- demonstration of surveillance and maintenance technologies for high radiation areas, such as MSR containments;
- development of a safety approach dedicated to liquid-fueled reactors.

National MSR demonstration and development highlights

On 1 October 2019, the Euratom-funded 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 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 by experiments. The overall objective is to ensure that the MSR can comply with all expected regulations in 30 years' time. SAMOSAFER is coordinated by TU Delft and will run until 2023.

In 2021, SAMOSAFER development work focused, *inter alia*, on several topics. A specific MSR oriented defense-in-depth approach has been setup by analyzing the safety functions of all fuel salt locations in the reactor and by defining the number of containment barriers. A list of MSR-specific postulated initiating events has been established and a simulation tool for the reactor has been developed.

In the Czech Republic, preparations continued for the start of so-called "hot" experiments with the hot inserted FLIBE[1] zone in the LR-0 experimental reactor of the Řež Research Centre. At the end of the year, the filling of this zone with Li-7 FLIBE melt thus began, along with some minor design modifications to the zone which had proven necessary. The actual "hot" experiments – i.e. measurements of the FLIBE neutronic characteristics at temperatures ranges from 550 to 750°C, originally planned for 2021, should be carried out in 2022.

In France, the R&D program initiated by the CEA continued and was further developed with important contributions from the Centre National de la recherche scientifique (CNRS). Three options are being considered, all in a fast spectrum configuration using molten chloride: an isogenerator, a plutonium (Pu) burner, and a minor-actinide transmuter.

1. FLIBE is a molten salt (i.e. a mixture of lithium fluoride and beryllium fluoride).

This program, which is aimed at sketching an MSR, is multidisciplinary. It covers:

- the reactor system (i.e. neutronics, materials, components);
- the associated fuel cycle (i.e. salt behavior, corrosion, salt polishing);
- neutronics calculation with a key contribution from the CNRS in Grenoble;
- multi-physics and chemistry modeling and simulation (including MOSARELA for the definition of the reactor operation conditions and the salt treatment strategy).

In Russia, Rosatom continued preliminary MSR design development for:

- a 10 MWt test and large-power 2.4 gigawatt thermal (GWt) units with homogeneous core;
- its fuel salt clean-up unit at the Mining and Chemical Combine site (Zheleznogorsk) in order to demonstrate 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 Li,Be,An/F and Li,Na,K,An/F MSRs with circulating fuel for the transmutation of long-lived actinides from used LWR fuel;
- development of a preliminary design for the Li,Be,An/F MSR test and required materials to obtain a license for its placement.

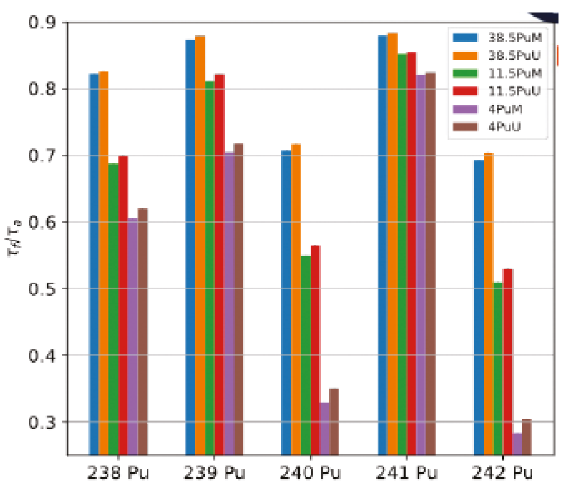
The main R&D efforts in 2021 were focused on the following topics:

- development of multi-physics modeling tools;
- identification and ranking for main safety function phenomenon;
- purification of the fuel/coolant salt;
- salt property measurement (liquid fuel and coolant);
- development of analytical methods for monitoring the salt composition;
- corrosion studies in natural convection loops;
- development of forced flow salt loop;
- construction of corrosion facility for fuel salt processing materials;
- fuel salt and container materials irradiation and post irradiation examination at the research reactor.

In the United States, a number of both salt-cooled and salt-fueled molten salt reactor supportive activities were performed in 2021. MSR demonstration projects supported by the DOE Office of Nuclear Energy (NE) continue to make progress. Both the construction of an integral effects test facility and the development of the molten chloride reactor experiment (MCRE) are underway, led by Southern Company Services in partnership with TerraPower. Operation of the integral effects test facility is anticipated in 2022, and first criticality of the MCRE is planned for 2026. Kairos Power's construction permit application for a low-power demonstration reactor was accepted for review by the US Nuclear Regulatory Commission (NRC).

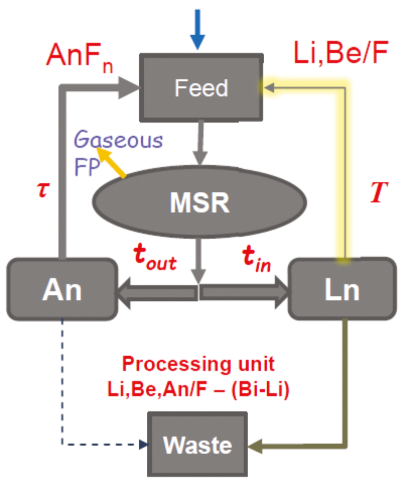
The Abilene Christian University-led effort to construct a university research reactor continues to make progress.[2] Also, a DOE-sponsored integrated research project to design, construct and operate a near core molten salt test loop at the MIT Research Reactor (MITRR) continues to make anticipated

Figure MSR-4: Neutronic studies of an MSR converter of Pu with NaCl-MgCl2 salt



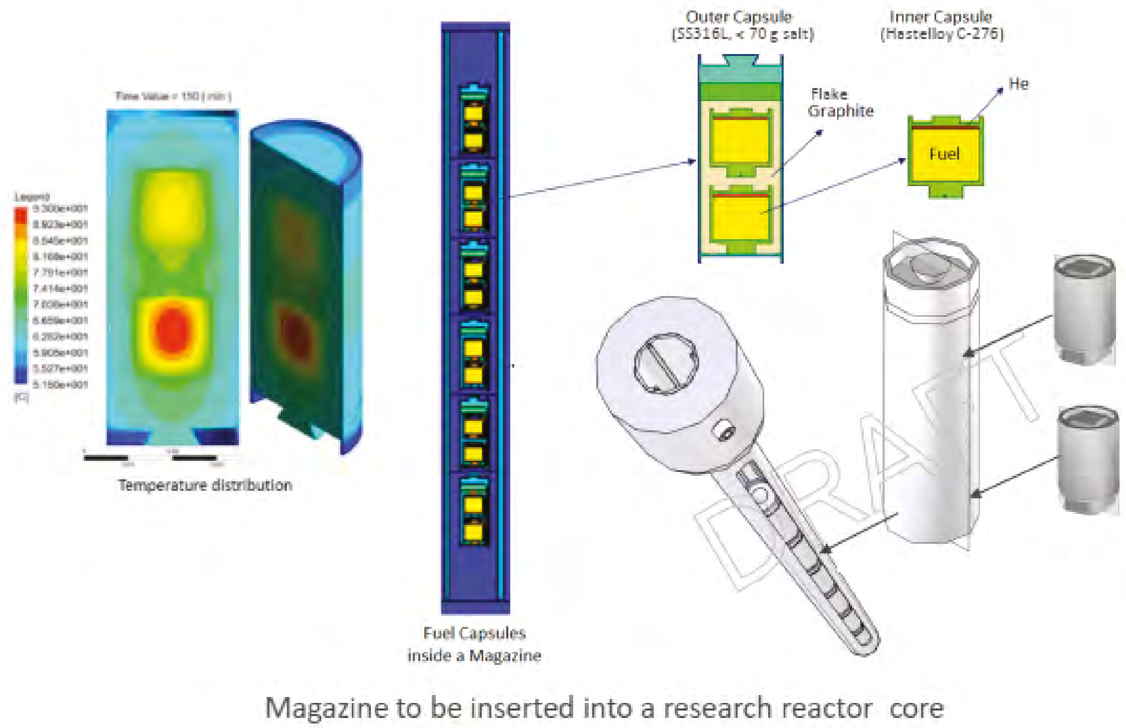
Source: Czech national program on MSR technology development.

Figure MSR-5: Li,Be,An/F MSR plant flowsheet for the transmutation of long-lived actinides from used LWR fuel



Source: 31st MSR pSSC meeting, CEA.

2. See: www.ans.org/news/article-3504/universities-to-host-a-new-generation-of-advanced-reactors.



Source: Laboratory of Subatomic Physics & Cosmology, CNRS.

Figure MSR-6: Design of a fuel salt irradiation rig for low power irradiation in a research reactor

progress.[3] The DOE's Advanced Research Projects Agency – Energy is also sponsoring the generation of molten salt irradiation data at the MITRR.[4]

The NRC continues its activities to develop technology-inclusive, performance-based, risk-informed licensing practices for advanced reactors. The NRC also continues to sponsor work to develop an efficient MSR fuel salt qualification methodology (Holcomb et al., 2021) and to incorporate MSR characteristics into its accident progression evaluation tools. A DOE-NE sponsored application of the phenomenon identification and ranking table technique to MSRs in 2021 (Holcomb et al., 2021). The DOE-NE also continues to sponsor development of procedures and acceptance criteria for in-situ structural materials surveillance for MSRs (Messner et al., 2021).

The ORNL annual MSR workshop, sponsored by the DOE-Gateway for Accelerated Innovation in Nuclear program, was held virtually in October 2021. A virtual meeting of the molten salt chemistry thermal properties working group was also hosted by the University of South Carolina in November 2021.[5]

In Canada, the Canadian Nuclear Laboratories (CNL) continued to develop expertise and capabilities in support of molten salt SMR concepts. Under the auspices of the Canadian Federal Nuclear Science and Technology Work Plan, the CNL continued to develop molten salt capabilities across a wide range of areas, including molten salt fuel behavior in accident conditions, salt chemistry and thermodynamic properties, multi-physics core behavior, thermal-hydraulics modeling and system behavior and decay heat removal. In 2021 the work focused on:

- development of procedures for synthesis of actinide salts using non-gaseous reagents;
- development of experimental protocols and QA practices for the determination of thermodynamic properties of molten salts and impact of impurities on melting points of selected salt mixtures;
- development of atomistic simulations to predict molten salt transport and thermodynamic properties;
- design of a fuel salt irradiation rig for low-power irradiation in a research reactor with the intent to follow the irradiation experiment with fission product release tests in the CNL's hot cells;
- construction of a corrosion loop to measure the corrosion of structural materials;
- construction of a molten salt natural circulation heat transfer loop to support fluid flow and heat transfer studies of chloride and fluoride salt mixtures.

3. See: https://neup.inl.gov/FY%202020%20CINR%20Abstracts/IRP-20-22026.

4. https://arpa-e.energy.gov/technologies/projects/generation-critical-irradiation-data-enable-digital-twinning-molten-salt.

5. See: https://sc.edu/study/colleges_schools/engineering_and_computing/research/research_centers_and_institutes/general_atomics_center/molten_salt_working_group/index.php.

In Switzerland, MSR research continued in 2021 at PSI with the major aim of monitoring technology, the education of new experts, and the development of knowledge and simulation capabilities in: fuel cycle, system behavior and thermodynamics areas of molten salts research. MSR fuel cycle simulation techniques (see Figure MSR-7) were verified and extended.

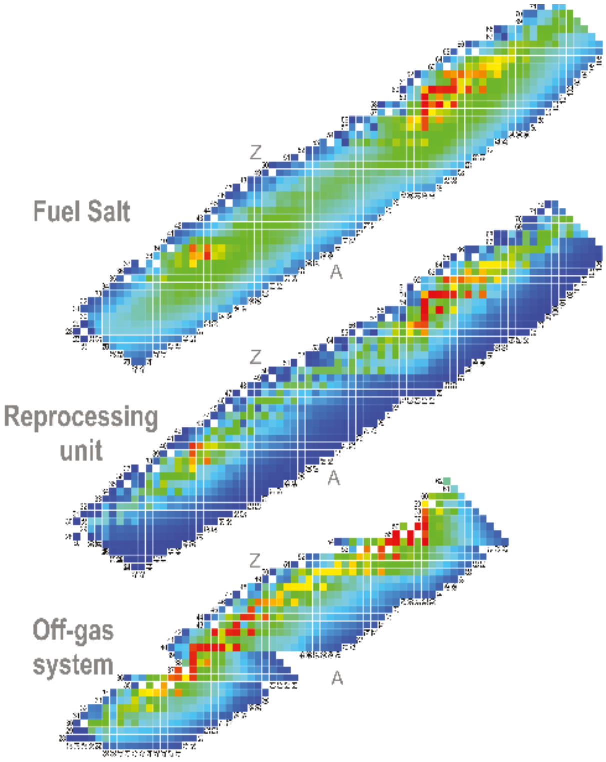
The system behavior study with Open-FOAM based solver ATARI was finalized in 2021 (De Oliveira, 2021). It focused on the assessment of the freezing phenomena in printed circuit heat exchangers and the conceptual design of an MCFR core with tube-in-tube and baffles options.

In China, the Shanghai Institute of Applied Physics, CAS China Academy of Sciences is steadily promoting the related work of their liquid fuel thorium-based molten salt experimental reactor (TMSR-LF1). As a result, the installation of experimental reactor equipment continues and is entering the final stage.

Research work based on the follow-up development of the MSR is also being carried out, notably in relation to:

- high-temperature alloy – the alloy structural material GH3539 for the MSR at 850°C has been preliminarily developed, and a comprehensive performance evaluation is being carried out;
- materials high-temperature evaluation method – a preliminary creep-plastic constitutive equation of the alloy has been established;
- electrochemical separation of MSR fuel – the influence of fluoride ions on the electrochemical behavior of Th(IV) in LiCl-KCl molten salt was confirmed;

Figure MSR-7: Fission product distribution in a molten salt fast reactor system at three locations: fuel salt (top), reprocessing unit (middle) and off-gas system



Source: Rosatom Project, Kurchatov Institute.

Note: The illustrative values correspond to cumulative values after five years of irradiation and are not in scale, because the fission products mass cumulates outside of the fuel salt.

Figure MSR-8: FLiNaK test loop

Source: Sukegawa Electric.

Figure MSR-9: Freeze-valve experiment

Source: Aji.

- molten salt structure – the structures of thorium fluorides in molten FLIBE and FLiNaK[6] were determined.
- electrolytic processing of oxide spent fuel – electrolytic reduction of lanthanide oxides in LiCl-KCl molten salt was achieved.

In Japan, Thorium Tech Solution Inc. plans to perform tests at a FLiNaK loop (15L/min) with the Sukegawa Electric Co., Ltd (see Figure MSR-8). Also, a molten salt pump for FUJI is under design. The FLiNaK (50L/min) loop at a fusion blanket system in the National Institute of Fusion Science (NIFS) is now proceeding with freeze valve tests for the MSR, through collaboration with MOSTECH/MSLab, Kyushu University, UEC, and NIFS (MSR-9).

References

De Oliveira, R. G. (2021), "Improved methodology for analysis and design of Molten Salt Reactors", Doctoral thesis, EPFL Lausanne, Switzerland.

Holcomb, D. E., W. P. Poore, III and G. F. Flanagan (2021), *MSR Fuel Salt Qualification Methodology*, Draft NUREG/CR for comment, ML21245A493.

Holcomb, David E., Alexander J. Huning, Michael D. Muhlheim, Richard S. Denning and George F. Flanagan (2021), *Molten Salt Reactor Fundamental Safety Function PIRT*, ORNL/TM-2021/2176, DOI: 10.2172/1824962.

Messner M. C. and T. -L. Sham (2021), *Preliminary Procedures and Acceptance Criteria of in-situ Structural Materials Surveillance for MSR*, ANL-ART-229.

Stéphane Bourg
Chair of the MSR pSSC,
with contributions from MSR members

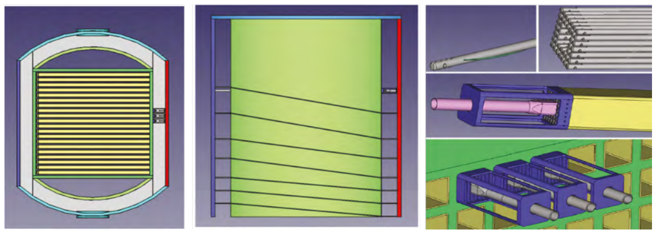
6. FLiNak is a salt mixture of lithium fluoride, sodium fluoride and potassium fluoride.

Super-critical water reactor

Signatories of the System Arrangement for collaboration on supercritical-water-cooled reactor (SCWR) research and development are the GIF members: Canada, China, Euratom, Japan and Russia. Three technical projects have been established for GIF collaborations:

- the provisional SCWR system integration and assessment, with all signatories;
- SCWR materials and chemistry, with Canada, China and Euratom as members;
- SCWR thermal hydraulics and safety, with Canada, China and Euratom as members.

Figure SCWR-1: Schematic diagram of the EU SMR design



Source: T. Schulenberg, I. Otic, Suggestion for design of a small modular SCWR, ISSCWR-10, Prague, Czech Republic, 15-18 March 2021.

Main characteristics of the system

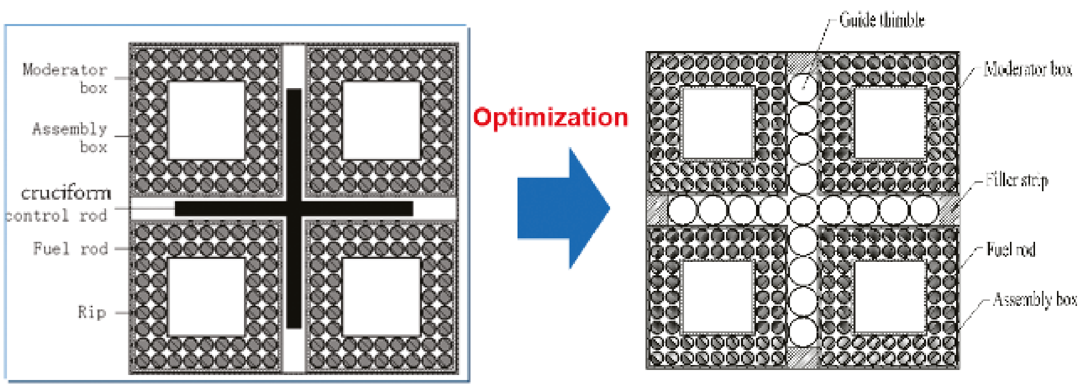
The SCWR is a high temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point (374°C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: pressure-vessel concepts proposed first by Japan and more recently by a Euratom partnership and China, and a pressure-tube concept proposed by Canada. Other than the specifics of the core design, these concepts have many similar features (e.g. outlet pressures and temperatures, thermal neutron spectra, steam cycle options, materials). The R&D needs for each reactor type are therefore common, which enables collaborative research to be pursued.

The main advantage of the SCWR is improved economics because of high thermodynamic 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.

Key critical-path topics have been identified in the SCWR System Research Plan, which are grouped into the three technical projects:

- System integration and assessment – definition of a reference design, based on the pressure tube and pressure vessel concepts, which would meet the Gen-IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable 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.
- Materials and chemistry – qualification of key materials for use in in-core and out-core components of both the 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 an understanding of water radiolysis.

Figure SCWR-2: FA optimization design of SCWR 1000



Source: Presented at the TH&S SCWR PMB meeting.

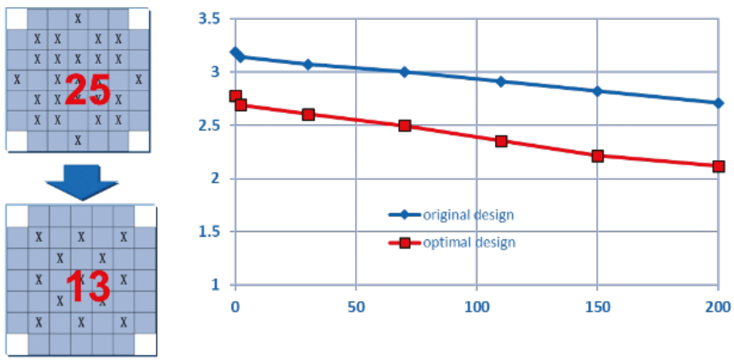
Technical highlights – system integration and assessment project

EURATOM is developing a SCW-type SMR through the Joint European Canadian Chinese project (ECC-SMART). Electric power output of the SMR should be around 200 to 300 MW. The specific plant erection costs (EUR/kW installed electric power) should be 20% less compared to those for SMR concepts that are based on the pressurized water reactor (PWR). The power plant shall remove residual heat without the need of electric power at least within a period of three days. The specific fuel cost (EUR/MWh electric power) will be smaller than those of SMR concepts based on a PWR, which may be accomplished by a higher efficiency compensating higher fuel production costs. A preliminary concept has been proposed for further research.

China has proposed two kinds of large scale SCWR concepts, and one SCW-SMR is ongoing, namely the CSR1000, SCWR-M, and the CSR-150 respectively. The CSR1000 design has been improved. A new kind of fuel assembly (FA) with guide thimble in a cruciform array is being proposed to slow the pressure drop through hydraulic buffer. The square FA consists of four sub-FA welded together by four filler strips. 17 guide thimbles arranged close together in the cross-shaped passage, surrounded by the walls of four sub-assembly boxes. The design of optimized FA and cruciform control rods were feasible for SCWR in terms of the dropping behavior. For the optimal design of the CSR150, 25 control rods are used in the CSR150 core to obtain adequate reactivity control ability. The number of control rods is large and not conducive to uniform power distribution. In this optimal design, Er_2O_3 and control rods are used simultaneously for reactivity control.

CNL is now developing a tool to support decision analysis by studying the interplay of key variables

Figure SCWR-3: Optimization design of CSR150

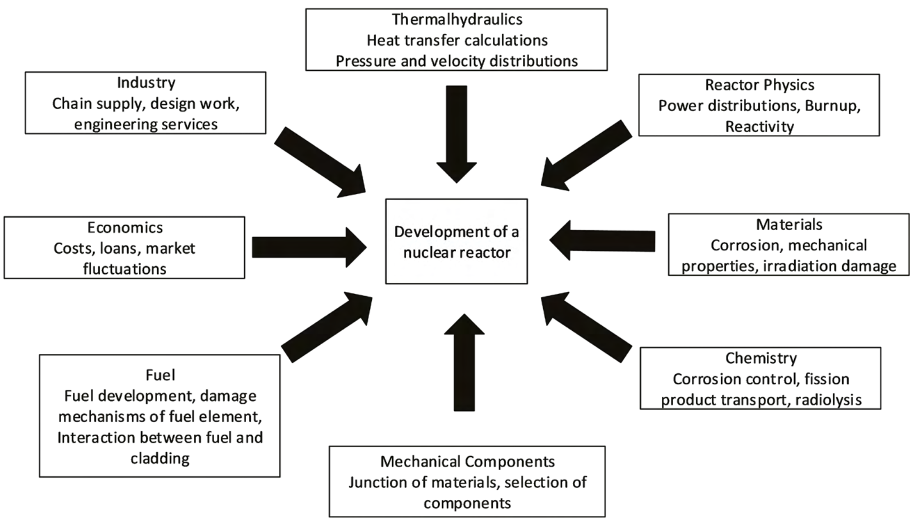


Source: Presented at the TH&S SCWR PMB meeting.

that are used for the conceptualization of the SCWR and SCW-SMR, based on the experience and know-how developed during the conceptualization of the Canadian SCWR. The tool currently consists of four modules, namely: 1) thermodynamics and energy transport; 2) thermal hydraulics; 3) safety analysis; and 4) economics. The following subsections give a short description of the purpose of each module, as shown in Figure SCWR-4.

Russia is considering two variants of SCWR concepts: with the single-circuit direct power conversion system (i.e. the Russian light water power pressurized reactor model ,VVER-SKD) and with the two-circuit indirect power conversion system (VVER-SCP). The VVER-SCP-600 is believed to demonstrate the following main advantages: 1) the design conditions of primary coolant provide coolant hydraulic stability and reliable heat transfer at the surface of the core fuel rods; 2) the two-circuit indirect scheme ensures

Figure SCWR-4: Nuclear reactor design interrelation



Source: Canadian Nuclear Laboratories project on SCWRs.

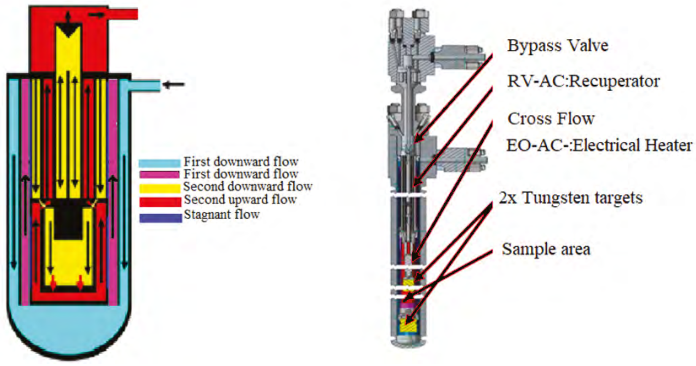
the radiation safety of the turbine plant at the VVER-1200 level; 3) the fast-resonant neutron spectrum in the reactor core provides good neutron-thermal hydraulics stability; and 4) the reactor installation has a compact layout for the primary equipment, and the high quality of the secondary SCW steam at the turbine inlet promotes compactness in terms of the turbine installation, as well as an effective combination of high-pressure and mid-pressure cylinders, and a decreasing number of low-pressure cylinders.

Technical highlights – thermal hydraulics and safety project

Euratom's main activity focused on the continuation of work initiated in previous years. The supercritical water loop, presented in Figure SCWR-5, remains the cardinal point in the materials testing and safety assessments being carried out in the Centrum výzkumu Řež (CVR) in the Czech Republic. During 2021, new activities were performed in order to improve the knowledge of SCW using ATHLET 3.1A for the simulation of accident and abnormal transient scenarios. Several pressure tests and experiments were performed in the out-of-pile configuration. A benchmark activity was later performed using the obtained data.

Using an algebraic heat flux model developed in the STAR-CCM+ code, the University of Pisa has in recent years been analyzing via Reynolds-averaged Navier-Stokes (RANS) model CO_2 data, water data and other data produced by several researchers. The RANS model was assessed and improved through 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, the subject of a fluid-to-fluid similarity theory for heat transfer at

Figure SCWR-5: Active channel thermocouples map

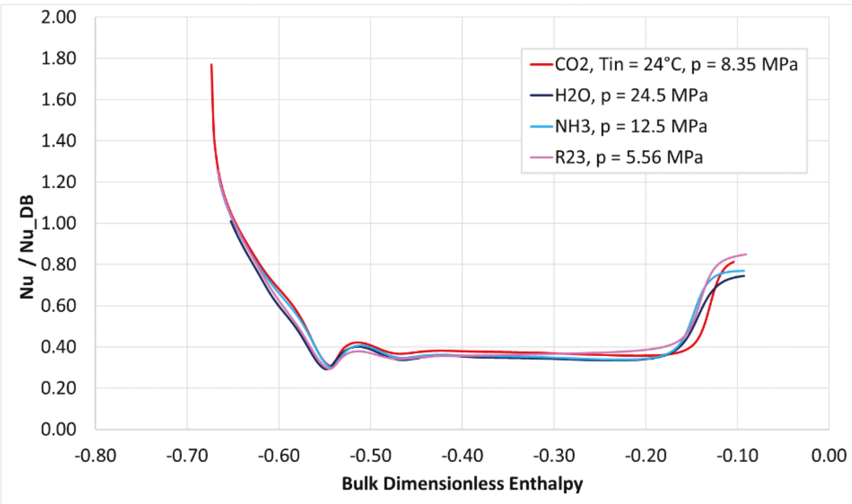


Source: SCWR SA Original work.

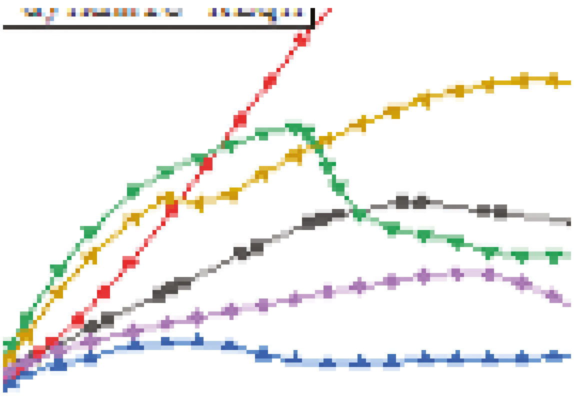
supercritical pressure was further developed. Figure SCWR-6 shows one of the cases addressed from Kline's database (i.e. the trend of the ratio between the computed Nusselt number for four different fluids) and one predicted by the classical Dittus-Boelter correlation.

The Karlsruhe Institute of Technology (KIT) in Germany focuses on experimental heat transfer studies, system design and safety requirements and numerical simulations of turbulent heat and mass transfer under SCWR conditions. At the KIT Model Fluid Facility, experimental investigation of the influence of corrosion on the heat transfer to supercritical fluid has now been designed. This work is being performed in cooperation with the CVR in the Czech Republic. The CFD modeling approach has been developed and applied. The numerical work currently covers RANS and large eddy simulation (LES) approaches.

Figure SCWR-6: Ratio of the computed Nusselt number to the value from the Dittus-Boelter correlation for the data by Kline with q"=20 kW/m² and T_{in} = 24°C

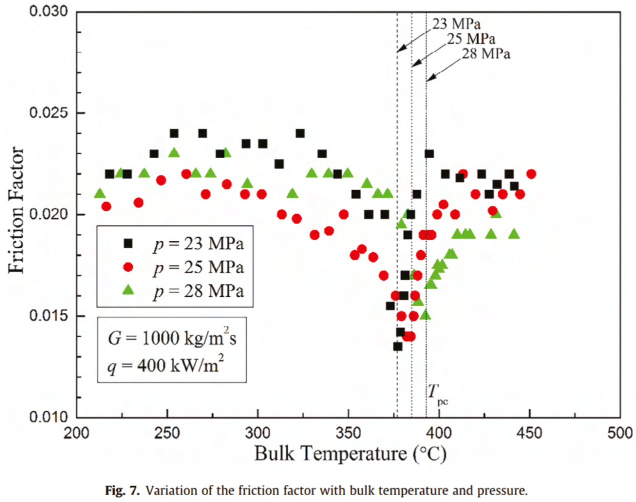


Source: Sara Kassem, Andrea Pucciarelli, Walter Ambrosini, Insight into a fluid-to-fluid similarity theory for heat transfer at supercritical pressure: Results and perspectives, International Journal of Heat and Mass Transfer 168 (2021) 120813.



Source: Presented at the TH&S SCWR PMB meeting.

Figure SCWR-7: Wall temperature distribution comparison



Source: Presented at the TH&S SCWR PMB meeting.

Figure SCWR-8: Variation of the friction factor with bulk temperature and pressure

At the University of Sheffield (USFD) in the United Kingdom, research primarily focuses on using DNS, to produce detailed data and provide insights into fundamental physics. The USFD has been working on extending the novel sub-channel CFD, a coarse mesh method that combines CFD and sub-channel approaches, to SCWR applications. Sub-channel CFD and convectional CFD, both based on the CFD solver Code_Saturne, have been used in the benchmarking exercise organized by the Nuclear Power Institute of China (NPIC).

In Hungary, the Centre for Energy Research and the Budapest University of Technology and Economics have been dealing with the thermal-hydraulics and safety-related research of SCWRs, including experimental (i.e. not only heat transfer) activities, theoretical research of water chemistry and thermal-hydraulic issues related to SCW. The KTH Royal Institute of Technology in Sweden is working on the Supercritical Water Applications in Nuclear Systems project, using the high-pressure water test loop. In the project, detailed measurements are performed, where both the heat transfer at the wall and the internal flow structure are measured using a new experimental approach.

In China, two technical seminars for SCW thermal hydraulics and a safety benchmark were held in May and December 2021, respectively. The benchmark activities, now part of the ECC-SMART project, were launched by the NPIC and Xi'an Jiaotong University (XJTU) last year to address SCW thermal-hydraulics and safety-related issues, including the 2×2 bare rod channel, the local warped wired rod and the full-length warped wired rods, as well as parallel channel instability. The next stage of the seminars will continue in 2022.

XJTU developed the SCTRAN V2.0 by introducing the two-phase flow model. A wide range of heat transfer models, from subcritical to supercritical, are being realized in this tool. The capabilities of the developed SCTRAN have been verified by calculating the rapid blowdown phenomenon and heat transfer during the trans-critical process. XJTU is also working on developing CFD codes to propose a preliminarily verification of the prediction of heat transfer deterioration by the LES model, as well as the DNS model. The three LES sub-lattice models effectively predicted the heat transfer deterioration and the turbulent recovery mechanism, but failed to match-the higher fidelity DNS data (see Figure SCWR-7). The pressure drop and friction of SCW in a 2×2 rod bundle were investigated experimentally. The variations of total pressure drop, frictional pressure drop, gravitational pressure drop and acceleration pressure drop with bulk enthalpy were also studied. It was found that the pressure drop varies notably in the pseudo-critical and high-enthalpy regions due to the strong variations of the thermo-physical properties. The friction factor exhibits a local valley as the bulk temperature approaches the pseudo-critical temperature (Figure SCWR-8).

The group from the Institute of Engineering Thermophysics, Chinese Academy of Sciences (IET-CAS) is studying the 4-m long vertical heat transfer transition phenomena and mechanism analysis of SCW cases. Heat transfer is simulated under 9 groups of grids in a 4-meter tube, and the change of the outlet temperature is compared with the results of the loops experiment data in literature, showing good agreement. In addition, the IET-CAS group also used the newly proposed BP neural network. Using a homemade code, a feedforward process is designed and trained based on error back propagation. The 2 834 groups of experimental data

(IET-CAS and the University of Ontario Institute of Technology, Canada) were compared to the neural network model, which performs well in the simulated $T_c<T_{bulk}<T_{pc}$ region with partial point deviation in the $T_{bulk}<T_c$ region. The BP network model also shows a good performance in simulating the supercritical heat transfer, with an uncertainty of ±10% (see Figure SCWR-9).

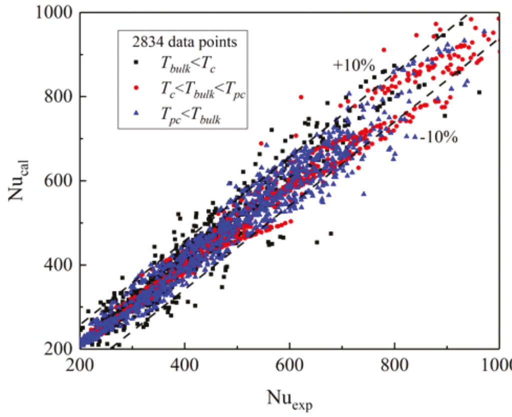
Using the CNL developing tool, an analysis of thermodynamics and energy transport, thermal hydraulics, safety and economics was carried out with the ultimate goal of developing tools to allow the optimization of a reactor concept. Work related to thermodynamics and energy transport is intended to understand the effects across the possible temperature and pressure-operating ranges. The thermal-hydraulics module uses the hottest channel as a bounding case to predict the maximum cladding temperature and enthalpy distribution across the proposed fuel bundle. Figure SCWR-10 represents an analysis of the CANFLEX vis-à-vis the 64-element bundle under different thermodynamic conditions. The safety analysis module that assesses the dynamic response of the reactor was constructed. The economics module provided a cost analysis of proposed cycles (shown in Figure SCWR-11). The development of this tool is ongoing, and the current modules will be reviewed and verified.

Technical highlights – materials and chemistry project

EU activities related to the SCWR design are focused on two European funded projects: mitigating environmentally-assisted cracking through optimization of surface condition (MEACTOS) project and ECC-SMART. Although the SCW environment is used in both projects, the role of SCW is different: MEACTOS uses SCW as an accelerating corrosion environment whereas ECC-SMART is focused on the development of technology for a future SMR based on SCW coolant.

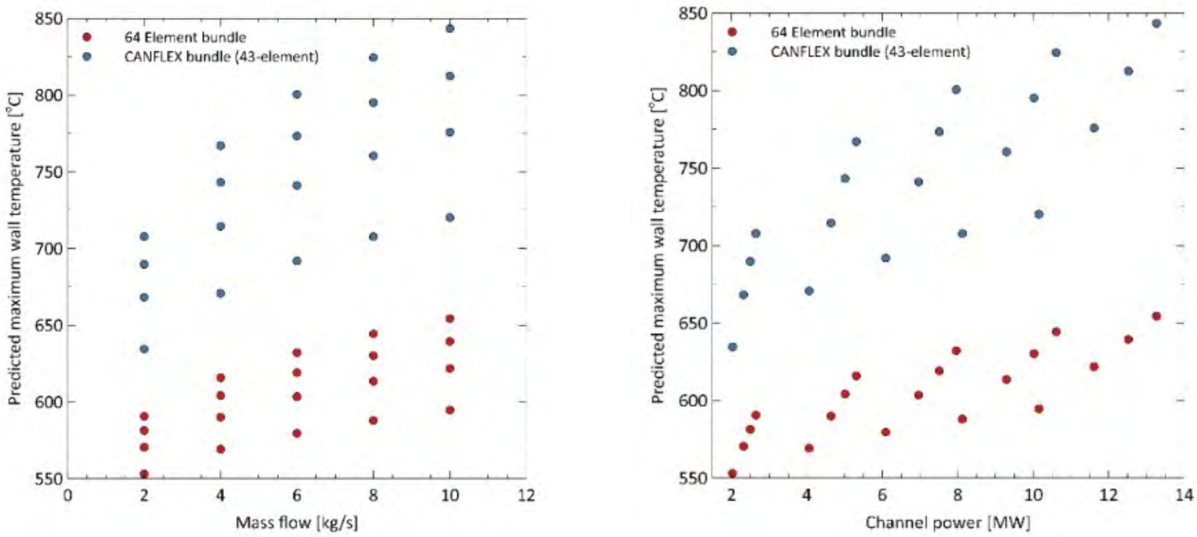
The main objective of MEACTOS is to study the effect of new machining technologies, such as supercritical CO_2, and the minimum quantity of lubricant on the corrosion resistance of LWR structural materials (e.g. stainless steels and Ni based alloys). The

Figure SCWR-9: Contrast between BP neural network result with experimental Nu (pin = 24 MPa, Tin = 320-350°C, G = 1 000-1 500 kg/m²s, q = 391-1 256 kW/m²)



Source: Presented at the TH&S SCWR PMB meeting.

Figure SCWR-10: Maximum wall temperature predictions for the 64-element and CANFLEX bundles



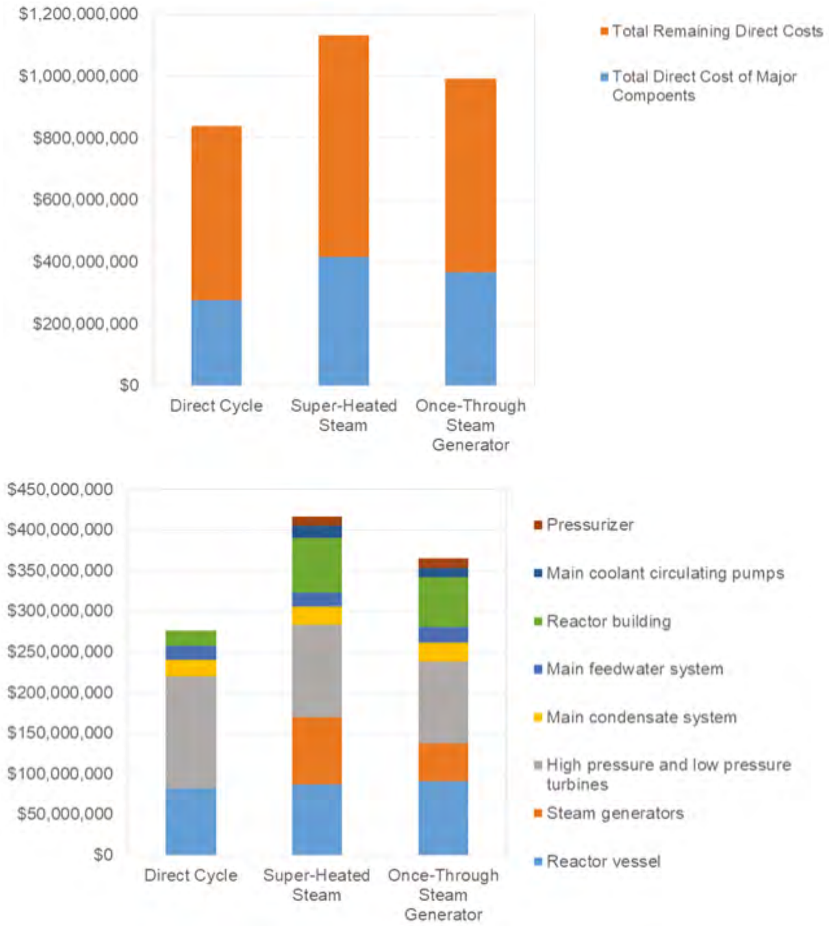
Note: The inlet coolant temperature ranges from 350°C to 380°C with an increase of 10°C. The outlet coolant temperature is fixed at 400°C.

Source: Canadian Nuclear Laboratories project on SCWRs.

methodology used to carry out the corrosion tests is based on the methodology developed in previous EU funded projects (e.g. MICRIN +). In this case, tensile specimens with tapered gauge section are tested by dynamic tests and constant load tests. Results obtained from the constant load tests performed in SCW did not show a clear acceleration of the corrosion processes in comparison with specimens tested in liquid water at lower temperatures. It is supposed that the weak electrochemistry within the supercritical region prevails over the temperature, at least in pure SCW. Throughout 2021, materials (stainless steel S 310 S, A 800H and alumina forming alloy based on 310 S) have been purchased or manufactured and characterized (Figure 12). The oxidation and tensile specimens were then machined. The first oxidation and electrochemical tests began in mid to late 2021, and the first results are expected around February/March 2022.

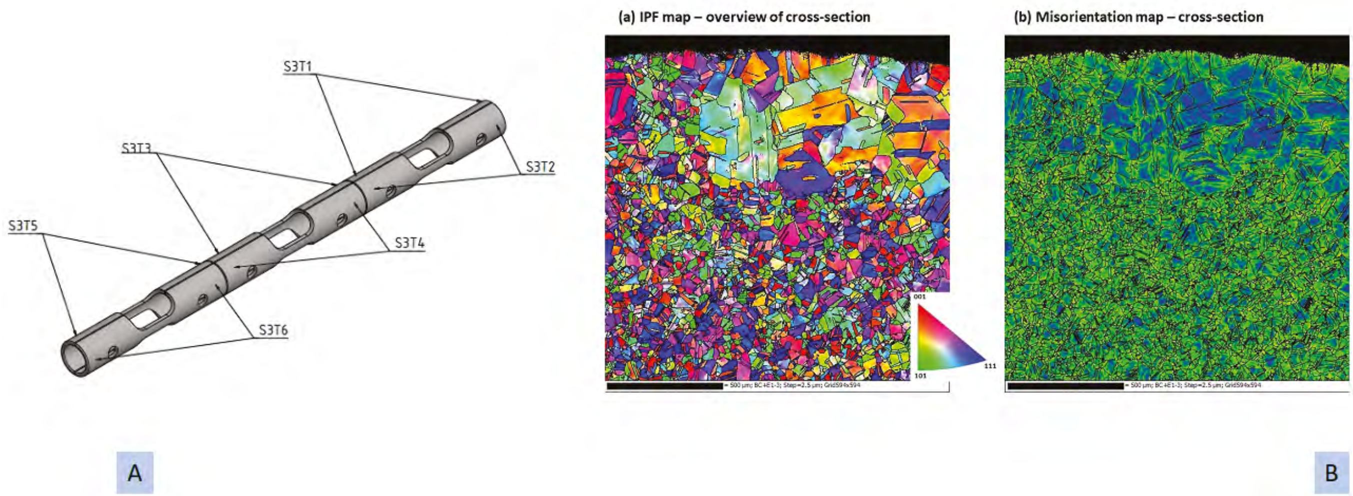
In China, materials and chemistry activities focused on the design and preparation of novel candidate materials for the SCWR fuel cladding tube. Alumina-forming austenitic (AFA) and oxide dispersion-strengthened (ODS) steels have been recognized as potential candidates for SCWR applications. Based on the work of previous years, AFA steels with 3.5Al-25Ni-15Cr and 3Al-25Ni-20Cr

Figure SCWR-11: Economic analysis of major components



Source: Canadian Nuclear Laboratories project on SCWRs.

Figure SCWR-12: A) Tensile specimens machined from a tube and tested in ECC-SMART; B) Inverse pole figure map (left) and misorientation map (right) obtained by electron back scattered diffraction from the cross section of a stainless steel 310 S tube



Source: SCWR SA original work.

were prepared and named as A2515MNC and A2520, respectively. Table SCWR 1 shows the chemical compositions of the new AFA alloys. High-temperature creep performance of forged A2515MNC was evaluated at a temperature of 700°C with constant load from 100 up to 180 MPa. AFA alloy shows good creep performance, with a creep rate at 100 MPa, or about 6.4×10-6 of the as-forged sample. AFA alloys with 2.5Al-26Ni-19Cr and different concentrations of Si were prepared and exposed in deaerated and oxygenated SCW at temperatures of 500°C, 550°C and 600°C, and a pressure of 25 MPa. A cracking mechanism is proposed for alloy 800H in an SCW environment, based on the experimental and analysis results. The schematic diagram is shown in Figure SCWR-13. The creep and corrosion of materials in a high-temperature SCW environment play important roles in the cracking process.

Table SCWR-1: Chemical compositions of AFA alloy prepared by USTB, wt.%

Sample	Ni	Cr	Al	Si	Mo	Nb	C	Fe
A 2515MNC	25	20	3	0.3	2	0.65	0.02	Bal.
A2520	25	15	3.5	0.3	2	0.65	0.02	Bal.

Canada continues to assess the viability of neutron-transparent alternatives to austenitic stainless steels and nickel-based alloys for fuel cladding in a scaled-down SCWR.

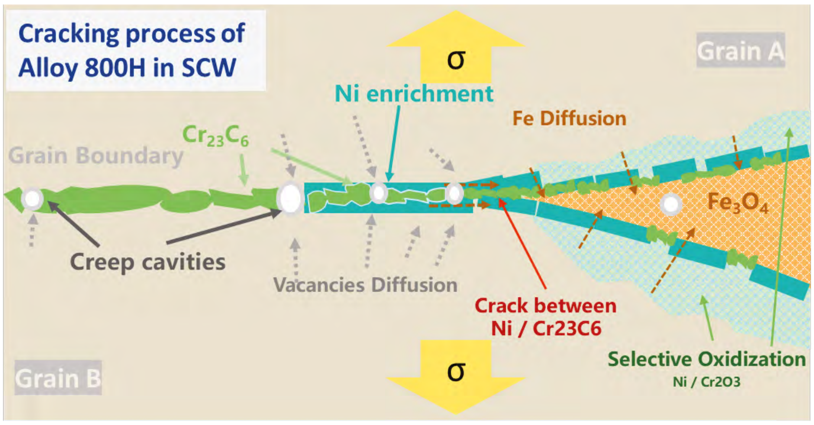
Corrosion studies were conducted on potential fuel cladding materials for small SCWR applications. Figure SCWR-14 shows the autoclave oxidation results from three groups of materials (Cr-coated Zr2.5Nb, Cr coated ZrCrFe and FeCrAl) tested at the operating conditions of scaled-down SCWRs. The operating conditions were at 500°C and 23.5 MPa, with 630 μg/kg oxygen in the feed water. The first two groups are zirconium alloys coated with approximately 10 μm of metallic chromium. These materials were machined with the largest surface area in longitudinal and transverse directions or with the main surface in parallel or perpendicular to the rolling direction or extrusion direction. At similar coating thicknesses, among r-2.5Nb, the pressure tube (solid green line) shows a better corrosion resistance of up to about 1 200 hours. The coated ternary alloy zirconium-chromium-iron (Zr-1.2Cr-0.1Fe) exhibited a 25-50% reduction in weight gain compared to the coated Zr-2.5 Nb. For the same materials, the longitudinal and transverse coupons show similar corrosion behavior with improved corrosion resistance compared to bare (i.e. no coat) coupons. The third material, which is another ternary alloy of Fe-Cr-Al (Fe-21Cr-5Al) shows a very good corrosion resistance with hardly any weight change up to 550 hours. Cross-section microscopy samples were prepared from corroded specimens to investigate the oxidation kinetics and to quantify adherence of Cr coating as well as oxide scales.

Table SCWR-2: Summary of irradiation damage and helium generation of candidate cladding materials for the SCWR-SMR, calculated using SPECTER

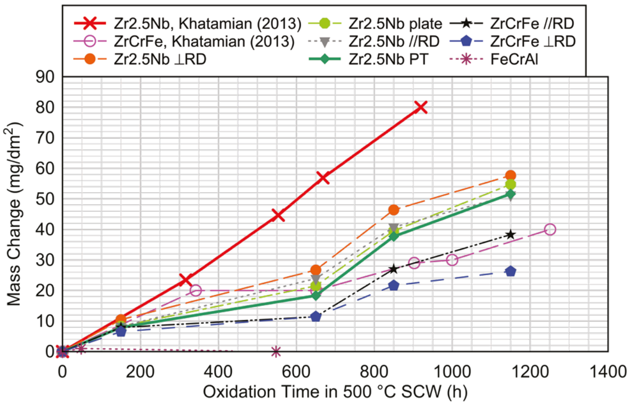
Fuel cladding material	Zr-2.5Nb + 10 μm Cr	Alloy 800HT	Type 310 stainless steel
5-year dpa	5.52	5.92	5.94
5-year He generation	0	62.2	39.7

Dpa and helium damage studies were performed on alloys 800H, 310 SS and weld joints of Ni based alloys. Evaluation of fuel cladding performance in an SCWR requires assessment of the effects of irradiation on physical properties and corrosion. Neutron damage

Figure SCWR-13: Cracking mechanisms of alloy 800H

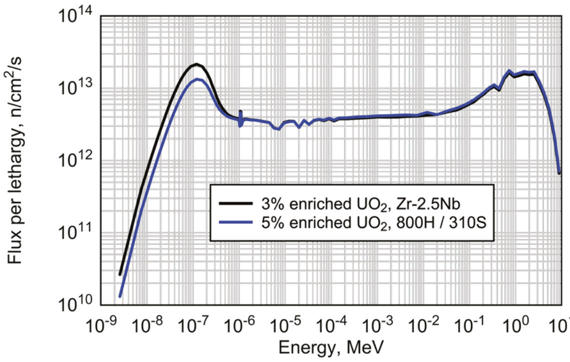


Source: SCWR SA original work.



Source: SCWR SA original work.

Figure SCWR-14: Weight gain of coated materials in 630 ppb oxygenated SCW at 500°C and 23.5 MPa



Source: Haozhan SU, et al. ISSCWR 10 paper "Effect of creep and dissolved oxygen on the cracking behavior of Alloy 800H in supercritical water environment", ISSCWR-10, Prague, Czech Republic, 15-18 March 2021.

Figure SCWR-15: Neutron spectrum generated using Serpent 2 for the inner fuel ring, using 3% enriched UO_2 fuel with Zr-2.5Nb cladding coated with 10 µm Cr and 5% enriched UO_2 fuel with alloy 800HT and type 310 stainless steel

and helium generation calculations were performed on candidate fuel cladding materials for SCWR SMR applications using the industry standard code SPECTER as an important first step in this assessment. The calculations employed neutron energy spectra that had been generated in early 2021 using the Monte Carlo reactor physics code Serpent 2; fresh 3% enriched UO_2 fuel was used to generate spectra for Zr-2.5Nb fuel cladding (Figure SCWR-15), while 5% enriched UO_2 fuel was used for Alloy 800HT and type 310 stainless steel fuel cladding. The total dpa and helium generated over a five-year in-service period for the three candidate fuel cladding materials are summarized in Table SCWR-2. The results show that the three materials would experience similar neutron damage (~ 6 dpa) in a five-year service period. Alloy 800 HT has a larger amount of nickel than 310 SS and therefore would produce a larger amount of helium, about 62 atomic parts per million (appm) He in a five-year service period compared to about 40 appm He for 310 SS.

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

Sodium-cooled fast reactor

The System Arrangement for Gen-IV international R&D collaboration on the sodium-cooled fast reactor (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 CEA, France; the DOE, United States; the JRC, Euratom; the Japan Atomic Energy Agency (JAEA), Japan; the Ministry of Science and Information and Communication and Technology (ICT), Korea; the China National Nuclear Corporation (CNNC), China; Rosatom, Russia; and the Department for Business, Energy and Industrial Strategy, the United Kingdom. Four technical projects have been established for GIF collaborations:

- SFR system integration and assessment, with all signatories participating;
- SFR safety and operations, with all signatories participating;
- SFR advanced fuels, with all signatories participating;
- SFR component design and balance-of-plant, with France, Japan, Korea, and United States as members.

Main characteristics of the system

The SFR system uses liquid sodium as the reactor coolant, allowing high-power density with low-coolant volume fraction. Because of the advantageous thermophysical properties of sodium (high boiling point, heat of vaporization, heat capacity and thermal conductivity), there is 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 are summarized in Table SFR-1. Plant sizes ranging from small modular systems to large monolithic reactors are being considered.

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: Chinese sodium fast reactor (CFR1200, China), the Japanese sodium-cooled fast reactor (JSFR, Japan), Korea advanced liquid metal reactor (KALIMER, Korea), the European sodium fast reactor (ESFR, Euratom), the BN-1200 (Russia) and the advanced fast reactor (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.

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

Note: GWD/MTHM = gigawatt days per metric ton of heavy metal.

Technical highlights – system integration and assessment project

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.

From 2010 to 2019, the CEA developed the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID).

The ambitious objectives of ASTRID were to fulfil Gen-IV requirements. It has led to the implementation of innovative technological solutions that go beyond current feedback. For ASTRID, a coherent conceptual design configuration (Figure SFR-1)

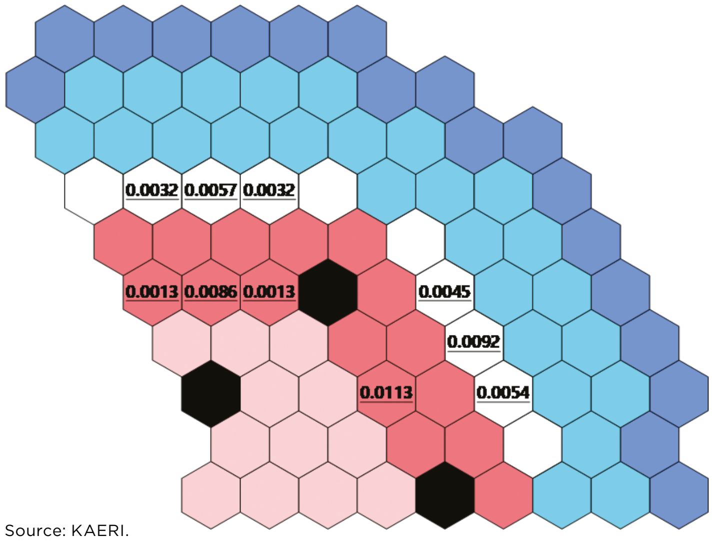
Figure SFR-1: ASTRID reactor view

Source: CEA.

has been proposed with innovative techniques and systems across all domains: core, fuel assembly technology, nuclear island, civil engineering, energy conversion system, plant layout, in-service inspection and repair (ISI&R), fabricability, and even project management. In 2021, the CEA provided an overview of the significant innovations that have been considered for ASTRID.

In 2021, Euratom worked on guidelines for the development and assessment of the ESFR innovative safety design options. In particular, the definition of ambitious guidelines for an advanced SFR is being discussed: enhanced safety and robustness, such as independence of the provisions of the different levels of defense in depth, prevention of core melting, technology breaks, and cost reduction. Furthermore, Euratom is also discussing safety inputs for the assessment of the innovative safety design options in response to specific demands: transients to be analyzed, performances requirements, etc.

The Korea Atomic Energy Research Institute (KAERI) has designed a long cycle core concept of almost 20 years by modifying the prototype Generation IV sodium-cooled fast reactor (PGSFR) ten-month cycle core. The core geometry modification was conducted for the concept design of a long cycle, accommodating the constraint of the minimal geometrical change to effectively utilize most of the structures, systems and components of the original PGSFR. Increasing assembly pitch and active core height, and decreasing lower/upper reflector height from the initial trial satisfied requirements of assembly weight and core shroud diameter. The cycle length of 18.5 years full power operation was achieved, and maximum cumulative damage fraction of cladding (see Figure SFR-2) meets the requirement of safety criteria, including manufacturing uncertainty.



Source: KAERI.

Figure SFR-2: CDF map after 18.5 years of full power operation

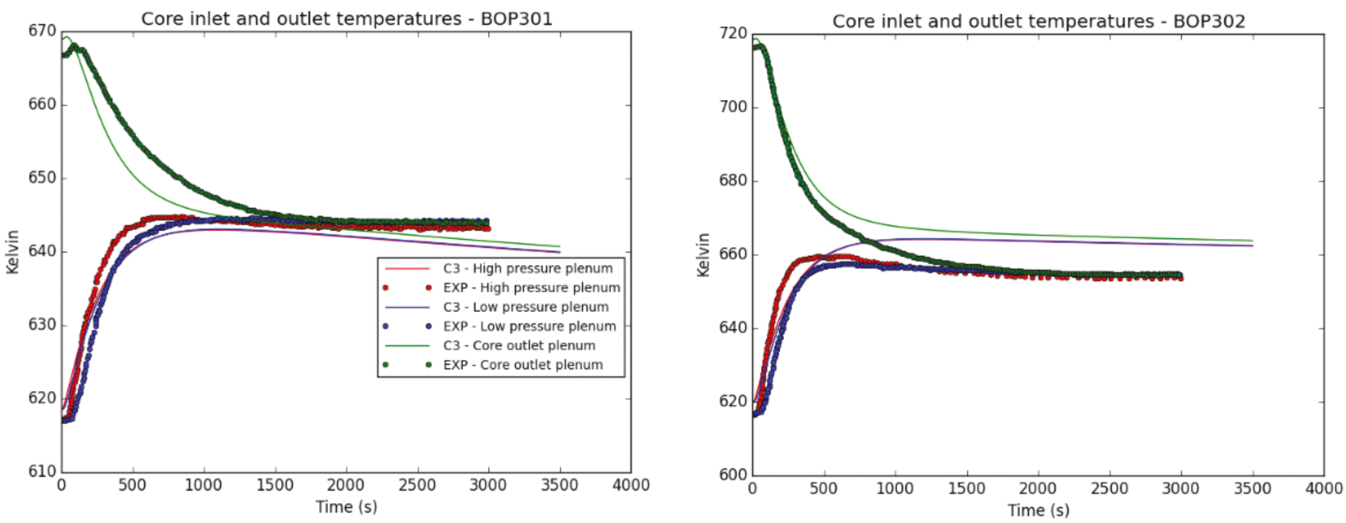
Technical highlights – safety and operations project

The safety and operations project is arranged into three work packages: 1) methods, models and codes for safety technology and evaluation; 2) experimental programs and operational experience; and 3) studies of innovative design and safety systems.

The project is also pursuing a "common" project that consists of two benchmark analyses (the EBR-II test and Phenix dissymmetric tests), which 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. In 2021, the CEA started the "open phase".

As part of this benchmark comparison, the CEA performed a preliminary analysis of EBR-II BOP 301 and 302R tests using the CATHARE code (see Figure SFR-3). The results have been compared to

Figure SFR-3: CATHARE inlet and outlet core temperatures for BOP-301 BOP-302



Source: CEA.

experimental results. This comparison showed a good agreement in thermal-hydraulics results, but improvements of the core modeling are needed for the BOP 302R test: the core power is over-estimated in the CATHARE calculation (point kinetics). These results will be provided for a detailed comparative analysis. A document including multiscale results will be issued in 2022.

Within the framework of the H-2020 the European Sodium Fast Reactor Safety Measures Assessment and Research Tools (ESFR-SMART) project, Euratom performed the assessment of the initial core performance and burn-up calculations for the new ESFR-SMART core design, which was developed through the safety-related modification and optimization of the ESFR-working horse core design from the 7th Framework Programme (FP7) Collaborative project on European Sodium Fast Reactor (CP-ESFR). The deliverable discusses the neutronics characterization of this new ESFR-SMART core, including the evaluation of the safety-related neutronics parameters of the fresh core; once-through burn-up calculation; preliminary assessment of the impact of Pu and minor actinides (MAs) recycling on the reactivity and safety related parameters and realistic batch-wise burn-up calculation and establishing an equilibrium core loading pattern. A substantial effort was also dedicated to calibration and verification of the computer codes used in the analyses.

The JAEA has been developing an advanced computer code, SIMMER-III/IV, for the analysis of a core disruptive accident. For the validation of the SIMMER-III code, the JAEA presented a comparison of analysis results for an inhibiting effect of non-condensable gas on the condensation. The JAEA has applied the validated model to a series of transient large-scale single bubble condensation behavior experiments, and the analysis has indicated a lesser inhibition effect of non-condensable gas on condensation at the bubble surface which is important in safety assessments. A series of numerical studies have increased the reliability of the SIMMER-III code in SFR severe accident analyses.

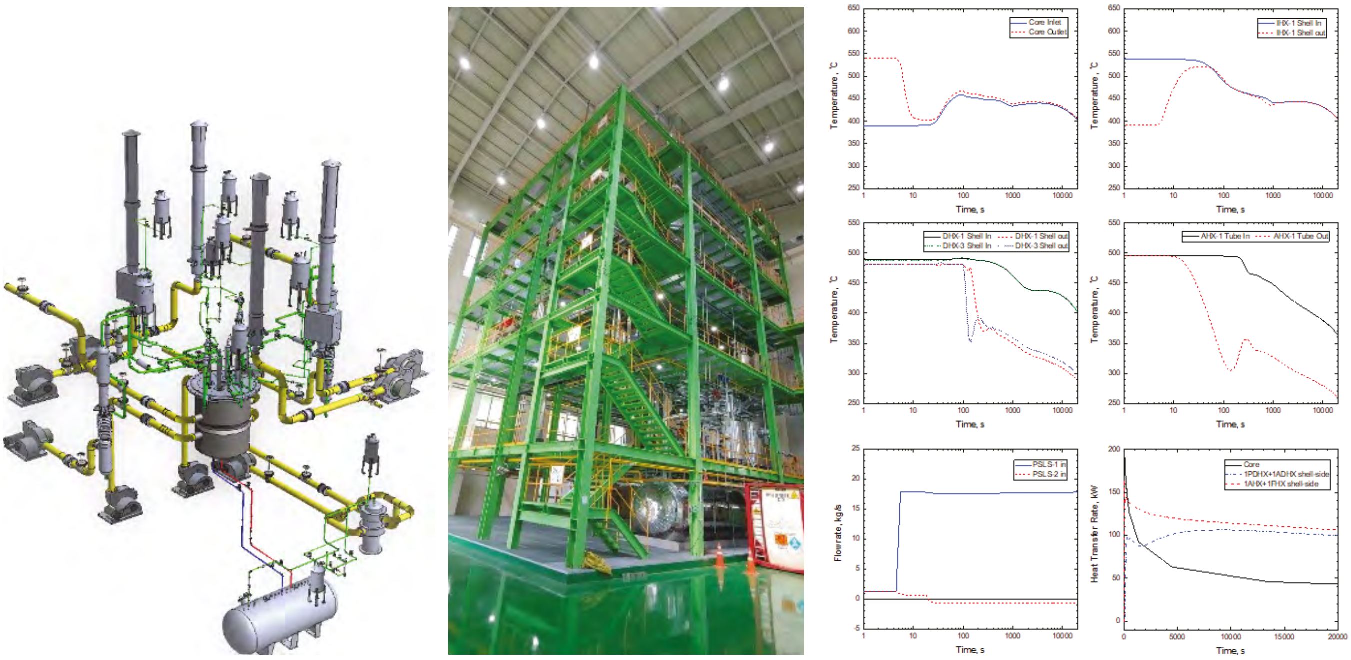
The JAEA also presented the results of the assessment during a postulated, unprotected loss-of-flow (LOF) accident for small-scale SFRs to show that the decay heat generated from the relocated fuels would be stably removed in the post-accident-material-relocation/post-accident-heat-removal phase, where relocated fuels mean fuel discharged from the core into low-pressure plenum through control-rod guide tubes and fuel remnants in the disrupted core region (non-discharged fuel). As a result of the assessments, it can be concluded that the stable cooling of the relocated fuels was confirmed and the prospect of in-vessel retention was obtained.

Rosatom performed a computational analysis of China experimental fast reactor (CEFR) spent fuel assembly mock-up heating with fuel element simulators in a closed volume. Comparison was made of the results of the thermal analysis calculations with the results of experiments at an experimental facility studying the heat transfer between spent fuel and the argon atmosphere during unloading from the CEFR. A total of nine steady-state test modes were considered: three power levels and three environmental temperatures. The TWSG code calculates the time-dependent temperatures of elements in the cross section of the active part of the fuel assembly located in the tube of the handling mechanism during its residence in a gas environment. A good consistency of the experimental and calculated values was obtained. By calculating the temperature and heating time of the spent fuel assemblies, it is shown that overheating can be avoided during transport through the argon atmosphere in the reactor building.

For the experimental program's work scope, Euratom, within the framework of the H-2020 ESFR-SMART project, performed experimental tests at two facilities on the interaction between corium jet simulant and sacrificial solid material aimed at the study of the ablation process of the ESFR-SMART core catcher. The post-test analyses of the tests have been compiled. At the HAnSoLO test facility, located at the University of Lorraine in the Theoretical and Applied Energetics and Mechanics Laboratory, tests were performed using simulants: hot water jet for corium and transparent ice for core catcher material. The main advantage of using transparent ice is to visualize the ablation process and the formation of the cavity, which is important to better understand the physics and to develop heat transfer correlations, according to the kinetics of ablation for the design of the core catcher. This experimental test section has been designed to capture the entire process of ablation starting from the impact of the incoming jet, beginning of ablation and finally to the formation of the cavity. At the MOCKA test facility located at Karlsruhe Institute of Technology, two experiments, JIMEC-1 and JIMEC-2 (Figure SFR-4), were performed with simulants: –metallic melt jet and ablated metallic material of the core catcher – to reproduce the jet impingement behavior on a thick metal wall in the presence of the so-called "pool effect". In both experiments, the ablation process slowed down after the melt jet mass was first

Figure SFR-4: The JIMEC-2 test setup

Source: JRC.



Source: KAERI.

Figure SFR-5: STELLA-2 facility and experiment results

retained in the molten pool inside the ablated pit. The ablation velocity before and after this event was constant. Given this pool effect, the deep melt pool prevents the direct energetic contact of the melt at the pit bottom. The main conclusion of the experiments is that the ablation rate of a metallic plate by a high-temperature and large-diameter metallic jet with moderate velocity is very high (more than 1 cm/sec). The jet diameter has no distinguishing effect on the average ablation rate. A small jet diameter can lead to a slightly faster ablation before the pool effect, and an earlier timing of the pool effect.

KAERI conducted experiments of the representative transient in the prototype Generation IV sodium-cooled fast reactor (PGSFR) with various initiating events using the Large-scale Sodium Integral Effect Test Facility, STELLA-2 (see Figure SFR-5). The target transient was LOF with loss of off-site power conditions, and the test matrix included various combinations of decay heat removal system (DHRS) failure modes and asymmetrical PHTS flow conditions. The experiment result will be shared within the safety and operations Project for a future benchmark study, and additional/supplementary experiments are scheduled in 2022.

Innovative design and safety systems

The CEA has presented studies related to a new core design called CADOR. The objective of the CADOR concept is to prevent violent mechanical energy releases in the case of fast transients overpower. Excessive power excursion is prevented through the design of a core with a large inherent Doppler reactivity feedback. Furthermore, the fuel temperature in normal operating conditions is minimized in order to maximize the Doppler feedback integral between the nominal and the fuel melting conditions. The design of the core and the verification of its associated safety criteria depend on the precision of the Doppler constant and of the fuel temperature evaluations. To estimate the importance of the fine prediction of these two parameters, three main transients were studied: a fast transients overpower caused by a gas bubble flow through the core, a fast transients overpower caused by the break of the core support structure and an unprotected LOF. Sensitivity studies made it possible to define the first margins on Doppler constant and fuel temperature evaluations to avoid fuel melting or sodium boiling during these transients.

Technical highlights – advanced fuels project

The advanced fuel (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 AF project consists of three work packages: SFR non-MA-bearing driver fuel evaluation, optimization and demonstration; MA-bearing transmutation fuel evaluation, optimization and demonstration; and high-burn-up fuel evaluation, optimization and demonstration.

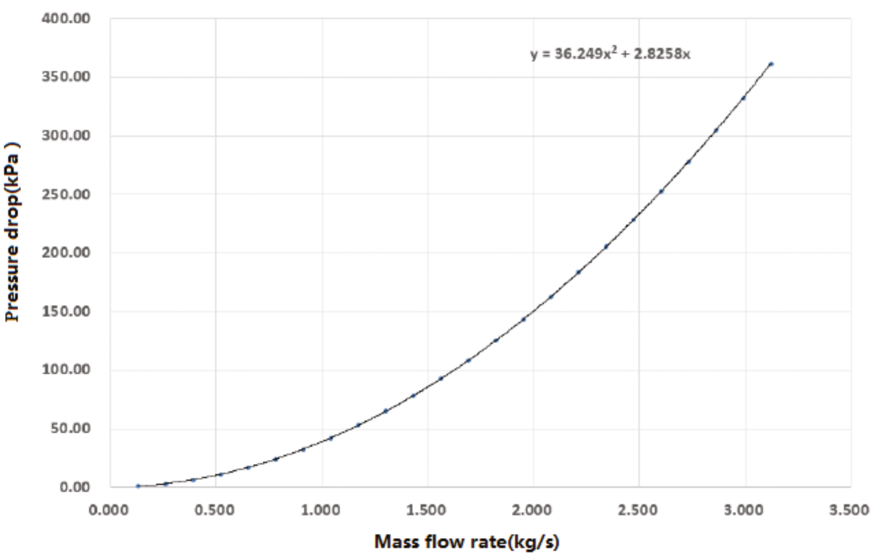
The China Institute of Atomic Energy (CIAE) is preparing to undertake some irradiation tests. It has finished the fabrication of dummy irradiation assemblies and out-of-pile hydraulic tests (hydraulic characteristic experiments). In 2021, out-of-pile hydraulic and mechanical supplementary tests were conducted to ensure the future safety of in-pile irradiation assemblies (see Figure SFR-6 below).

The CEA focused on the improvements in the simulation of the thermal-mechanics and thermal-hydraulics behavior of the fuel pin bundle under irradiation, knowledge of which is necessary for safety assessments. Since the phenomena responsible for the deformation of the fuel bundles, namely the swelling and creep of the cladding, strongly depend on temperature, a coupling between the thermal hydraulics and thermal mechanics in the fuel assembly exists. A novel methodology has been proposed, based on coupling between DOMAJEUR2, the CEA code computing the diametral deformation of the claddings, and the industrial CFD code STAR-CCM+, computing the associated temperature distribution in the bundle. The sodium mass flow rate reduction in the deformed geometry drives a sodium temperature increase resulting in a strain reduction for the cladding, compared to a non-coupled simulation (Figure SFR-7) (Acosta et al., 2019). Lastly, an efficient method to estimate the fuel cladding temperature evolution based on a limited number of CFD simulations has also been proposed Acosta et al., 2021).

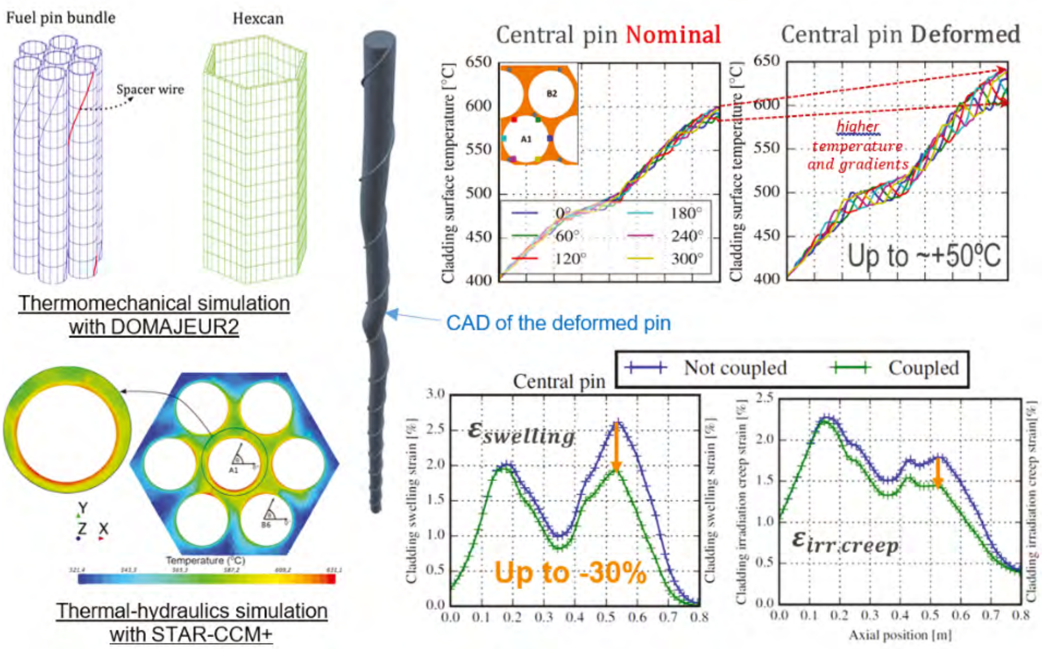
The Euratom Joint Research Centre recently proposed an innovative approach for the safe and secure synthesis of oxide based nuclear fuels. The new method is based on the hydrothermal decomposition of oxalates at low temperature, and it significantly reduces the required energy and number of process steps compared to currently applied methods. Within the SFR AF project, hydrothermal decomposition has been applied, for the first time, to the production of a nanometric-sized solid solution of the target composition $U_{0.90}Am_{0.10}O_2$ and $U_{0.80}Am_{0.20}O_2$, in order to demonstrate its feasibility towards fuels for the heterogeneous transmutation strategy. The fabricated discs were fully characterized by visual inspection and density determination, as well as by crystallographic and chemical analysis methods. The material showed a good sintering behavior without cracking and high densities up to 96% theoretical density.

KAERI has been developing metal fuel for the PGSFR. The fuel assembly is being designed to satisfy its requirements for core performance and safety. To this end, event categories are classified as non-operation and normal operation, anticipated operational occurrences and design-basis accidents, and the stress limits for each event category were established. Also, the allowable crack size is determined when the brittle fracture of a duct is of concern during fuel assembly handling. The details of the development process and results of the stress limits, together with the design criteria, was described. Crucible candidate material using cold isostatic press followed by a sintering process has been fabricated to develop reusable crucible for metal fuel cast. Sintering conditions are controlled to exhibit single-phase ceramic structure. The interaction-preventing effect of the sintered body is inves-

Figure SFR-6: Relation curve of pressure drop and mass flow rate of dummy irradiation assembly



Source: CIAE.



Source: Acosta et al., 2019 and 2021.

Figure SFR-7: Coupled thermomechanical and thermal-hydraulics simulation in a 7-fuel pin bundle

tigated using the sessile drop test. U-Zr-RE melt was fabricated for the sessile drop test via cast. Interaction behaviors and the defects of the ceramic body were analyzed after the droplet test.

The CIAE will conduct a program to undertake 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 finished (Figure SFR-8). The mechanical properties of CN-1515 and CN-FMS have been tested and have been used to evaluate the irradiation assemblies.

The JAEA observed the microstructure of 9Cr ODS-tempered martensitic steel cladding and 12Cr-ODS ferritic steel cladding after very long-term internal pressurized creep tests. The stability of the martensite and ferrite matrix structure, including carbide distribution, dislocation density and lath boundary, and that of nano-particle distribution was investigated (Oka. H et al., 2021).

KAERI developed a barrier cladding tube technology to suppress fuel-cladding chemical interaction for the usage of MA-bearing metal fuel. As a part of the option, duplex cladding is applied at the inner surface of the cladding tube lined with 50 µm thickness of Zr at the 500 mm length of HT9 cladding. Evaluation of the manufacturing parameter for fabrication of the Zr liner and HT9 tube, as well as the assembling process, was reported.

Technical highlights – component design and balance-of-plant project

The component design and balance-of-plant (CD&BOP) 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, new sodium testing capabilities and decommissioning. The main activities in ECS include: 1) the development of advanced, high-reliability steam generators and related instrumentation; 2) the development of advanced ECS based on a Brayton cycle with supercritical CO_2 or nitrogen as the working fluid; and 3) the study of decommissioning of sodium facilities for guidance on future plant design.

The CEA has studied the capability of leaky Lamb waves in view of inspection from the outside of the main vessel. The first phase involved studying the propagation properties of leaky Lamb waves (equations, phase velocity, group velocity). Then, dispersive and multi-modal propagation was highlighted (with attenuation of leaky Lamb waves due to the re-emission of energy into the surrounding fluid). Now that the propagation of leaky Lamb waves in a second plate is established, the next step is to identify parameters of interest for non-destructive testing in the second plate and then to optimize them. During the last ten years, the CEA List Institute developed its own electromagnetic acoustic transducer (EMAT) sensors for the purpose of inspecting SFRs, which must then operate in sodium at 180°C. An EMAT sensor consists of generating ultrasound in a conductive medium with the help of Lorentz forces, created by the superposition of a permanent magnetic field and eddy currents. The advantage of this type of sensor is that it avoids the problem of wetting. The first sensors evaluated at the CEA were commercially available single-element sensors, but the CEA quickly began to develop and design more efficient multi-element sensors using the CIVA simulation software, which made it possible to improve the arrangement of magnets and coils in order to improve signal quality. The various prototypes, some of which have been evaluated in sodium, are presented in this report.

In relation to the under-sodium viewing technology, KAERI started to develop the plate-type ultrasonic waveguide sensor array. In 2021, as a first step, KAERI conducted preliminary studies to design the waveguide sensor array. A transmitting waveguide sensor and various shaped reflective targets were fabricated, as well as a test facility for acoustic field measurements to investigate radiation characteristics of target reflected waves. Then, based on the investigation results, positions and numbers of receiving waveguide sensors in the array were selected. In addition, KAERI also investigated the performance of the receiving waveguide sensor according to the size variation of the receiving face to find the optimal sensor size.

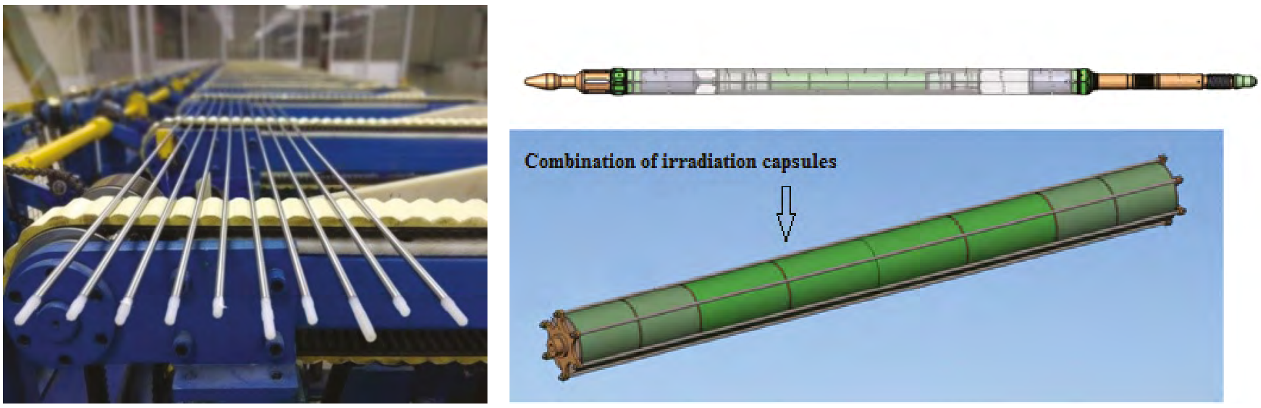
JAEA developed a failed fuel detection and location (FFDL) system in a sodium-cooled large fast reactor. The role of FFDL is to identify the failed fuel subassembly. For the better performance of FFDL, the sampling method for identifying failed fuels under the upper internal structure slit (UIS) is especially important because direct sampling nozzles for measurement cannot be installed above subassembly. In this study, sampling performance for the subassemblies under the UIS slit has been evaluated. In addition, the detection capability of the FFDL was presented to achieve design conditions and operation modes, and procedures of the FFDL system were also investigated.

The CEA worked on the SOVEXP model, it was developed to represent the sodium-water reaction (SWR) as a fast dynamic process that generates pressure waves carrying part of the reaction energy. This model applies to configurations that can be explosive, especially when water is in excess compared to the sodium amount at stake. SOVEXP can be used to represent domains that are closed (overpressure calculation) or not (shock waves). In this work, SOVEXP is described and applied to closed domains, representing the CREON, EVAPO and AUTOCLAVE experimental campaigns performed at the CEA. Its results are compared to the experimental ones, exhibiting in particular the stepwise nature of the SWR (i.e. the sodium mass does not react all at once).

In 2021, the DOE conducted the operations of an intermediate-scale sodium test facility for the purpose of testing systems and components (mechanisms) in prototypical sodium environments. This facility, called the Mechanisms Engineering Test Loop (METL) facility consists of four experiment test vessels (of two sizes). Sodium is fed to these test vessels from a main loop with a storage tank. The test vessels have been designed to provide an independent testing environment from each other, if isolated from the main loop. The facility has been continuously operational from September 2018. It was drained and frozen for the first time in April 2021 in order to conduct repairs on the whole building's fume scrubbing system. The water piping system for the scrubber was deteriorating and required replacement. METL facility operations were restored in the third quarter of 2021 after being suspended for about six months.

The DOE described in 2021 the design and construction of a new experimental test article called the

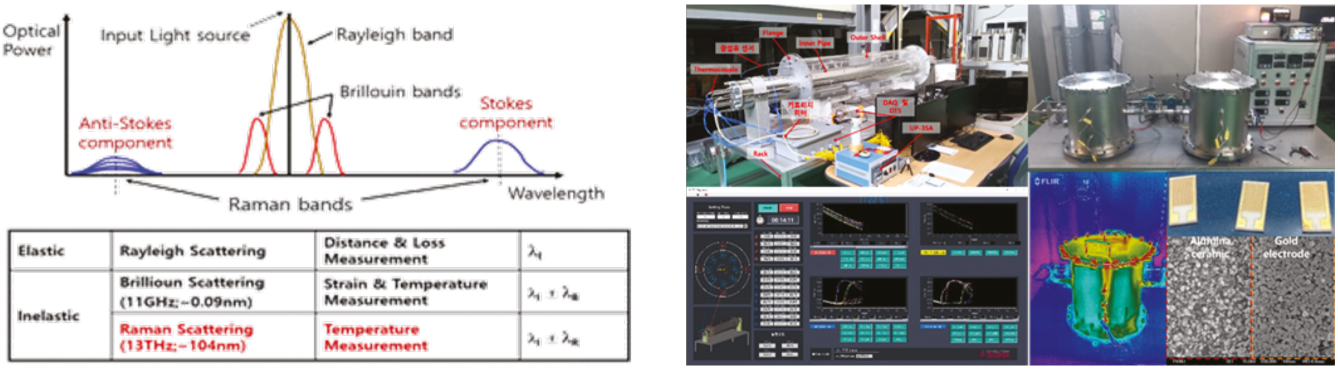
Figure SFR-8: Cladding tubes of CN-1515 and the irradiation assembly of CN-1515 cladding tubes and CN-FMS hexagonal ducts



a) Cladding tubes of CN-1515.

(b) Irradiation assembly of CN-1515 and CN-FMS.

Source: CIAE.



Source: KAERI.

Figure SFR-9: Progress of the development of advanced instrumentation techniques for liquid sodium applications

Thermal Hydraulic Experimental Test Article, which is designed as an electrically heated pool plant and will be used for the validation of thermal-hydraulic codes. The facility consists of a mechanical centrifugal pump, a submerged flow meter, an electrically heated core, an intermediate heat exchanger and an intermediate heat transport system, as well as multiple conventional temperature sensors and specialized fiber distributed temperature sensors.

KAERI continued a study to develop advanced sodium instrumentation techniques based on high-fidelity distributed temperature sensor and ultrasonic technology (Figure SFR-9) (Ferdinand, 2014). The innovative techniques are expected to be utilized to determine some useful ways for measuring and monitoring various process variables, as well as for better operation of large-scale sodium systems or facilities. The set of specific sodium technologies would also be applied to sodium feeding, draining, accident prevention, high-temperature operation and measurement, etc.

The CEA has provided information on the decommissioning of Rapsodie, Phenix and Superphenix reactors. For the dismantling strategy regarding sodium (Na), the Na-cleaning process was described (use of NaOH with the CEA patented carbonation and washing hydrolysis); the Na cleaning process validation with supporting experiments was also presented.

The JAEA gave information on the experience of the upper core structure exchange in Joyo, which was performed in 2014 as a repair technology, information on dismantling.

References

Acosta F., et al. (2021), “Application and Benchmarking of a Novel Coupled Methodology for Simulating the Thermomechanical Evolution of Sodium-cooled Fast Reactors Fuel Subassemblies” *Nuclear Engineering and Design*, 374, 111048.

Acosta, F., et al. (2019), “A new thermal-hydraulics/thermomechanics coupling methodology for the modeling of the behavior of sodium-cooled fast reactors fuel subassemblies under irradiation” *Nuclear Engineering and Design*, 348, 90–106.

Ferdinand, P. (2014), “The evolution of optical fiber sensors technologies during the 35 last years and their applications in structure health monitoring”, EWSHM-7th European Workshop on Structural Health Monitoring.

Oka, H., T. Tanno, Y. Yano, S. Ohtsuka, T. Kaito and Y. Tachi (2021), “Microstructural stability of ODS steel after very long-term creep test”, *Journal of Nuclear Materials*, 547, 152833.

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

Very-high-temperature reactor

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

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

Main characteristics of the system

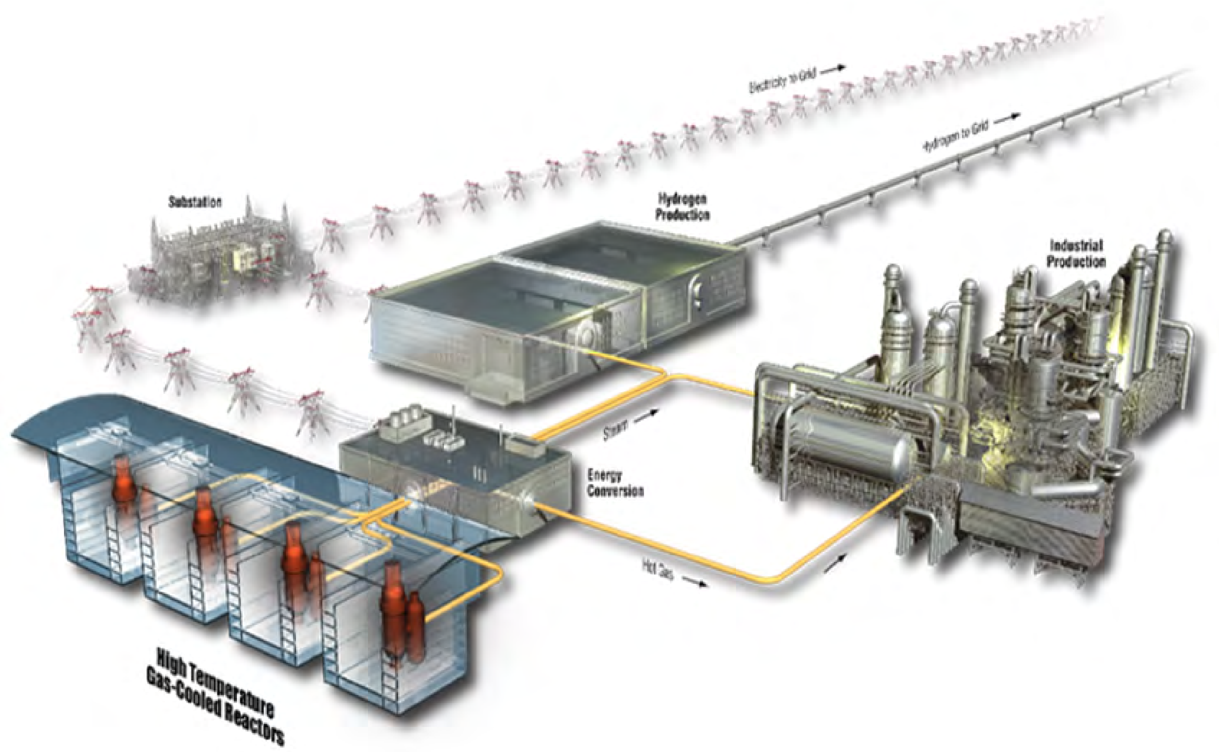
High-temperature gas-cooled reactors (HTRs or HTGRs) are helium-cooled graphite-moderated nuclear fission reactors using fully ceramic tri-structural isotropic (TRISO) coated particle-based fuels. They are characterized by inherent safety features, excellent fission product retention in the fuel, and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

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

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

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

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



Source: Gougar, H., (2014).

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

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

R&D objectives

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

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

Technical highlights – fuel and fuel cycle project

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

PIE and safety testing of fuel specimens from the advanced gas-cooled reactor-2 (AGR-2) UCO and UO_2 TRISO fuel irradiation experiment were completed in the United States in 2021. A final report has been issued on the results (Stempien et al., 2021), which represents a seven-year effort that included extensive destructive examination of 14 as-irradiated fuel compacts and particles, and 16 high-temperature post-irradiation safety tests at temperatures of 1 500°C to 1 800°C with subsequent destructive examination of the fuel specimens. Over the course of this work, thousands of individual particles were gamma counted and examined microscopically with a variety of techniques. The combined results have provided extensive information on the performance of UO_2 and UCO TRISO fuel particles across a broad range of conditions, as well as key data on the transport of fission products.

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

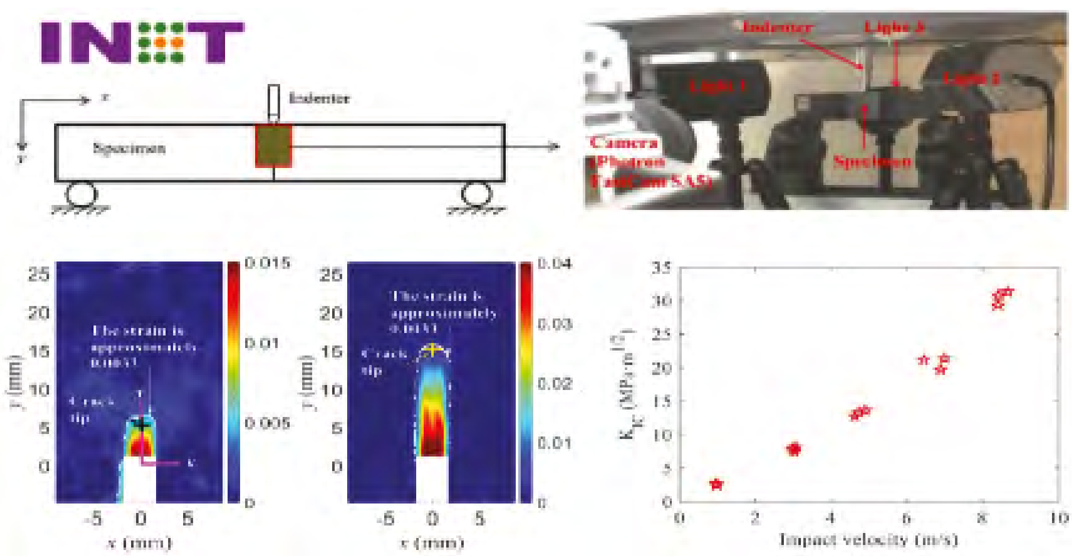
PIE of fuel pebbles discharged from the Chinese HTR-10 reactor is in progress at the newly commissioned hot cells at the Institute of Nuclear and New Energy Technology (INET). It includes burnup measurement and pebble deconsolidation. Hot testing of the new KÜFA furnace at INET, and subsequent accident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

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

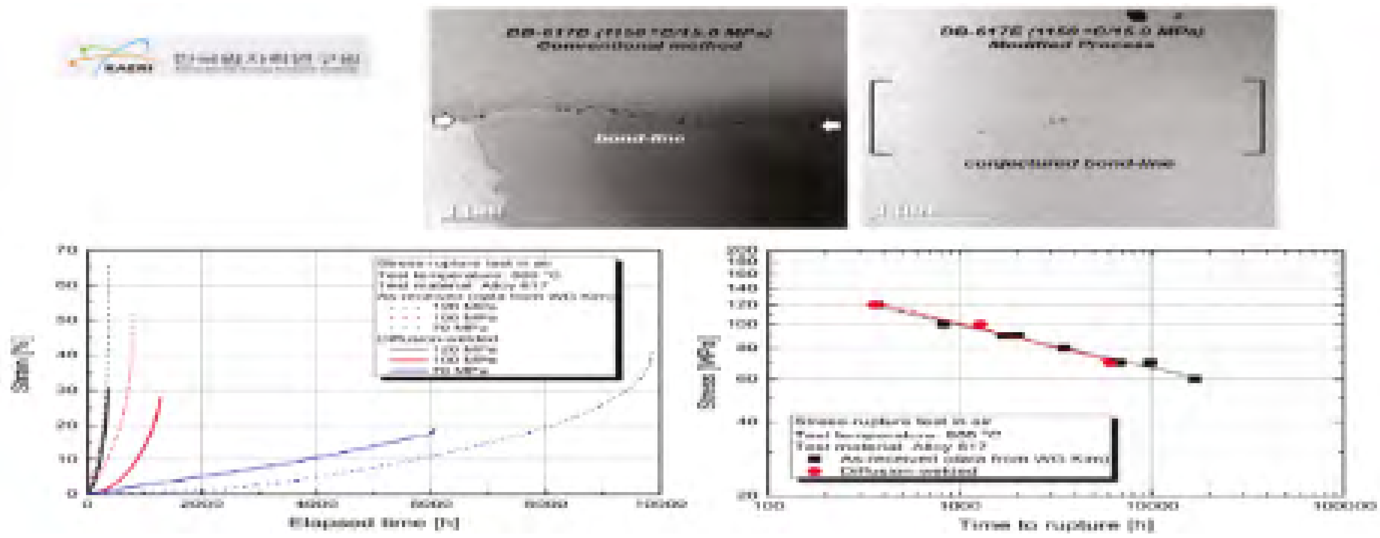
Several PMB members are pursuing the development of advanced TRISO fuel. In China, studies on UCO fuel kernel processing and fabrication of ZrC coatings are in progress. KAERI is currently developing fabrication methods for large (~800-μm-diameter) UO_2 kernels (with potential application in accident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

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

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



Source: INET.



Source: KAERI.

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

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

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

Technical highlights – materials project

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

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

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

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

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

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

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

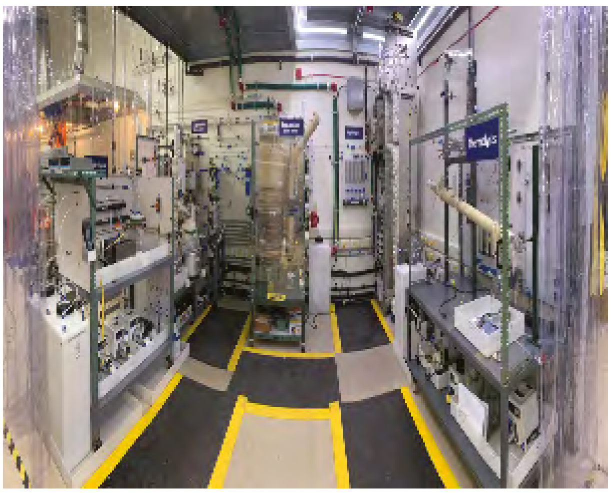
In the near/medium term, metallic alloys are considered as the main option for control rods and internals in VHTR projects, which target outlet temperatures below about 850°C. However, future projects are considering the use of ceramics and ceramic composites where radiation doses, environmental challenges,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

Technical highlights – hydrogen production project

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

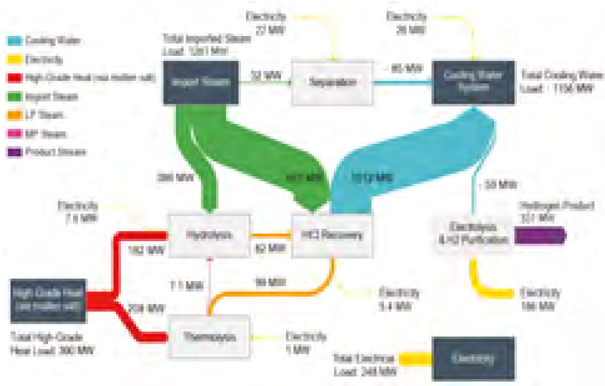
Over the last year, the Canadian effort has focused on setting up in laboratory a 50 NL/h hydrogen production system based on the copper-chlorine cycle. Figure VHTR-4 below shows the four steps of the process set up for integrated continuous operation.

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



Source: CNL.

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



Source: CNL.

Source: INET.

Figure VHTR-6: Prototype of the sulfuric acid decomposer

Source: INET.

Figure VHTR-7: Installation of the helium loop

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

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

France has made the decision to accelerate the industrialization of the high-temperature electrolysis system to produce clean hydrogen from nuclear energy and renewable energy. Thanks to the R&D efforts of the CEA, developments have passed the first generation of cell and stack, and the manufacturing step at industrial scale is in progress with a new public-private partnership to develop a pilot line and produce high-power modules of stacks. The CEA is now developing a second generation, higher performance and durability cell by combining numerical and experimental approaches at different scales from raw material to the single cell. Through modeling and characterization of the microstructure using the European Synchrotron Radiation Facility, it has been possible to predict the performance of the cell by incorporating a mass transportation model. The CEA also performed long-term tests (over 10 000 h) and proposes a mechanism of degradation of the cell by migration of Zr. Indeed, a loss of Zr4++ from the electrolyte might be due to the formation of zirconates that could facilitate the inter-diffusion of gadolinium (Gd), reducing the local ionic conductivity and thus significantly contributing to the largest increase in the ohmic resistance observed. Figure VHTR-8 shows the observed phenomenon. The next critical step focuses on the cell fabrication to increase reproducibility on stack and to reach a degradation rate of <1%/1 000h @ 1.3V and 0.85 A/cm^2 on a larger cell area (200 cm^2).

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

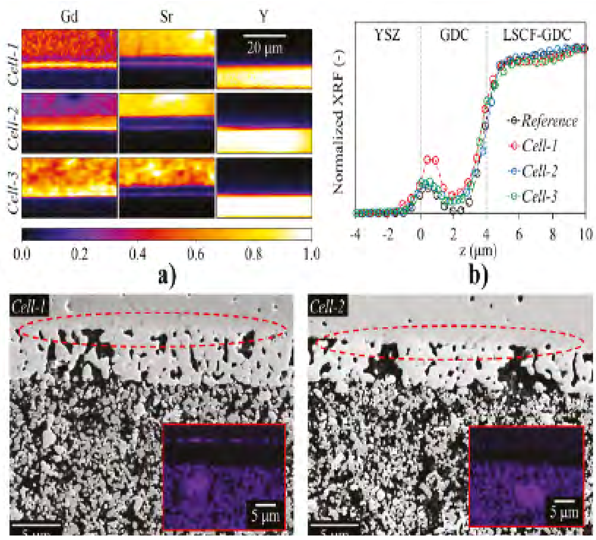
Comparative SO_3 splitting screening experiments with catalyst-coated SiSiC honeycombs, for periods between 100-950 hours onstream, demonstrated that platinum-based catalysts did not exhibit satisfactory conversion and suffered deactivation with prolonged on-stream exposure times at 650°C. This was attributed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

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

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

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

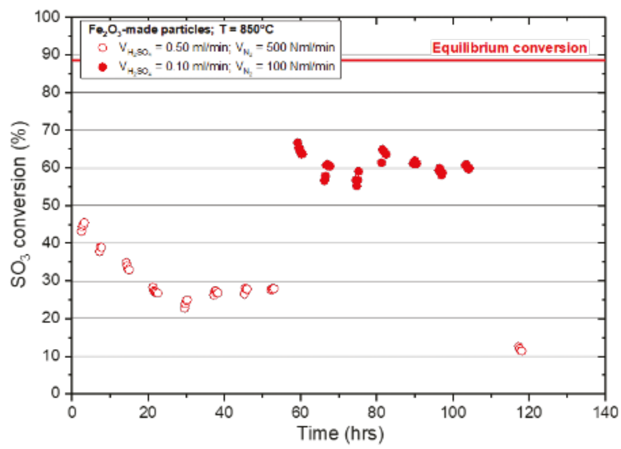
Figure VHTR-8: Analysis of operated cell



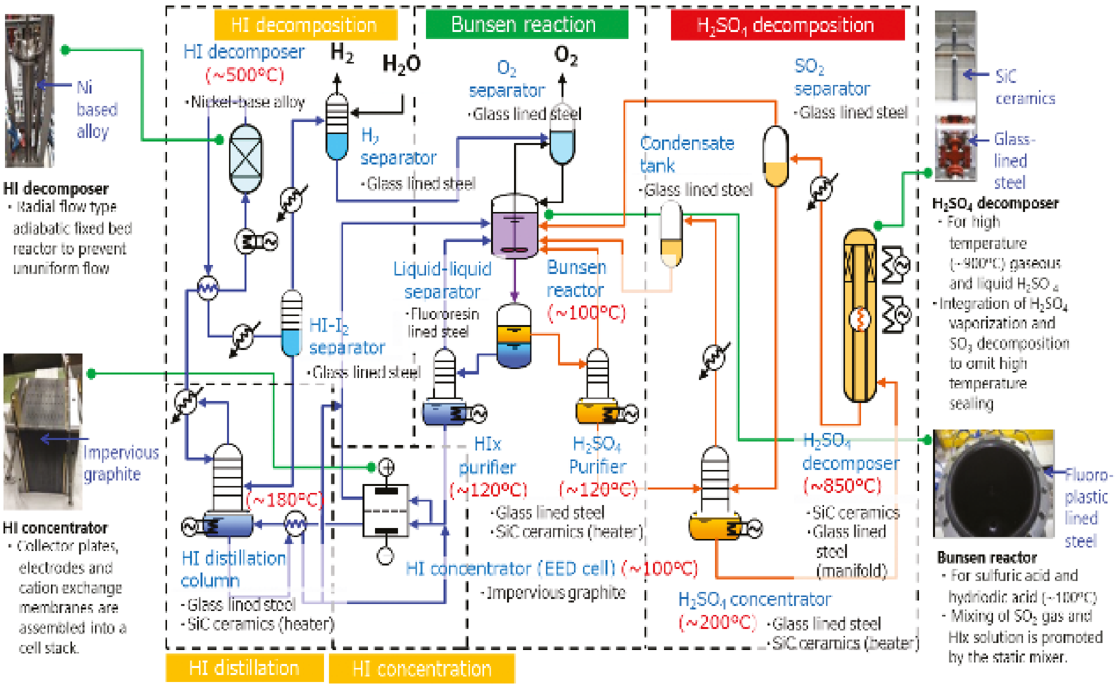
a) Normalized 2D element distribution obtained from the WRF signal at the electrode/barrier layer/electrolyte interface for cell-1, cell-2 and cell-3; b) Comparison of the integrated XRF signal of Sr between the pristine cell and the aged samples; and c) SEM images of the LSCF-GDC/GDC/8YSZ region for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

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

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



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



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

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

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

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

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

Technical highlights – computational methods validation, and benchmarks project

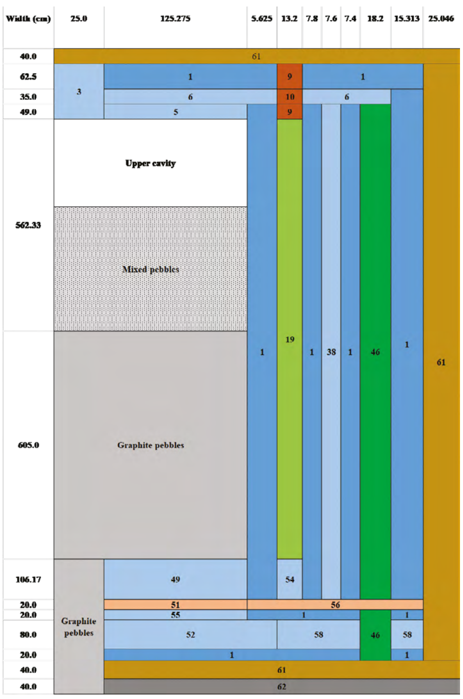
Validation of new computational methods and codes in the areas of thermal hydraulics, thermal-mechanics, core physics and chemical transport are needed for the design and licensing assessment of reactor performance in normal, upset and accident conditions. Code validation needs will be carried out through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

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

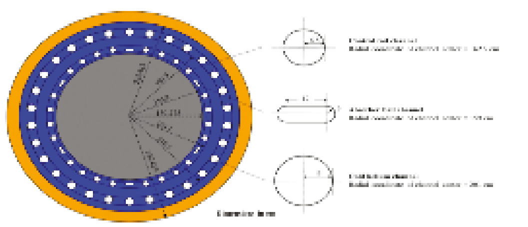
In Japan, the JAEA restarted the operation of the HTTR, a 30 MW research reactor on 30 July 2021, representing another major milestone for international HTGR activities in 2021. Safety demonstration tests under the framework of a OECD/NEA project will be carried out in 2022. There are also plans to perform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

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

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



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



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

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

References

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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