Chapter 4. **System reports**

Gas-cooled Fast Reactor (GFR)

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

The reference concept for GFR is a 2 400 MWth plant operating with a core outlet temperature of 850°C enabling an indirect combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high power density necessary for good neutron economy in a fast reactor core. This means the biggest challenge in the development of GFR system. Less significant challenge for GFR is to ensure the decay heat removal in all anticipated operational and fault conditions. Therefore, in the development of commercial GFR it is necessary to establish a type of experimental demonstration reactor for qualification of the refractory fuel elements and for full-scale demonstration of the GFR-specific safety systems. Actually, the ALLEGRO project reactor is to be the proposed demonstration reactor for the reference GIF GFR concept.

The ALLEGRO Gas-cooled Fast Reactor Demonstrator project

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

The original design of the ALLEGRO consists of two He primary circuits, three decay heat removal (DHR) loops integrated in 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 function not only as a demonstration reactor hosting GFR technological experiments, but also as a test pad for using the high temperature coolant of the reactor in a heat exchanger for generating process heat for industrial applications and a research facility which, thanks to the fast neutron spectrum, makes it attractive for fuel and material development and testing of some special devices or other research works.

The 75 MWth reactor shall be operated with two different cores (see Figure GFR.2). The starting core with 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. The second core will consist solely of the ceramic fuel and will enable to operate ALLEGRO at its high target temperature.

Central European members of the European Union, the Czech Republic, Hungary and the Slovak Republic are traditionally prominent users of nuclear energy. They intend to use nuclear energy on the long run and besides the lifetime extension of their nuclear units, each country decided to build new units in the coming years. Therefore, four nuclear research institutes and companies of the Visegrad-4 region (ÚJV řež, a.s. – Czech Republic, MTA EK – Hungary, NCBJ – Poland, VUJE, a.s. – Slovak Republic) decided to start joint preparations aiming at the

construction and operation of the ALLEGRO demonstrator of the concept of Generation IV Gas-cooled Fast Reactor (GFR) based on a Memorandum of Understanding signed in 2010. CEA, France, as promoter of the GFR concept since 2000, supports the joint preparations, bringing its knowledge and its experience in building and operating experimental reactors, in particular fast reactors. In order to study safety and design issues and also the medium and long-term governance and financial issues, the four aforementioned organizations created in July 2013 a legal entity, the V4G4 Centre of Excellence, which performed the preparatory works needed to launch the ALLEGRO Project. V4G4 Centre of Excellence is also in charge of the international representation of the project. As a result of the preparatory works it turned out that during the earlier works certain safety and design issues remain unsolved and in several aspects a new ALLEGRO design has to be elaborated. Therefore in 2015, when the ALLEGRO Project was launched, a detailed technical program was established with a new time schedule.

Figure GFR1. The GFR reactor system

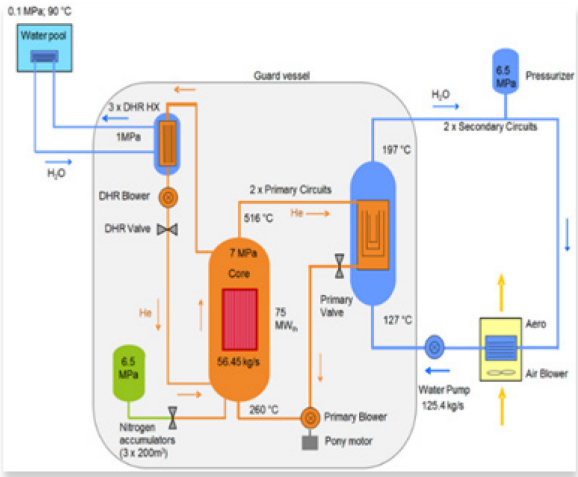
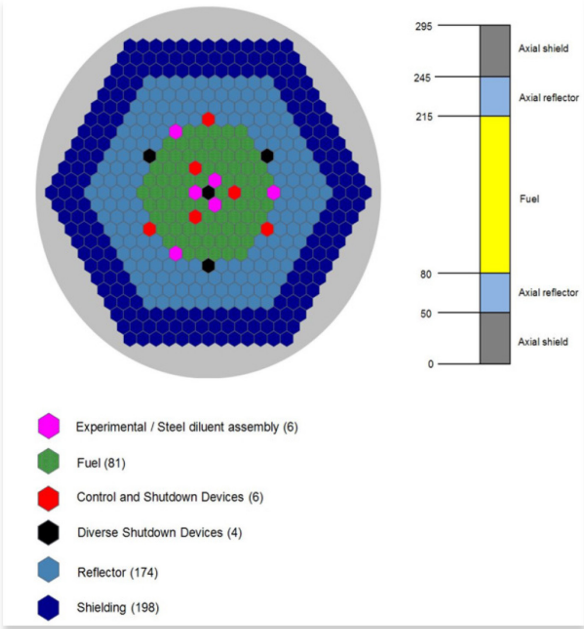


Figure GFR 2. The GFR core concept



Fuel cycle and fuel

Fuel development efforts must be conducted in close relation with reactor design efforts so that both the fuel meets core design requirements and that the core operates within fuel limits. Technology breakthroughs are needed to develop innovative fuel forms, which

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

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

Candidate fuel types already identified are:

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

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

Data on potential ceramic (particularly SiCf/SiC) and refractory alloys for cladding materials are patchy. These materials need to be adapted in order to cope with the different loads (thermal gradients, interaction fuel-barrier, dynamic loads, etc.), which means that their composition and microstructure need specific dedicated developments.

The main goal of the high temperature experiments is the investigation of the behavior of 15-15 Ti 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 load bearing capability of cladding after high temperature treatments. The cladding microstructure will be examined by SEM and metallography.

The development of qualification procedure for start-up fuel will include the specification of the steps for MOX/UOX fuel with 15-15 Ti 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. The validation of the code should be based on sodium-cooled fast reactor fuel past histories.

Testing of SiC claddings in high temperature helium will be carried out to track the potential changes. Mechanical testing and the examination of microstructure with SEM and metallography is planned with the samples after high temperature treatment. In particular, Ion irradiation effect on SiC composites will be investigated in order to evaluate the importance of the significant volume change observed for Hi-Nicalon type-S fiber and C fiber coating. High dose ion irradiation will be carried out with various temperature range including GFR operating temperature for SiC composites. High dose irradiation effect on SiC composites will be examined.

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

The following topics will be analyzed in short term:

Design of the ALLEGRO reactor core:

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

Development of fuel behavior codes for ALLEGRO fuel:

- Collection of material data for fast reactor materials.
- Derivation of reactor physical parameters needed for the FUROM code.
- Implementation of fast reactor material data in the FUROM code.

Tasks related to ALLEGRO fuel qualification and specification:

- ALLEGRO fuel related acceptance criteria.
- Review of fuel candidates for the first core of ALLEGRO.

- Selection of the components of optimal ceramic fuel for ALLEGRO.
- Development of ceramic fuel qualification procedure.

Tasks related to research on fuel materials:

- Review of SiCf/SiC cladding materials.
- Testing UOX/MOX fuel cladding in high temperature He.
- Mechanical testing of UOX/MOX fuel cladding.

Advanced components and materials

Concerning in-core structural materials for the GFR (cladding, reflector, control rod guides, etc.), the main challenge is to develop materials able to withstand fast neutron induced damage together with high temperature ranges. Thus, ceramic materials (monolithic, composite) are the reference option and as a back-up selected composite cermet structures, refractory alloys, and inter-metallic compounds. In addition, the reflector should have specific neutronic properties to reduce neutron leakage efficiently and to protect the surrounding vessels; an inter-metallic compound of Zr and Si is the favored at this stage for this component.

Special issues and technologies

The improvement of decay heat removal capabilities aims at defining and optimizing a simple and robust combination and sequencing of complementary fail-safe solutions during a fast or slow depressurization transient. The main topics to be addressed are:

- Increase core thermal inertia.
- Optimization of key design parameters to enhance natural convection. Determination of required back-up pressure.
- Optimization of DHR Systems:
  - Coupled primary-secondary turbomachines;
  - Injection systems, Heavy gas accumulators.
- Guard vessel and system containment.

Branislav Hatala

Chair of the GFR SSC and all Contributors

Lead-cooled fast reactor (LFR)

Preamble: For a sake of homogeneity among all the system reports within this Annual Report, this chapter has been intentionally synthesized in a reduced number of pages. The full extended version of the 2019 LFR system report with the complete list of publications can be uploaded on the GIF website.

Main characteristics of the system

The LFR features a fast neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of minor actinides, both self-generated and from reprocessing of spent fuel from light water reactors (LWR), and as a burner/breeder with thorium matrices. An important feature of LFR is the enhanced safety that results from the choice of a relatively inert coolant.

The system identified by GIF includes three reference concepts. The options considered are a large system rated at 600 MWe (ELFR EU) intended for central station power generation, a system of intermediate size (BREST 300 Russia), and a small transportable system of 10-100 MWe size (SSTAR US) that features a very long core life. The expected secondary cycle efficiency of each of the LFR reference systems is at or above 42%. These three GIF LFR reference concepts cover the full range of powers. It has therefore the potential to provide wide electricity needs: from remote or isolated sites or to serve as large inter-connected power stations. Important synergies exist among the different LFR systems, so that a co-ordination of the efforts carried out by participating countries is a the key point of LFR development. The typical design parameters of the GIF LFR systems are summarized in **Table LFR.1**.

Table LFR.1 **Key design parameters of 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	564
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 T (°C)	450	505	550

R&D objectives

The System Research Plan (SRP) for the LFR is based on the use of liquid lead as the reference coolant and lead-bismuth as the back-up option. Given the R&D needs for fuels, materials, and corrosion control, the LFR system is expected to require a two-step industrial deployment: demonstration reactors operating at relatively low primary coolant temperature and low power density by 2025; then high-performance reactors by 2040. Following the reformulation of GIF LFR PSSC in 2012 the SRP was completely revised, and an updated final draft is in preparation. The approach taken in the SRP is to consider the research priorities of each member entity, and

to propose a co-ordinated research programme to achieve the objectives of each member while avoiding unnecessary duplication of effort.

The committee notes that there are significant potential commonalities in research and design among these three system thrusts. The plan proposes co-ordinated research along parallel paths leading to one (or more) pilot facilities that can serve the R&D needs of the reference concepts. The needed research activities are identified and described in the SRP. Co-ordinated efforts can be organized in four major areas and formalised as projects: System Integration and Assessment (SIA); lead technology and materials; system and component design and fuel development.

System integration and assessment (SIA) project: The ultimate goal of the SIA project is to ensure the feasibility of the LFR system to meet GIF objectives, taking into account schedule and cost. The LFR SIA activities are to be carried through an iterative process aimed at ensuring that the R&D projects, either individually or jointly, satisfactorily address the GIF criteria of safety, economy, sustainability, proliferation resistance and physical protection.

System and component design project: System design activities are envisioned in the preliminary design of central station and small-scale plants, design of prototypes and demonstration plants, and co-ordination of cross-cutting activities including safety approach, component development, balance-of-plant, etc.

Fuel development project: The LFR fuel development project is a continuing long-term process consisting of tasks designed to meet progressively more ambitious requirements. It includes efforts in the areas of core materials development, fuel fabrication, fuel irradiation and tests aimed at fuel qualification. Strong synergies are existing with the parallel SFR fuel development.

In the near term, one essential goal is to confirm that at least some technical solutions exist so that fuel can be provided in an early time frame suitable for demonstration reactor systems. This “fuel for the Demo” milestone achievement will provide the assurance of the feasibility of a safe and competitive LFR for electricity production.

In the mid-term, it is necessary to confirm the possibility of using advanced minor actinide-bearing fuel at levels representative of the specified equilibrium fuel cycle in order to assure minimization of long-lived nuclear waste and fuel cycle closure. This second goal is therefore to confirm the possibility of achieving higher fuel burn-up compared with that reached in current liquid metal cooled reactors.

In the long term, it is important to confirm the potential for industrial deployment of advanced MA-bearing fuels, and the possibility of using fuels that can withstand high temperatures to exploit the advantage of lead (margin to boiling) to increase plant efficiency for electricity generation and also provide of high-temperature heat production. This “advanced high temperature fuel” milestone achievement will demonstrate the sustainable, multipurpose capability of the LFR technology.

Lead technology and materials project: In the near term it is necessary to maximize the use of available materials thereby limiting material qualification activities only in their new environment. To establish reactor feasibility, it is necessary to provide a technologically viable structural material capable of withstanding the rather corrosive/erosive operating conditions of an LFR. In the mid- and long term, the high boiling point of lead is advantageous for high temperature operations of the reactor, extending the LFR mission towards higher efficiency cycle and hydrogen production. Those missions require the development of new materials both for mechanical components and fuel cladding, or industrial processes to protect existing materials (coatings). These material developments will be time consuming and will be carried out with a flexible schedule depending on investments and technological achievements. Peculiar is the development of a fuel cladding resistant to high neutron doses (increased fuel burn-up) and at high temperature (increased coolant temperature and power density).

Main activities and outcomes

During 2019, the LFR pSSC has been strongly involved in the drafting or revising of several GIF reports that are expected to be issued in 2020:

- *LFR – System Safety Assessment (SSA)*. In collaboration with RSWG, the first SSA draft report was finalized in December 2018 and sent to GIF experts in 2019. The final agreed version of the report will be issued early in 2020.
- *LFR – Safety Design Criteria (SDC)*. Throughout 2019, the LFR pSSC has worked on a revision of the LFR – SDC draft report based on comments received from RSWG members. The report has been updated and finalized. It will be transmitted back to RSWG in early 2020.
- *LFR – PRPP white paper*. A first draft of the PRPP paper has been developed in strong collaboration with PRPPWG. Following a dedicated meeting in December 2019, the document is now under finalization by the LFR pSSC. It is expected to issue it in 2020.

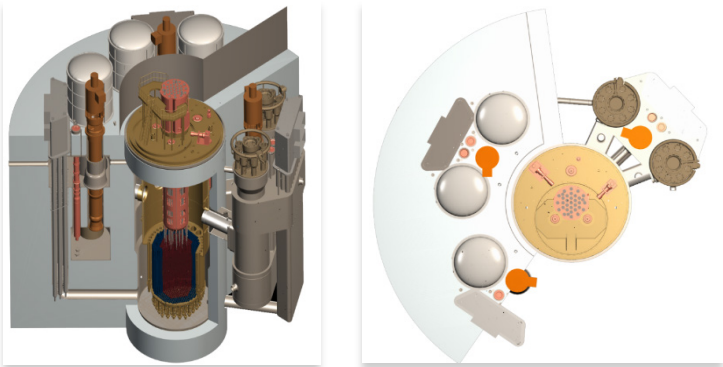
The LFR – pSSC has been also working actively with the GIF Task Force on Research Infrastructures and provided input to the Advanced Manufacturing Task Force (AMME). Finally, the LFR – pSSC was enlarged first in February 2018 by the LFR-MoU signature of USDOE, and then in Oct. 2019 by the signature of INEST (on behalf of the Chinese Academy of Sciences).

Main activities in Russia

The BREST-OD-300 fast neutron lead-cooled reactor (see **Figure LFR 2**) has been developed as the pilot and demonstration prototype of a baseline commercial reactor facility for future nuclear power. The BREST-OD-300 unit is intended for:

- practical confirmation of the key design approaches used in lead-cooled reactor facilities operating as part of a closed nuclear fuel cycle, and the fundamental principles of the inherent safety concept;
- phased justification of reactor component endurance for future commercial lead-cooled reactors;
- electricity generation.

Figure LFR 2. **BREST-OD-300 reactor**



The baseline principle behind the inherent safety of LFR consists of the preferential use of the favorable inherent neutronic and physicochemical properties of the incorporated fuel, coolant and structural materials, as well as design solutions that allow full realization of these properties to exclude entire classes of severe accidents (uncontrolled power excursions and loss of heat removal). The BREST-OD-300 reactor power level has been selected with regard for the feasibility to use the associated design concepts as a reference for future larger output reactor facilities.

The reactor core design uses mixed uranium-plutonium nitride fuel; low-swelling ferritic-martensitic steel as fuel cladding; and fuel elements contained in shroud-less fuel assemblies. The selected dense nitride fuel, in combination with the lead coolant, makes possible to have complete breeding of fissile material in the reactor core with a constant low reactivity margin, thus preventing any rapid or large neutron-power excursion during reactor operation.

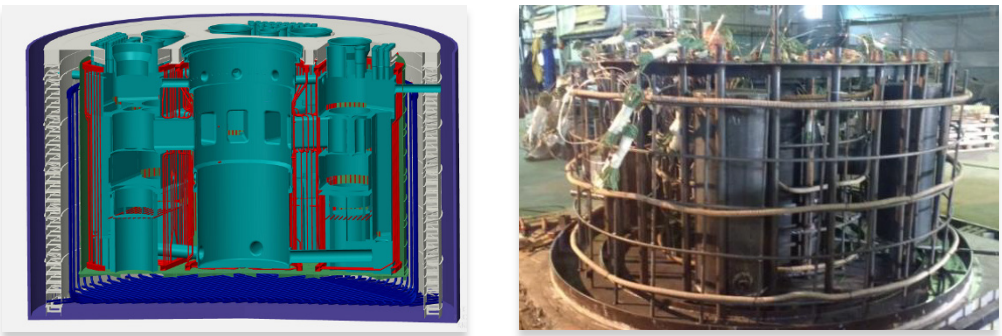
Until now, the dense nitride fuel technology has been implemented in pilot process lines. Technological processes are being improved and industrial production is being created for BREST-OD-300 (i.e. the fuel fabrication and re-fabrication module). To confirm serviceability of the fuel and the structural materials, fuel elements are being tested in BN-600 power reactor and in BOR-60 research reactor. Some of the fuel elements irradiated in BN-600 and BOR-60 have been completed and Post-Irradiation Examination (PIE) has been undertaken. They confirm in principle the fuel serviceability. The maximum burn-up achieved to date is -7.4% heavy atom (h.a.). Results required for fuel code verification have been obtained in these PIEs. The behavior of the fuel elements under irradiation meets, in principle, the pre-test analytical prediction. The obtained data demonstrates the feasibility of safe operation of the BREST fuel elements up to the parameters of the initial stage of operations (fuel burn-up of 6% h.a.).

Full-scale mock-ups have been manufactured for all types of fuel assemblies, reflector and shielding blocks. Hydraulic and vibration tests in water and liquid lead were performed. Data has been obtained which is required for updating reactor core calculations.

Neutronic calculations performed using a certified code have shown that the reactivity margin of the BREST-OD-300 reactor core life is in a range of 0.45-0.68 βeff. This reactivity margin can be ensured with regard for existing experience in fuel fabrication (fabrication accuracy is 1.2% δK/K), and the neutronic characteristic studies have been conducted, including with nitride fuel, at the BFS bench at IPPE (estimated error is 0.7% δK/K).

The specific design concepts used in the BREST-OD-300 reactor include an integral layout, absence of shutoff valves in the primary circuit, and use of passive and active-passive safety-related devices and systems. The integral layout, in a combination with the multilayer metal-concrete vessel, excludes accidents from loss of lead coolant. The justification of the metal-concrete vessel strength and serviceability (**Figure LFR 3**) is being performed based on data obtained by testing medium-sized metal-concrete structures (with typical dimensions of up to 7 m). Tests have been conducted to determine the properties of HT concrete grades under working temperatures and irradiation; the chemical inertness of the coolant with respect to concrete has been shown, and calculation procedures have been verified.

Figure LFR3. **Reactor vessel computational model and mock-up**



The BREST-OD-300 has a submerged-type once-through steam generator with a coiled heat exchanger. Silicon-containing austenite steel is used as the material of the heat-exchange tubes. Experiments were conducted to study the dependent failure of tubes caused by the rupture of one tube. The results of the experiments in lead coolant conditions simulating full-scale ones (temperature, pressure) have shown no dependent failure. Serviceability of the assembly for the

heat-exchange tube embedment into the tube sheet (see **Figure LFR 4**) was confirmed based on a thousand cycles of thermally loading a model (heat-up to 540°C, cooling down to 220°C). Studies of the weld and tube metal have not revealed intolerable defects.

Figure LFR 4. **Examination of the assembly for the heat-exchange tube embedment into the tube sheet**

The Main Circulation Pump (MCP) is vertical with an electric drive, axial type. The flow path has been optimized at water and lead test benches. The required head – flow rate characteristic has been obtained to ensure the pump operation in a range from 30 to 100 %. The full-scale MCP lower bearing has been designed and tested for endurance in liquid lead. No damage has been detected based on four intermediate withdrawals of the lower radial bearing's stator and rotor (30% of the design life achieved). A positive serviceability prediction has been formulated.

Radiation safety for the reactor facility conditions is based on data obtained as the result of out-of-pile and in-pile experiments using lead. Experiments have been completed and dependences have been determined to justify the release and transport of activation and fission products from the coolant at different temperatures (up to 680°C). The results of the radiation safety analysis have confirmed the implementation of target indicators, including no need for evacuation and resettlement of the public outside the site during anticipated operational occurrences with multiple failures (e.g. loss of power supply with scram failure, full reactivity insertion). The calculation results show that the FP released from the reactor for the first day is not more than $4.3 \cdot 10^{8}$ Bq (i.e. does not exceed the reference level for the allowable daily release during normal operation) in anticipated operational occurrences accompanied by multiple failures for a scenario with full reactivity insertion. The probability of core damage at the NPP with the BREST-OD-300 does not exceed $8.6 \cdot 10^{-9}$ 1/year, which makes it possible to ensure the acceptable level of safety for such type of nuclear power based reactor.

The reactor facility detailed design was developed subject to the fundamental requirements set forth in Russia's nuclear regulatory documents. The entire set of standards and regulatory documents, which take into account the peculiarities of lead-cooled reactors, are being developed in parallel with the detailed design and R&D performance. At the present time, the federal standards and rules have been updated based on the comments received and have been sent to Rostekhnadzor. Studies show that the BREST-OD-300 concepts can be used in large commercial reactor facilities while ensuring their competitiveness. The BREST-OD-300 unit design received a positive conclusion of the Glavgosexpertiza and currently is in the process of licensing with Rostekhnadzor.

Main activities in Japan

Fundamental experimental and theoretical studies for the LFR have been carried out by the Tokyo Institute of Technology. Experimental studies on chemical control and material compatibility of heavy liquid metal coolants (HLCMs) have been performed. Chemical compatibility of structural concrete materials with the HLMCs is important topic for the

development of LFRs, especially in the case of a coolant leakage accident. The chemical compatibility of various cement materials with liquid Pb and Pb-Bi was investigated by means of corrosion tests at 773 K. The coupon specimens made of Portland cement having different water/cement ratio were prepared and immersed into Pb and Pb-Bi at a static condition for 250 hours. After the tests, the chemical interaction between the cement specimens and the liquid metals was analysed. The results indicated that the chemical interaction between the HLMCs and the cement was limited (only small chemical interaction and mass transfer). These chemical behaviors were reasonable (cement materials are thermodynamically stable in the HLMCs at this temperature) and these results indicated the potential of the structural concrete as a coolant boundary.

In a theoretical study, innovative LFR concepts have been studied. The use of lead-alloy can provide for good neutron economy in fast reactors. The study on a new concept of a breed-and-burn reactor has been started utilizing the attractive features of lead-alloy. The new concept of this reactor is be based on a conventional reactor design. The reactor needs only natural uranium or depleted uranium for fuel once they come into an equilibrium condition. It is possible to achieve high burn-up of fuel without the movement of the burn-up region in the core in the equilibrium condition.

Main activities in Euratom

In June 2019, the European Commission (EC) co-organized the FISA 2019 and EURADWASTE '19 conferences in Pitesti (Romania) with the Ministry of Research and Innovation of Romania and the Institute for Nuclear Research (RATEN-ICN). The conference gathered some 500 stakeholders, presenting progress and key achievements of around 90 projects which are or have been carried out as part of the 7th and Horizon 2020 Euratom Research and Training Framework Programmes (FP). In that frame, a side workshop organized by the FALCON consortium on the ALFRED infrastructure attracted a significant number of participants stimulating the discussion of the state of R&D of heavy liquid metal technology and a road map for the LFR demonstrator in Europe.

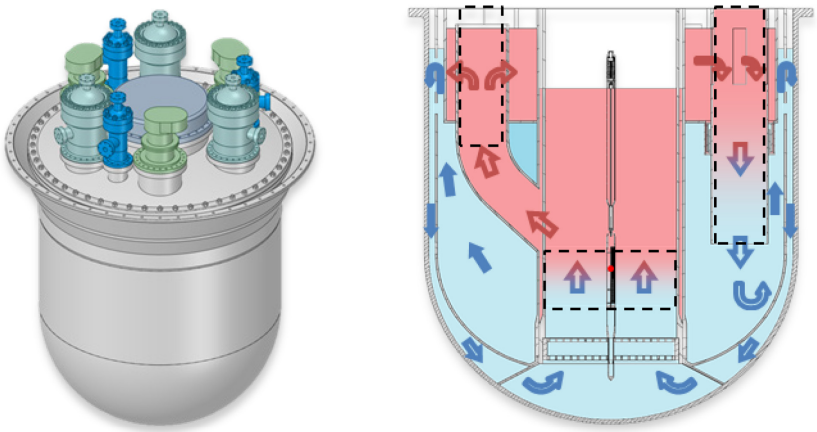
With regard to Euratom R&D projects, the already-running main projects related to LFR technology and Gen-IV fuels are GEMMA, M4F, INSPYRE and the LFR SMR INERI project (involving JRC and US DOE). A new project named PIACE has started related to the passive safety freezing prevention in LFRs. The project has had its kickoff meeting at the ENEA research lab in Brasimone and is presently under execution, expecting some experimental results to be delivered within 2020.

At the end of 2018 MYRRHA defined its road map for implementation of LBE technology for an ADS system. Belgium allocated EUR 558 M for the period 2019-2038 as follows:

- EUR 287 million for phase 1: building MINERVA (linear accelerator up to 100 MeV, 4 mA + proton target facility /PTF/) in the period 2019-2026;
- EUR 115 million for phase 2 and 3: phase 2 being the design and R&D of the second section of accelerator up to 600 MeV and phase 3 for further design and licensing activities related to LBE-cooled subcritical reactor, both to be carried out in the period 2019-2026;
- EUR 156 million for operation and experiments of the MINERVA for the period 2027-2038.

For the ALFRED project (LFR European demonstrator), the FALCON consortium made important steps during the period 2018-2019. First, a main step of the design review was completed, and a new system configuration was defined, consisting of three steam generators (SG) using single wall bayonet tubes, three dedicated dip coolers for the second decay heat removal (DHR) system, and three primary pumps (PP). Additional changes have been made in the primary system configuration by the definition of a hot and cold pool and a special arrangement of the primary flow path to completely eliminate the thermal stratification on the vessel (for forced and natural circulation conditions). The new configuration and its main characteristics are presented in the following **Figure LFR.5**. The DHR-1 is constituted by Isolation Condensers connected to steam generators (three units) and equipped with the anti-freezing system which is investigated in the PIACE project. A similar system is used for the DHR-2 system connected to a dip cooler using double-wall bayonet tubes.

Figure LFR 5. **ALFRED primary system flow-path configuration and externa view**



The FALCON consortium enlarged the community and extended the ALFRED project with the signature of several Memorandum of Agreements with partners willing to support in-kind the technical activities related to ALFRED development. The FALCON consortium also reached an important decision with regard to ALFRED operation and licensing: it was decided to approach both the operation and licensing using a step-wise approach to better face the known limits concerning materials in a representative environment. The idea is to follow a staged approach characterized by a constant primary mass flow and increasing power levels resulting in an increase of the maximum lead temperature:

- 1st stage (low temperature): Use of proven technology, proven materials, oxygen control, low T°, and Hot Fuel Assembly (FA) for in-core qualification of dedicated coatings for cladding;
- 2nd stage (medium temperature): Requires FA replacement, but uses the same SGs and PPs, and Hot FA for in-core qualification at higher temperature;
- 3rd stage (high temperature): Replacement of the main components for improved performance, representative of First-Of-A-Kind (FOAK) conditions for LFR deployment.

Consequently each stage is used to qualify the operation that will be carried out in the following stage. Each stage of the operation will need to be separately licensed but, using the confidence gained in the previous stage, the licensing process is expected to be a continuous process. The following table provides the main parameters of the envisaged staged approach:

Table LFR 2. **ALFRED staged approach main parameters**

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

During 2019, the Romanian government awarded RATEN-ICN (the Romanian research lab) funding of EUR 2.5 million in the frame of a project dedicated to "Preparatory activities for ALFRED infrastructure development in Romania". The project will last 15 months from September 2019 to November 2020. RATEN-ICN also responded to a call for proposals from the Romanian government with a project "ALFRED – step 1, experimental research support infrastructure: ATHENA (Lead pool-type facility) and ChemLab (Lead chemistry laboratory)".

A budget of EUR 133.9 million has been allocated in the 2019-2020 Euratom Fission Call for H2020 project proposals. Several project proposals were submitted in domains related to LFRs. The selected projects are expected to start by mid-2020. Finally, the SESAME Euratom collaborative project was finalized in 2019 with a workshop and the issue of a book dedicated to thermal-hydraulics aspects of liquid metals.

Main activities in Korea

In Korea, the LFR R&D has been redirected towards marine propulsion and space power development, by taking advantage of the excellent safety, very long refueling intervals and economic potential of LFR. Since the first Korean study begun in 1996 at Seoul National University (SNU), a new university consortium named Micro Nuclear Energy Research and Verification Arena (MINERVA) was formed to carried out a four-year conceptual design development of a non-refueling marine propulsion reactor cooled by LBE, in support of the Ministry of Science, Information and Technology (MSIT). The Ulsan National Institute of Science and Technology (UNIST) leads the MINERVA consortium with the participation of SNU, the Korea Advanced Institute of Science and Technology (KAIST), Kyunghee University, Ulsan University, KEPCO International Nuclear Graduate School (KINGS) and Moojin-Keeyeon Company. The Korean LFR Program has presently two main objectives:

- micro-modular reactors for marine propulsion, including ice breakers for opening the Northern Sea Route (NSR) that will cut CO_2 emission up to 40% for civilian vessels between Europe and Northeast Asian countries (including ROK). It is envisaged to expand it to container ships and bulk carriers, in support of the International Maritime Organization (IMO) resolution to ameliorate climate change.
- a technology development requirement for sustainable power generation using energy produced during nuclear waste transmutation has been reformulated towards increased safety.

To meet the first goal, a non-refueling micro-modular reactor called MicroUranus has been designed by MINERVA consortium based on URANUS as the reference. MicroUranus has innovative features including a compact core with the help of pony pumps and inherent natural circulation while keeping the reactor core life up to 40 years covering the entire life cycle of icebreakers and container ships without refueling. The power rating of MicroUranus is being optimized in the range between 15 MWe and 30 MWe. In order to assure the reliability of reactor systems overcoming aging phenomena including corrosion, Functionally Graded Composite (FGC) materials are envisioned to be used. As part of this material development, a group of researchers designed a FGC tube pilgering process using three-dimensional finite element analysis (FEA).

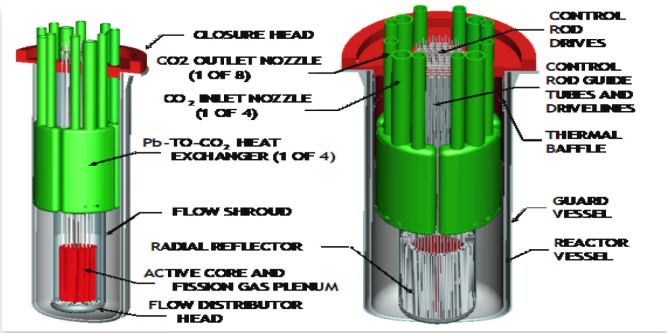
To meet the second goal, the Korean first LFR-based burner PEACER (Proliferation-resistant Environment-friendly Accident-tolerant Continual energy Economical Reactor) has been developed to transmute long-lived wastes into short-lived low-intermediate level wastes. In 2008, the Korean Ministry of Science and Technology (MOST) selected the SFR as the technology for long-lived waste transmutation. Recently, LFR R&D for transmutation in Korea has turned its direction towards an ADS-driven Th-based transmutation system designated as TORIA (Thorium-Optimized Radioisotope Incineration Arena) by a consortium led by SungKyunKwan University with the participation of Seoul National University and UNIST. For both objectives, large-scale test loops are employed for materials and thermo-hydraulic testing and model benchmarking. Korea's first large-scale LFR test facility, HELIOS, has been moved from SNU to UNIST where MINERVA is led. At SNU, a URANUS mock-up, designated as PILLAR (Pool-type Integral Leading test facility for lead-alloy-cooled small modular Reactor), has been designed, built and operated since 2018.

Main activities in United States

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

demonstrate continued interest in LFR technology. With regard to reactor design concepts, of particular relevance is the past development of the Small, Secure Transportable Autonomous Reactor (SSTAR) shown in **Figure LFR 6**, carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other organizations over an extended period of time. SSTAR is a small modular reactor (SMR) that can supply 20 MWe/45 MWt with a reactor system that is transportable and sealed. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life (15-30 years) with whole reactor or cassette refueling; and an innovative supercritical CO_2 Brayton cycle power conversion system. This concept represents one of the three reference designs of the GIF LFR pSSC. Even if this concept is no longer under development, it is still retained as a reference system by the pSSC to represent the small/very small size category for LFRs.

Figure LFR 6. **ALFRED Small, Secure Transportable Autonomous Reactor (SSTAR) view**



Past national laboratory efforts related to the LFR, in addition to the SSTAR reactor design efforts, include lead and lead-alloy performance and material compatibility studies activities at Los Alamos National Laboratory (LANL) with the Delta Loop. This facility has since been discontinued.

Current national laboratory activities include conceptual design, advanced material development and performance research, and instrumentation for monitoring steam generator status, principally conducted as industry-government partnerships under the USDOE GAIN (Gateway for Accelerated Innovation in Nuclear) program, at Oak Ridge National Laboratory (ORNL) and the Pacific Northwest national Laboratory (PNNL) in association with industry participants Westinghouse, Hydromine and Columbia Basin Consulting Group (CBCG). In the US industrial sector, current LFR reactor initiatives include the three companies mentioned above. Westinghouse Corporation maintains an ongoing initiative to design and commercialize a new advanced LFR system. Hydromine, Inc. is developing a new LFR reactor concept identified as LFR-AS-200 (Amphora-Shaped) in the 200 MWe size range as well as a family of smaller (microreactor) systems, and CBCG is developing a new conceptual design for a LBE reactor concept.

The Westingouse LFR aims at economic competitiveness, even in the most challenging global markets, through a simple and robust design, passive safety and life cycle requirements embedded in the design from the early design phase. It is a 950 MWt (~450 MWe) reactor, being developed starting with a lower-power prototype unit for technology demonstration. It utilizes a hybrid, micro-channel type heat exchangers to reduce vessel size/weight, and a thermal energy storage system to provide load following with minimum variations in-core thermal power. Additionally, it features a supercritical CO_2 power conversion system with air as the ultimate heat sink. The prototype unit will use oxide fuel and a pure lead coolant maintained at temperatures below 550°C. Advanced fuels and higher temperatures will be implemented after the prototype demonstration phase.

Hydromine's LFR-AS-200 concept is a compact 200 MWe LFR in which a high degree of compactness (<1 m^3 primary system volume/MWe output) is achieved by elimination of components and other design optimizations utilizing the favorable characteristics of pure lead

as a coolant. This compactness metric is estimated to be from 2 to 5 times lower than other metal-cooled fast reactors previously designed or in current design stages. In addition to this 200 MWe concept, Hydromine also envisions a family of very small (micro) reactors (5-20 MWe) known as the LFR-TR-X family with similar compactness and simplification of design. In these designs, control and shutdown rods are located outside the core, and the reactors are able to operate continuously for 15 years without refueling. The LFR-5 could be deployed in the near term, owing to its lower operating temperatures and use of qualified materials.

CBCG is taking an integrated approach to clean energy production by developing a nuclear plant design with load-following capabilities as an integrated grid-scale battery concept. Both the nuclear plant and the grid-battery are new designs by CBCG – when paired as an integrated facility, demand load-fluctuations are accommodated by the battery, while the nuclear plant remains at baseload operations. The nuclear plant uses LBE coolant with operation in a fast reactor spectrum. Initial efforts are focusing on licensing and regulatory requirements. As part of its ongoing research, CBCG is developing a Polonium mitigation system to reduce containment building requirements and off-site release potentials by eliminating the principal radiological release hazard associated with this technology.

Main activities in China

In February, 2019, INEST, the Chinese Academy of Sciences (CAS) was appointed as the Chinese representative for the Lead-cooled Fast Reactor (LFR) program in the Generation IV International Forum by the Ministry of Science and Technology of China. In this role, INEST co-ordinates the GIF LFR activity framework of domestic organizations inside China. On 18 October 2019, INEST signed the LFR memorandum of understanding (MoU) on behalf of China during.

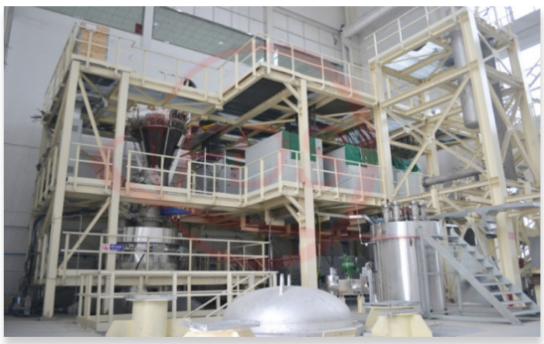
The Chinese government has provided continuous national support to develop lead-based reactor technology since 1986, by the CAS, the Minister of Science and Technology, the NSF. Following the last 30 years of research on lead-based reactors, the China LEAd-based Reactor (CLEAR) was selected as the reference reactor for both ADS and fast reactor systems, and the program is being carried out by the INEST/FDS Team, CAS. The activities on CLEAR are reactor design, reactor safety assessment, design and analysis software development, lead-bismuth experiment loop, key technologies and components R&D activities are being carried out.

Several "13th Five-Year" plans by the government related to lead-based reactor have been published. The CLEAR-M project aiming to construct a small modular energy supply system has been launched. The engineering design for the first prototype mini-reactor CLEAR-M10a with power of 10 MWth was carried out. To promote the engineering and commercial application of CLEAR-M, the China Industry Innovation Alliance of Lead-based Reactor (CIIALER) and the International Co-operative Alliance for Small LEad-based Fast Reactors (CASLER), both led by INEST, were established and supported by over 100 companies, and a related industrial park began to be built.

For an ADS system, several concepts and related technologies are under assessment. For example, the detailed conceptual design of CLEAR-I with the final goal of MA transmutation having an operational capability of subcritical and critical dual-mode operation has been finished. An innovative ADS concept system as an advanced external neutron source driven traveling-wave reactor for energy production, CLEAR-A, was proposed. The CiADS project aiming at building a 10 MWth subcritical experimental LBE-cooled reactor coupled with accelerator was approved, and preliminary engineering design is underway. The project was conducted by the collaboration of CAS and other industrial organizations.

In order to support the China LEAd-based Reactor projects as well as validate and test the key components and integrated operating technology of lead-based reactors, three integrated test facilities have been built and commissioned since 2017, including the lead-based engineering validation reactor CLEAR-S (see **Figure 12 LFR.7**), the lead-based zero power critical/subcritical reactor CLEAR-0 coupled with HINEG neutron generator for reactor nuclear design validation, and the lead-based virtual reactor CLEAR-V. In 2019, a loss-of-flow benchmarking test based on the pool-type CLEAR-S facility was prepared, and is planned to be conducted in 2020.

Figure LFR 7. **Lead-based Engineering Validation Reactor CLEAR-S**



In recent years, several other organizations started paying greater attention to LFR development. China General Nuclear Power Group (CGN) is carrying out CLFR reactor conceptual design and related research. China National Nuclear Corporation (CNNC) is developing LFR technologies such as core neutronics characteristics testing. The State Power Investment Corporation (SPIC) is focusing on the 100 MWe BLESS reactor conceptual design. Several universities, such as Xi'an Jiaotong University (XJUT), the University of Sciences and Technology of China (USTC), are carrying out fundamental LFR technologies R&D, including materials testing, thermal-hydraulic analysis, safety analysis, etc., to support LFR development in China.

In December 2019, the domestic co-ordination meeting of GIF LFR was held in INEST. Representatives from more than ten Chinese organizations who were involved in LFR R&D attended this meeting. The domestic LFR joint working group was proposed and INEST was suggested as the lead of the working group to co-ordinate the participation and co-operation of related organizations and activities in China.

Alessandro Alemberti

Chair of the LFR SSC and all Contributors

Molten Salt Reactor (MSR)

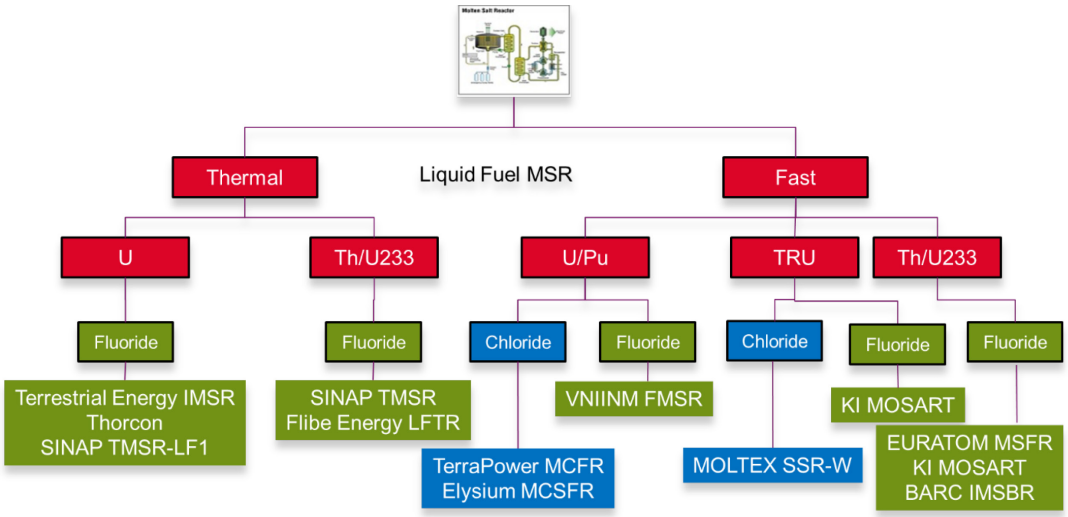
Introduction

Molten Salt Reactor (MSR) concepts have been studied since the early 1950s, but with only one test reactor operated at ORNL in the 1960s. For about 15 years, there is now a renewal in the interest of such a reactor technology, in particular for its acknowledged inherent reactor safety and its flexibility.

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

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

Figure MSR 1. **The most studied MSR concepts, with the key players (RTOs or Vendors)**



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

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

R&D objectives

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

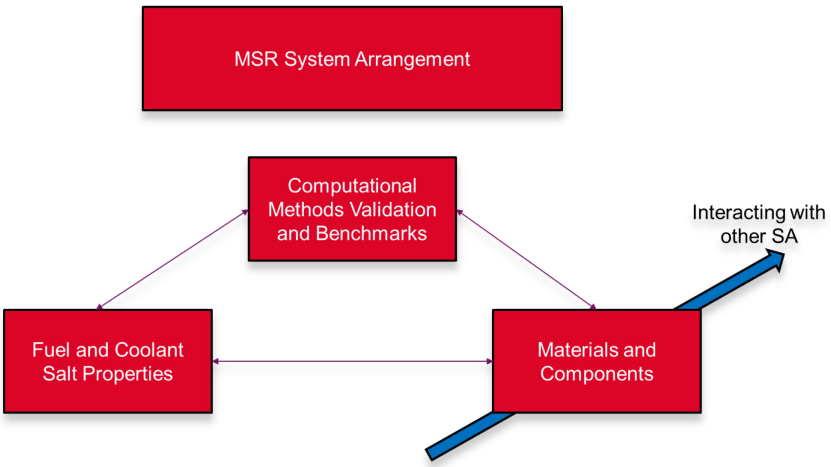
- studying the salt chemical and thermodynamic properties;
- system design: Development of advanced neutronic and thermal-hydraulic coupling models;
- studying materials compatibility with molten salt;
- Salt Redox control technologies to master corosion of reactor vessel;
- development of efficient techniques of gaseous fission products extraction from the coolant by He bubbling;
- salt reprocessing: reductive extraction tests (actinide-lanthanide separation);
- development of a safety approach dedicated to liquid fueled reactors.

Main activities and outcomes

MSR pSSC activity

In 2019, the key activity was the preparation of the System Arrangements with the definition of three potential Projects Arrangements, which would allow the community to contribute widely. Therefore, these PAs are quite transversal and not concept dependent but can support the development of any concept (see **Figure MSR 2**). They address the salt behavior, the materials properties and the system integration. The SA should enter into force in 2021.

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



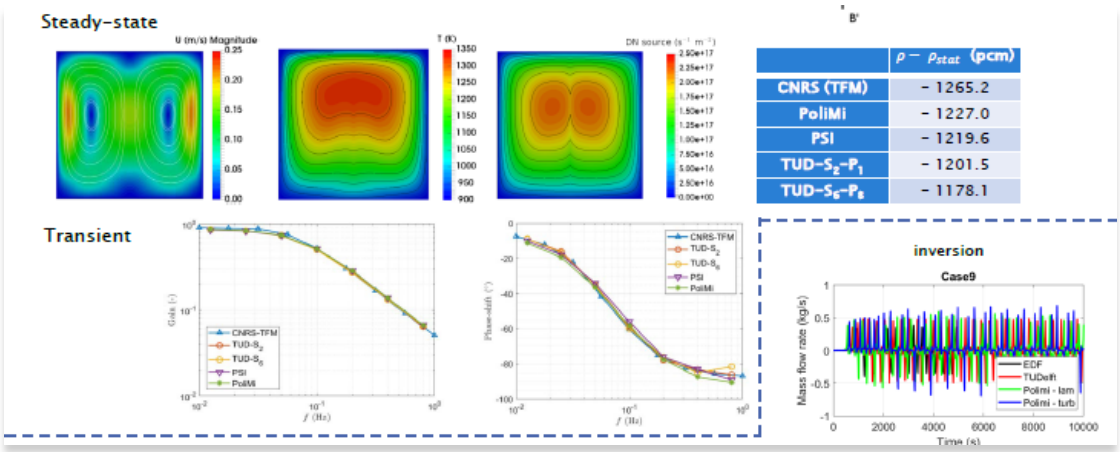
Euratom

European Project SAMOFAR/SAMOSAFER: In the European Union the SAMOFAR project has run successfully under co-ordination of TU Delft for a period of four years from 2015 to 2019 and has been closed by a festive meeting in the theatre of Delft, The Netherlands. This meeting consisted of a students' boot camp at TU Delft organized by the US project NuStem and the EU project SAMOFAR, followed by a two-day meeting presenting the final results of SAMOFAR and with

presentations of representatives from Canada, China, EU, Russia and the United States on the progress made in their respective national research programmes. The boot camp was attended by about 40 students worldwide, while the final meeting was attended by about 100 participants.

At 1 October 2019, the new SAMOSAFER project started aiming at new simulation models and tools, and on new safety barriers for the Molten Salt Reactor. The goal of this new project is to develop and demonstrate new safety barriers for more controlled behavior of Molten Salt Reactors in severe accidents, based on new simulation models and tools validated with experiments. The grand objective is to ensure that the MSR can comply with all expected regulations in 30 years' time. After successful completion of this project, the simulation models and tools can be used by the nuclear industry, and the innovative safety barriers can be implemented in new MSR designs. This will lead to increased safety margins in future Gen-IV Molten Salt Reactors to ensure they will comply with the latest and future safety standards. SAMOSAFER is co-ordinated by TU Delft and will run until 2023.

Figure MSR 3. **Multi-physics code for V&V. Benchmark of PoliMi and PSI codes**

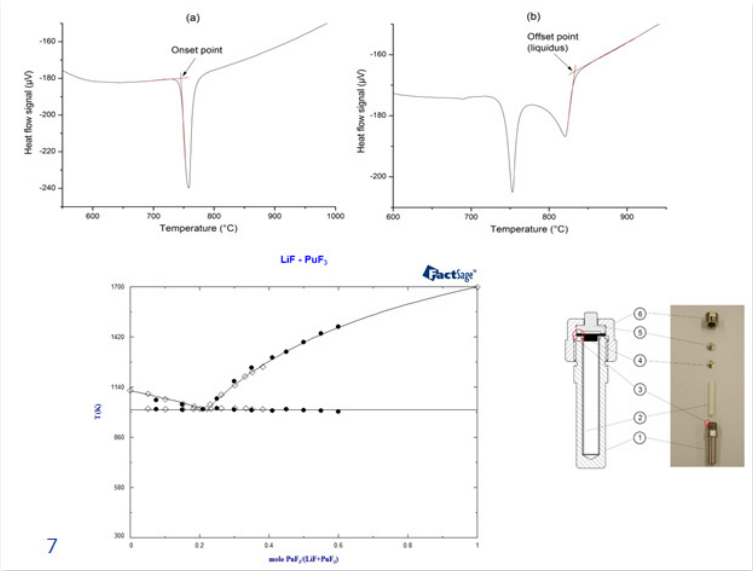


A special session on the progress in the Science & Technology of Nuclear Reactors using Molten Salts was organized in collaboration with the European Physical Journal, Nuclear Sciences & Technologies. Guest editors were Jan Leen Kloosterman (TU Delft), Elsa Merle (CNRS) and Jean Ragusa (Texas A&M).

In the Netherlands, the Salient-01 irradiations in the Petten High Flux Reactor were finalized. The samples are currently being investigated in the framework of SAMOSAFER in laboratories at NRG and JRC Karlsruhe. Follow-up irradiations are underway.

JRC Karlsruhe: After successfully establishing the method to synthesise highly pure actinide fluoride salts using HF fluorination line at JRC Karlsruhe, focus was put on development of a chlorination technique to synthesise actinide chloride salts with same high purity. The first tests were done on innovative synthesis of uranium chloride salts from uranium oxide, using a mixture of Cl_2 and CCl_4 gases (carbo-chlorination to convert UO_2 to UCl_4) with successive reduction by H_2 (to convert UCl_4 to UCl_3). By the end of 2019, the first step of the conversion was successful and small quantities of highly pure UCl_4 were obtained.

Figure MSR 4. **Phase diagram study pf the LiF-PuF3 system**



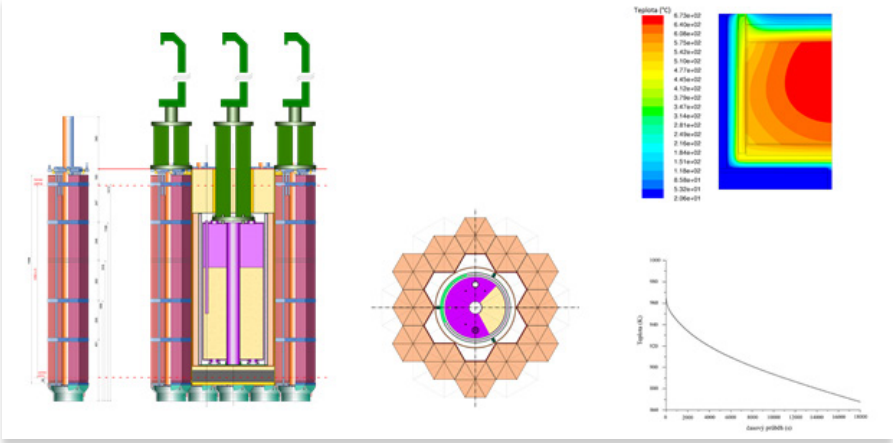
Work on high temperature properties investigation of fluoride salt systems has continued at JRC Karlsruhe with the following highlights:

- First measurements of the melting point of pure PuF_3 have been done. This study still needs optimization, but the experimental set up to determine melting points of actinide fluorides that melt above 1 250°C, which was a limit using standard encapsulation technique, has been established.
- A method to measure thermal conductivity of solid salts, simulating precipitates on the reactor vessel surface, has been improved, achieving successful and reliable measurements using laser flash technique. Thermal conductivity of the series of alkali fluorides has been measured, as well as solidified complex fuel mixtures containing UF_4 and ThF_4 components.
- First trial to test the method to determine solubility of gases in molten salts have been made utilizing high temperature infusion facility with successive quantitative determination of the dissolved gases using effusion cell coupled with quantitative mass spectrometry.

Among experimental studies on basic thermo-chemical properties, JRC is significantly involved in synthesis and fabrication of the fluoride fuel salt for the planned irradiation experiment SALIENT-03 in the HFR (High Flux Reactor) in Petten in a major collaboration with NRG.

Centre Research ŘEŽ: In 2019, research and development of MSR technology continued in the Czech Republic also as a part of a national MSR project supported by the Ministry of Industry and Trade and by the Technology Agency of the Czech Republic. The main effort was devoted to the preparation of neutronic measurements with the so-called "Hot Inserted FLIBE Zone" in the experimental reactor LR-0 of the Research Centre Řež. This is a new demanding experimental program whose aim is to determine the reactivity temperature feedback coefficients of the FLIBE melt in the working temperature range of MSR system (500-750°C). By the end of 2019, all the main components of the Hot Inserted FLIBE Zone had been produced. The active experimental program will start in 2020.

Figure MSR 5. **Design of the heated zone for FLIBE salt**



In addition, the development and testing of impeller pumps for fluoride melt media, in the program of MONICR alloy welding technology optimization and in the long-term MONICR alloy corrosion test program. Research work also continued in the field of the development of electrochemical separation methods from fluoride melt media and a new activity focused on experimental studies and verification of MSR volatile fuel components extraction by Fused Salt Volatilization method was launched in second half of 2019.

France

In addition to the work carried out in SAMOFAR and SAMOSAFER, French partners (CNRS, CEA, FRAMATOME…) worked at building a MSR community around a common roadmap including neutronic and its modelling, fuel salt selection, materials and corrosion to develop a concept of fast reactor using a U/Pu fuel cycle in molten fluoride. In particular, a new project was created at CEA to co-ordinate their activities in the field.

Australia

The widespread deployment of molten-salt-based energy systems, including Molten Salt Reactors requires the development and qualification of materials and components capable of withstanding their challenging operation conditions. Hence, with a view to shortening the time to deployment of MSR low-emission energy generation systems, Australia, is working on the development, manufacturing and testing of suitable structural materials and coatings. In particular, ANSTO continues to collaborate with GIF partners to study and understand the corrosion in FLiNaK of candidate Stainless Steels and Nickel-based alloys, in particular, using ANSTO's large-scale infrastructure, (the OPAL reactor, the Australian Synchrotron, and the Centre for Accelerator Science).

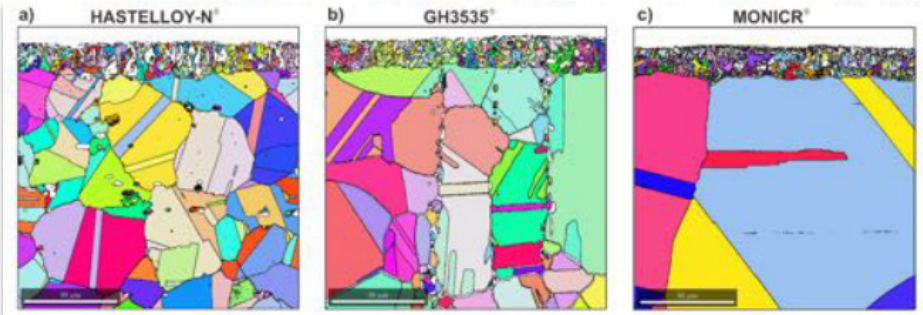
During 2019 Australia continued its initiatives to increase cross-cutting materials related research within GIF. In addition to chairing the GIF Advanced Manufacturing and Materials Engineering Task Force (AMMETF) a joint VHTR/MSR Materials and Components R&D Workshop was held between the MSR pSSC and the VHTR PMB meetings in Avignon in September 2019 designed to identify areas of common research interest.

The workshop was very successful and identified the following potential areas of collaborative R&D:

- advanced manufacturing;
- comparison of ion/neutron radiation damage design methodologies;
- development of creep, and creep-fatigue models;

- high-temperature performance, and radiation damage of graphite, C/C, SiC/SiC;
- small sample testing, and advanced surveillance methodologies;
- welding and joining.

Figure MSR 6. **FLINAK molten salt corrosion for GH3535, Hastelloy N and MoNiCr alloys**



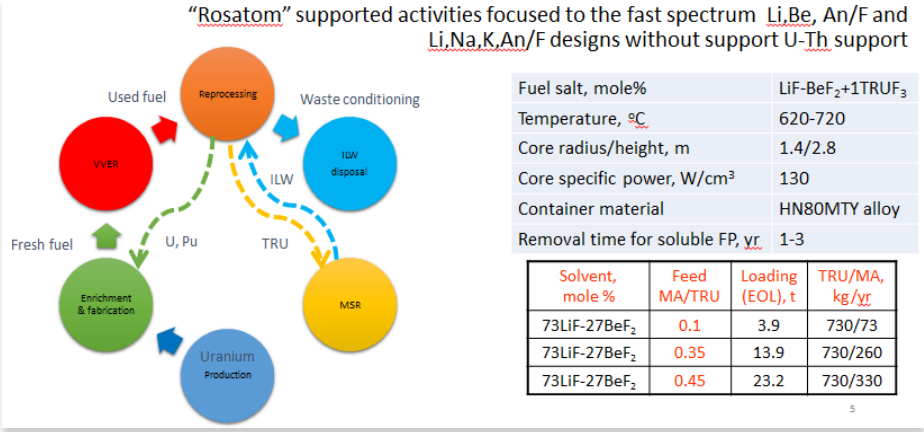
- EBSD orientation maps of HASTELLOY-N®, GH3535® and MONICR® alloys after the exposure to the FLiNaK molten salt at 650°C for 500h. The top edge represents the surface exposed to the molten salt.

Russia

A significant effort on the molten salt reactor development was performed in the RF in 2019. The State Corporation "Rosatom" has continued to provide support through a number of R&D programs to the single fluid MOlten Salt Actinide Recycler & Transmuter (MOSART) system, where minor actinides and plutonium from spent VVER-1000/1200 fuel are dissolved in the liquid fluoride salt. Last developments concerning MOSART design addressed advanced large power Li,Be,An/F and Li,Na,K,An/F units with the main objective to close the nuclear fuel cycle for all actinides, including Np, Pu, Am and Cm. The work was also continued on development of the test 10 MWt Li,Be,An/F MOSART design coupled with fuel salt clean-up unit at the site of the Mining and Chemical Combine (Zheleznogorsk) in order to demonstrate the control of the reactor and fuel salt management with different TRU loadings for start-up, transition to equilibrium, drain-out, shut down, etc. The range of MOSART technology developments currently underway includes:

- high neutron fluence and salt tolerant alloy design property evaluation for fuel Li,Be,An/F and Li,Na,K,An/F salt mixtures;
- both high and low fidelity computational modelling and tool development;
- physical and chemical property measurement for fuel Li,Be,An/F and Li,Na,K,An/F salts;
- molten salt pump & heat exchanger designs and its demonstration;
- instrumentation development;
- highly automated remote operations and maintenance technology development and demonstration;
- fuel salt clean-up demonstration and both solid and gaseous waste stream assessment.

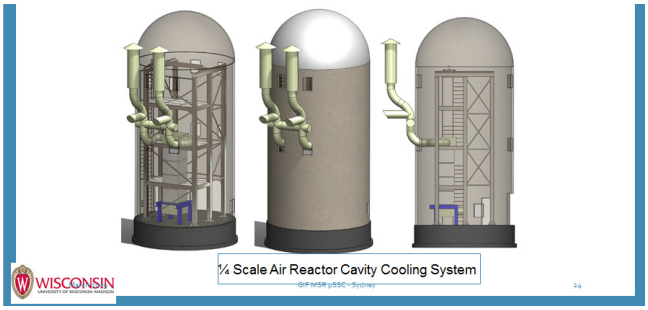
Figure MSR 7. **The MOSART Concept as an integral element helping to solve the problem of long-lived actinides**



United States

A broad range of both molten salt fueled and cooled reactor activities were performed in the United States in 2019. Notably, Kairos Power Inc. continued to file topical reports with the US Nuclear Regulatory Commission (NRC) in preparation for a license application. Also, the Canadian Nuclear Safety Commission and the NRC announced that they have selected Terrestrial Energy's Integral Molten Salt Reactor for the first joint technical review of an advanced, non-light water reactor under a memorandum of co-operation on advanced reactor and SMR reviews. The US government continues to provide support to the emerging US MSR industry through a number of cost-shared R&D programs. The NRC continues the process of modernizing its licensing requirements to better reflect the safety characteristics of advanced reactors. The US Department of Energy (DOE) continues to support both university and national laboratory activities at limited scale to overcome the remaining technical hurdles to MSR deployment. The US government has also continued work to develop MSR models and associated tools for safeguards analysis. The range of MSR technology developments currently underway includes high neutron fluence and salt tolerant alloy design and property evaluation, both high and low fidelity computational modelling and tool development, graphite testing, fuel salt thermo-physical and thermo-chemical property measurement, fuel salt thermodynamic database development, molten salt pump design and demonstration, instrumentation development, highly automated remote operations and maintenance technology development and demonstration, and both solid and gaseous waste stream assessment. Oak Ridge National Laboratory hosted the annual DOE-Gateway for Innovation in Nuclear supported MSR workshop which featured ~250 participants from industry, the national laboratories, government agencies, international organizations, and academia.

Figure MSR 8. **Air cooled RCCS designed and constructed at University of Wisconsin**



Canada

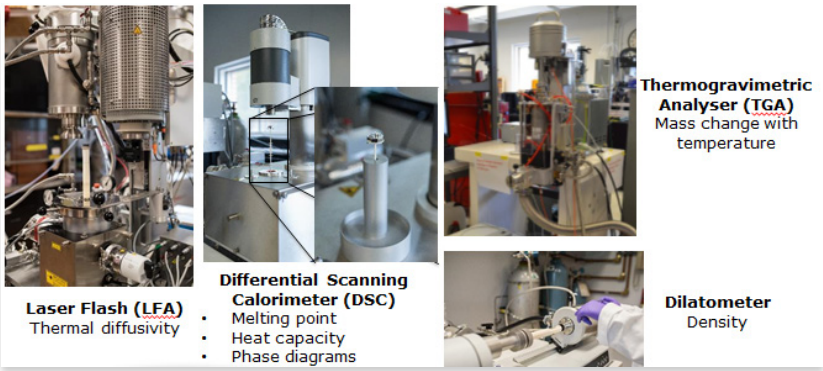
In 2019, Canadian Nuclear Laboratories (CNL) continued to develop expertise and capabilities in support of SMRs and launched a new cost-shared R&D program called the Canadian Nuclear Research Initiative (CNRI). The CNRI program was established by CNL to accelerate the deployment of SMRs in Canada by enabling research and development, and connecting the SMR industry with the facilities and expertise within Canada's national nuclear laboratories. Among the many benefits of the program, participants are able to optimize resources, share technical knowledge, and gain access to CNL's expertise to help advance the commercialization of SMR technologies. In the first intake to this new program three molten salt reactor vendors have submitted proposals with a varied program of work including electrochemical separation methods, tritium management, reactor physics, thermal-hydraulics and safeguards studies.

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

- development of actinide molten salt fuel synthesis;
- fission product retention in molten salt experiments; evaluation of passive cooling during a station blackout with experiments on coupled natural circulation heat transfer between water and molten salt loops and evaluation of molten salt plug melting in accident conditions;
- corrosion loop development for measuring corrosion of structural materials;
- modelling and simulation of Molten Salt Reactor Designs including evaluation of codes for advanced reactor coupled transient simulation toolset against ORNL MSRE: Physics (SERPENT, Rattlesnake); TH (RELAP5-3D, ARIANT); CFD (STAR-CCM+) and atomistic simulations to predict molten salt properties.

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

Figure MSR 9. **Sample encapsulation and measurement technique development at CNL**



Switzerland

The Swiss MSR research is co-ordinated by Paul Scherrer Institute (PSI). In 2019 the PSI continued to develop expertise and capabilities in the three selected areas: fuel cycle, system behavior, and thermo-dynamics of molten salts. The major aim of these simulation activities is the assessment of MSR safety and sustainability. Since PSI is a member of the SAMOFAR and SAMOSAFER projects, part of the PSI activities contribute to the EU progress report.

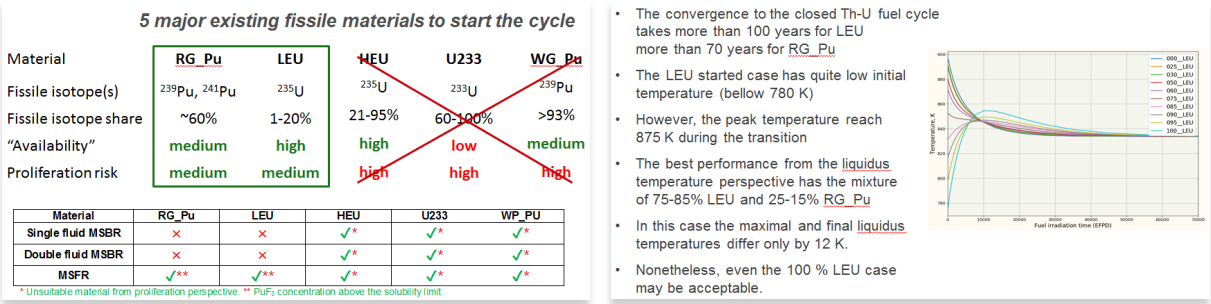
In the area of fuel cycle assessment, several past studies were published in 2019. The result of 16 different reactors comparison in equilibrium fuel cycle, inclusive 4 MSR types, were published. The performance in both U-Pu and Th-U cycles was compared. The breed-and-burn

fuel cycle study from B. Hombourger PhD thesis was published, taking the advantage of special MSR edition of the EPJ Nuclear Sciences & Technologies journal. The breed-and-burn fuel cycle study was further extended by parametric study within MSc thesis and the results will be presented at Physor 2020 conference.

The system behavior study with Open-FOAM based solver continued in 2019 for MSFR core as a H2020 project SAMOFAR contribution. The activities were still ongoing, after the project was finished, and the respective freezing model developed within the project still requires validation. The Open-FOAM based solver was also applied for conceptual designing of own breed-and-burn core thermo-hydraulics layout. The results will be also presented at Physor 2020 conference.

The thermo-dynamics simulation of molten salts was continuing with GEMS TM code, focusing on fluoride-based salts. The liquidus temperature and its evolution during transition from initial to equilibrium fuel cycle was simulated and the results published at ICAPP 2019 Conference. The thermo-dynamics code was also coupled to severe accident code MELCORE to improve the simulation of compounds evaporation from the fuel salt in severe accidents. Towards the end of the year the preparation of the GEMS database for chloride salts started.

Figure MSR 10. **Transition from initial to equilibrium cycle**



China

In 2019, Shanghai Institute of Applied Physics, Chinese Academy of Sciences (SINAP-CAS) is steadily promoting the related work of TMSR.

The design of the 2 MWth molten salt test reactor (TMSR-LF1) was completed and the Preliminary Safety Analysis Report (PSAR) submitted by SINAP to the National Nuclear Safety Administration has passed the examination and approval. In addition, the preparation of key components has been basically completed, and the construction of TMSR-LF1 has begun.

Figure MSR 11. **Progress of the TMSF-LF1**

The construction of the scaled experimental device (TMSR-SF0) has been completed, and the key commissioning activities such as primary and secondary circuit molten salt loading and main pump operation have been successfully completed. The experimental program will be launched in 2020.

Conceptual design of the flowsheets for TMSR fuels is ongoing with validation of some key techniques being finished. Fundamental studies on the chemistry of actinides and fission products in molten salt were started. The PIE experiments of several different kinds of nuclear graphite have been completed, and the evaluation method of irradiation life of nuclear graphite by ion beam irradiation was established. Additionally, the alloys used for high-temperature (750-850°C) MSR are being developed.

Stéphane Bourg

Chair of the MSR SSC and all Contributors

Super Critical Water Reactor (SCWR)

Preamble: *For a sake of homogeneity among all the system reports within this Annual Report, this chapter has been intentionally synthesized in a reduced number of pages. The full extended version of the 2019 SCWR system report with the complete list of publications can be uploaded on the GIF website.*

Main characteristics of the system

The SuperCritical Water-cooled Reactor (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, etc.). Therefore, the R&D needs for each reactor type are common; this enables collaborative research to be pursued.

The main advantage of the SCWR is improved economics because of the high thermodynamic efficiency and the potential for plant simplification. Improvements in the areas of safety, sustainability, and PR&PP are also possible and are being pursued by considering several design options using thermal and fast spectra, including the use of advanced fuel cycles.

There are currently three Project Management Boards (PMBs) within the SCWR System: System Integration and Assessment (provisional), Materials and Chemistry, and Thermal-hydraulics and Safety. Canada, China and Euratom signed the extension of the Project Arrangements for Thermal-Hydraulics and Safety as well as the Materials and Chemistry in 2017.

R&D objectives

The following critical-path R&D projects have been identified in the SCWR System Research Plan:

- System integration and assessment: Definition of a reference design, based on the pressure tube and pressure vessel concepts, that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance. An important collaborative R&D project is to design and construct an in-reactor fuel test loop to qualify the reference fuel design. As a SCWR has never been operated before, such generic testing is considered to be mandatory before a prototype reactor can be licensed.
- Thermal-hydraulics and safety: Gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed for validating 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 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.

Main activities and outcomes

System integration and assessment

Four SCWR core concepts with thermal spectrum have been proposed. Canada, EU and Japan have completed their concept development. China is continuing the development of core and plant concepts for their pressure vessel type thermal spectrum SCWR. The China Pressure vessel-type SCWR (named CSR1000) has the following characteristics: thermal neutron spectrum, light water as moderator, two flow-pass of coolant in core, direct once-through cycle. The reference CSR1000 has the 9X9 pin by pin fuel assemblies with center 5X5 pin taken by the water moderator box. Recently, the fuel assembly and core structure design are simplified. In the new design, the UO_2 fuel rods are set around the tube to get moderated homogeneously and sufficiently. MOX fuel rods are settled in the outer zone to match spectrum. No Water rod or solid moderator are needed. **Figure SCWR 1** presents the reference and new FA design. Aiming at the reactivity control requirements of the SCWR core and its strong nuclear thermal coupling characteristics, a new type of control rod loading design was invented to overcome the shortcomings of the traditional "checkerboard" control rod loading design method. The new loading method reduces the number of control rod drive mechanism arrangements, reducing the difficulty in designing the SCWR pressure vessel top cover, and simplifying the control rod operation management procedure. Two project proposals have been approved by the China Ministry of Science and Technology in 2019 to promote the China SCWR design. The two projects start from 2020 and end in 2022. The international review of China SCWR design is supposed to be completed during this period.

Figure SCWR 1. **SCWR Thermal Spectrum Core Concepts**

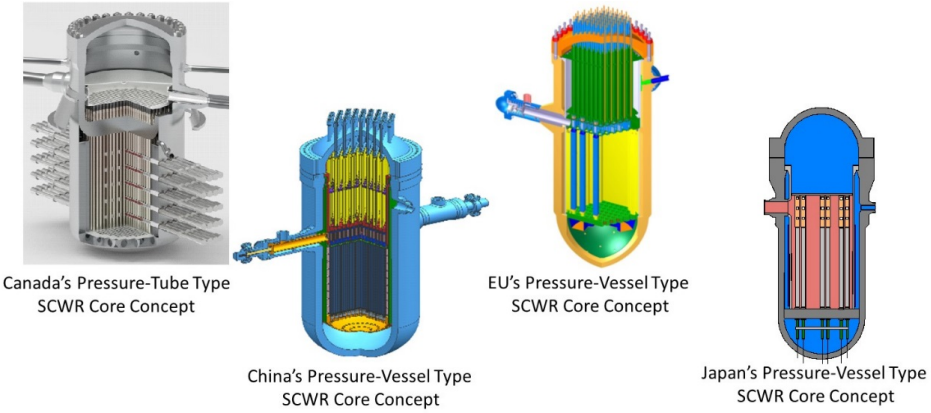
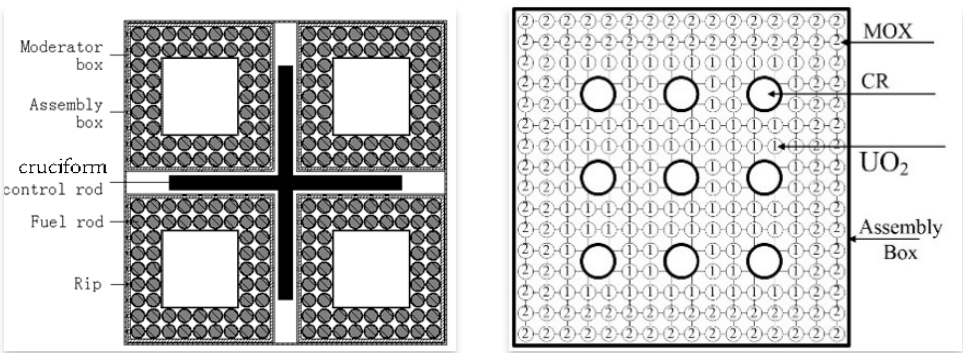


Figure SCWR 2. **China SCWR fuel assembly design improvement**

(a) Reference design (b) Improved design



Canada has developed a preliminary small pressure-tube type SCWR concept with similar core configuration. Currently the 43-Element bundle is the preferred candidate for the fuel bundle housing 170 fuel channels. The operating pressure 25 MPa with outlet temperatures about 450°C. Work on finalizing this concept is ongoing.

One of the main general activities covering almost all fields of the SCWR development was the preparation of the European-Canadian-Chinese Small Modular SC-Water Reactor Technology (ECC-SMART) proposal. The proposal joined the significant institutions working in the field of SCWR development from Europe, China, Canada and Ukraine to create very strong multi-international consortium. The proposal covers the major knowledge gaps in the thermal-hydraulics and safety and materials and chemistry issues for SCWR technology as well as the specific SMR topics related mostly to the scaling of the technology and legislation aspects.

The 9th International Symposium on SCWRs was held in Vancouver, Canada, March, 2019, hosted by Canadian Nuclear Society. The conference was supported by Natural Resources Canada, Atomic Energy of Canada Limited and Canadian Nuclear Laboratories. About 60 participants from Canada, China, EU and Japan attended this conference and 58 presentations covering concept development and technology areas were presented.

Thermal-hydraulics and safety

The TH&S PMB activities include flow and heat transfer experiments and correlations development, critical flow and flow instability investigation, numerical investigation and code development.

A project was established at the Canadian Nuclear Laboratories (CNL) to outline a framework of current prediction methods, the parameters associated with these methods, and the fluids they are applied to. The goal of this project is to complete a summary of literature reviews of heat transfer prediction methods in SuperCritical fluids (SC-fluids). From the literature review, the heat transfer to SC-fluids is dependent on at least six system parameters and four fluid flow parameters. The six system parameters include geometry, flow orientation, fluid flow direction, fluid type, heat flux direction, and power profile. The commonly used geometries and fluid types reflect the class of research and industrial applications where SC-heat transfer occurs. A more in-depth analysis of these parameters reveals that the majority of applications in which SC-heat transfer occurs is restricted to a narrow range of fluid parameters. These four fluid flow parameters are defined as: 1) Fluid pressure; 2) Fluid mass flux; 3) Surface heat flux; and 4) Fluid bulk enthalpy or bulk temperature. Six different SC-heat transfer prediction methods are currently used: 1) Correlations; 2) Semi-empirical models; 3) Look Up Tables; 4) Look Up Lists; 5) Neural Networks; and 6) Numerical/Computational Fluid Dynamics.

CNL conducted a study to investigate the applicability of a break discharge model that was specifically developed for supercritical conditions. To achieve this goal, the model was introduced in the Canadian thermal-hydraulics system code CATHENA. This model is hereafter referred to as the Modified Homogeneous Equilibrium Model (M-HEM). A comparison between the previously used homogeneous Equilibrium Model (HEM) and the Modified Homogeneous Equilibrium Model (M-HEM) model was performed. The assessment of the discharge models was performed by using experimental data in a simple geometry configuration (shows two representative results). The results of the assessment are used as a base to update the LOCA simulations used for the Canadian SCWR conceptualization.

The fuel channels of the Canadian SCWR undergo large density variation along the reactor core as condition of the coolant flow changes across the pseudo-critical point. To verify the stability of the design, CNL created a task aimed to verify, assess and develop a stability map for the Canadian SCWR design. This task was divided in two steps: i) assessment of the tools and ii) development of the Canadian SCWR stability map. Currently, CNL is focusing on pure thermal hydraulics instabilities and assessment of modelling tools. The tool selected was the system code CATHENA. Two datasets were selected to verify the applicability of the code: 1) the two parallel channel instability experiments carried out by NPIC (Nuclear Power Institute of China), and 2) the natural circulation numerical experiments conducted at the University of Manitoba. The simulation results showed that CATHENA is able to predict the flow oscillations, nonetheless the magnitude differs from the experimental data. However, given that the model

was simplified significantly, as recommended by the experimentalist, and the flow instability is highly dependent on geometry, this could have an impact on the simulation results. **Figure** SCWR4 shows two representative CATHENA prediction cases.

Figure SCWR 3. **CATHENA Predicted Coolant Mass Fluxes for 1-mm Orifice Diameter (Left) and 1.395-mm Orifice Diameter (Right) Supercritical Discharge Experiment at École Polytechnique de Montréal**

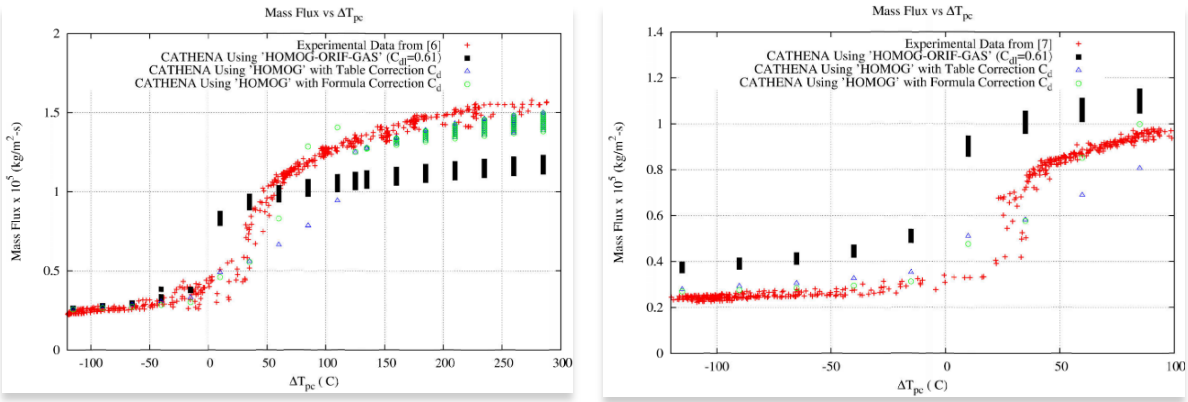
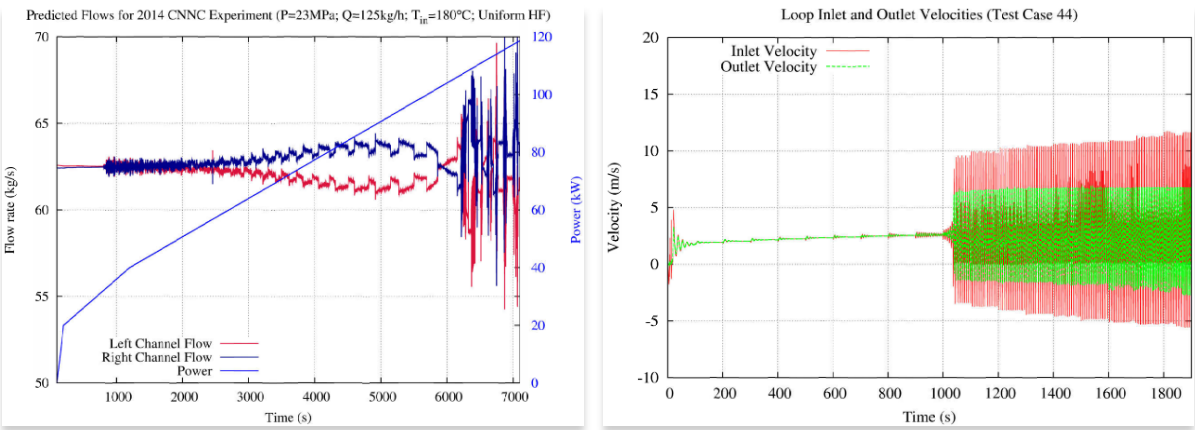


Figure SCWR4. **CATHENA Mass Flow Rate Predictions of Two Parallel Channel Instability (Left). CATHENA Predicted Velocities and Pressure Drops in a Natural Circulation Loop (Right)**



Heat transfer of in-tube SC-fluid cooling accompanying out-tube pool boiling was investigated in Xi'an Jiaotong University (XJTU). A smooth horizontal circular tube with an inside-diameter of 20 mm was submerged in a water pool at atmospheric pressure. Test parameters of in-tube were as follows: Pressure: 23-28 MPa, mass flux: 600-1 000 kg/m^{2}s, fluid temperature: 400-725 K, and the temperature difference between bulk and wall: 300–374 K. A thermal amplification system based on out-tube pool boiling was used to improve the measurement accuracy of local heat duty near pseudo-critical region. According to the experiment, the transition from nucleate boiling to film boiling in the pool occurred near the pseudo-critical fluid region. Sharp variation on thermo-physical properties led to the peak value of heat transfer coefficient in the pseudo-critical region. The pool boiling heat flux increased gradually to 1.19 MW/m^2 near the pseudo-critical point. Based on the experimental data, a modified Gnielinski equation was adopted to predict the heat transfer coefficient of in-tube SC-fluid cooling without-tube pool boiling.

Xi'an Jjiaotong University also performed experiments of heat transfer to supercritical Freon R134a flowing upward and downward in a circular tube with inner diameter of 10 mm with heat fluxes of 20-65 kW/m^2, mass fluxes of 400-1 000 kg/m^{2}s, bulk fluid temperature of 80-115°C at pressure condition of 4.2 MPa. The influences of heat flux, mass flux, flow direction, buoyancy force and flow acceleration on supercritical R134a heat transfer were discussed respectively. The influence of buoyancy force and flow acceleration on heat transfer were investigated and the non-dimensional parameters were obtained. New heat transfer correlations for upward and downward flow were proposed respectively.

Nuclear Power Institute of China (NPIC) performed Natural Circulation (NC) experiments and numerical analysis with water and carbon dioxide. For the supercritcal water NC instability, the preliminary analysis work has been done with the system analysis code. The code could predict the instability behavior of natural circulation. But some discrepancies exist which need further improvement. For the SC-CO_2 NC instability, based on the theoretical analysis of flow and heat transfer of SC-CO_2, a new explanation of the mechanism of flow oscillation in SC-CO_2 natural circulation has been put forward. The reliability of the new mechanism has been verified by experimental results.

China Institute of Atomic Energy (CIAE) performed the investigation of critical flow model for supercritical pressure condition. The model is derived to calculate discharge flow rate and critical pressure based on isentropic flow and thermal equilibrium assumptions. A correction coefficient of the influence of friction and local resistance is added. The model avoids the calculation of quality and is applicable to wide range which covering the subcooled water, two-phase mixture, steam critical flow under subcritical pressure and SC-pressure. The model calculated results agree well with the experimental critical flow data under SC-pressure.

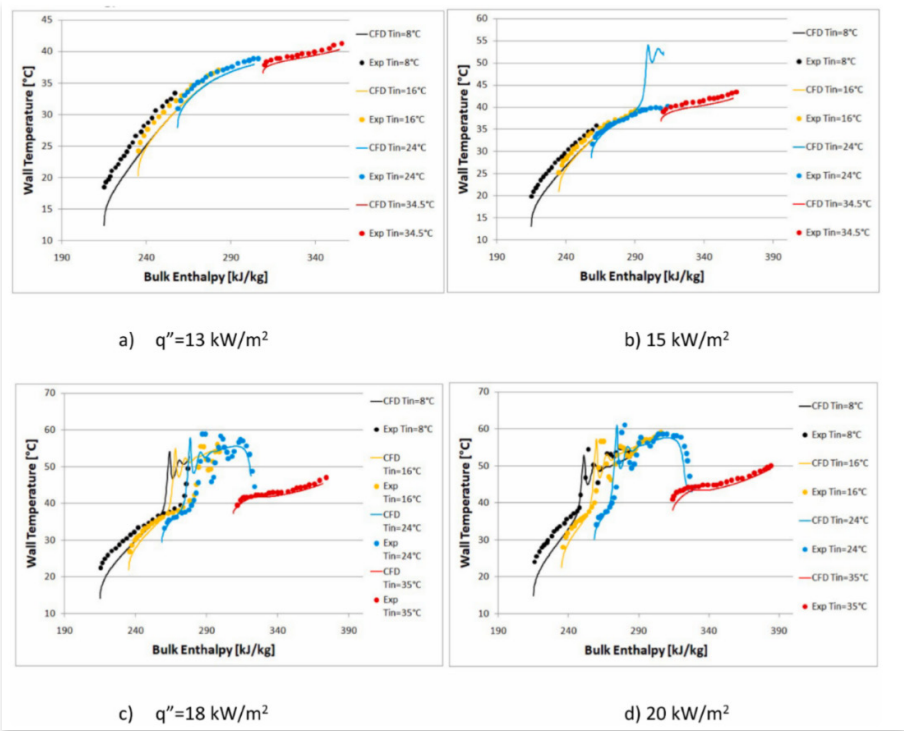
Numerical investigations have been performed by the researcher of BME NTI in 2019 in the field of SCWR TH&S. The thorium fueled SCWR concept was in the focus through a weakly coupled CFD – neutronics code-system. The thermal-hydraulics of the fuel assembly design of Th-SCWR has been simulated by the ANSYS CFX CFD code by a detailed 3D numerical model with and without wrapped wire spacer concept. The density field of SC-water (provided by the CFD results) was handed over to the MCNP Monte Carlo transport code as a boundary condition in each iteration step. The MCNP code calculated the field of heat source and this field was provided in return to the CFD code as a boundary condition. These calculations have proved that the wrapped wire spacer improves the heat transfer in most of the sub-channels within the fuel assembly and an axial and radial fuel enrichment distribution is essential for a viable fuel assembly design from the TH&S point of view. The linear heat source distribution has been optimized in the axial direction, but the maximum wall temperatures still seems to be higher than the melting temperature of currently available cladding materials. So, further optimization of the fuel enrichment is foreseen in the continuation of this research direction.

The CVR main activity on the thermo-hydraulic for Super Critical water Coolant (SCC) is based on the development and license activity that rotate in the insertion of the Super Critical Water Loop (SCWL) in the LVR-15 Reactor. For this reason, a consistent input deck of the facility was developed in ATHLET3.1A Patch1 code. From the several correlations adopted in ATHLET to simulate the SCC media, three were selected and qualified in Czech Republic: Watts-Chou, Mokry and Gupta. However, ATHLET3.1A Patch1 assessment was submitted to the thermo-hydraulic commission managed by the Regulatory State Office for Nuclear Safety (SONS) (Code and User Qualification) and it was qualified in March 2017. After the first revision of the flow regimes, all scenarios are reconsidered due to the lowering the operational pressure of the SCWL from 25 MPa to 24 MPa. The actual activity will focus in completing the flow regime scenarios according to the new specifications. Those selected scenarios analyses are used to verify the system performance in accordance with the safety criteria. A particular attention was given by providing operating regimes data in these conditions for structural analyses.

The University of Pisa developed RANS analyses of CO_2 data in 2017, making use of an Algebraic Heat Flux Model (AHFM) developed in the STAR-CCM+ code, on the basis of the Lien et al. model available in it. The RANS model is being assessed and improved on a variety of experimental data and the very systematic data by Kline offered the opportunity to understand capabilities and limitations of the improved AHFM as developed in this frame. The results, published in different steps showed a remarkable capability of the model to correctly simulate

heat transfer phenomena at relatively low flow rates. In particular, the phenomenon of deteriorated heat transfer termination at the transition to gas-like fluid was observed with reasonable accuracy, as shown in **Figure SCWR 5.**

Figure SCWR 5. **Results obtained for the cases with p=8,35 MPa, ID=4,6 mm and G=300 Kg/m2s**



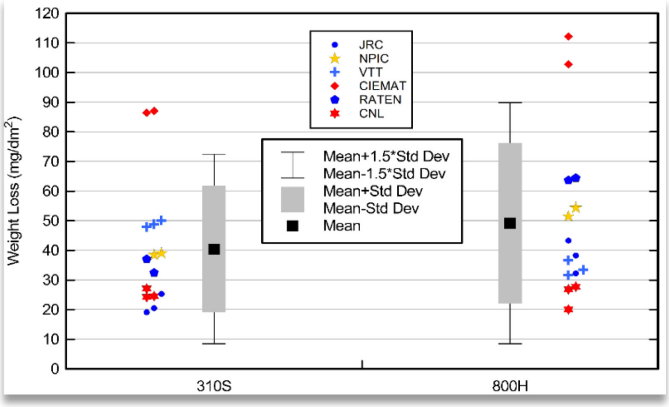
Materials and chemistry

The M&C PMB has been focusing on selection and qualification of candidate alloys for all key components in the SCWR. This includes general corrosion and stress corrosion cracking tests in autoclaves and loops as well as development work on test facilities, ionic irradiation tests of fuel cladding candidate alloy and development of novel alloys for fuel cladding. In addition, modelling of oxide film effects on fuel cladding heat transfer has been performed to better understand the interplay between general corrosion and thermal-hydraulics.

A major activity of the M&C has been the organization of a 2nd Round Robin corrosion test exercise between partners (Canada, China and Euratom) to compare the results of corrosion tests in different test facilities. Each laboratory used a standard test protocol with the coupon materials originating from the same batch and prepared by JRC-IEC. The tests were completed in 2017, and the coupons were sent to CNL for descaling. The results were reported in 2019. After 1 000 h exposure to 550°C supercritical water, coupons of Alloy 800H and Type 310S stainless steel were observed to have (descaled) weight losses of 54±26 mg/dm^2 and 41±22 mg/dm^2, respectively. Interestingly, the data were clustered by participant, with tight agreement between coupons of the same material exposed in the same facility at the same time, shown in Figure SCWR 6. It is not clear if the disagreement among participants is due to differences in flow velocity in the autoclave. It was proposed as a possible explanation.

In 2019, the Canadian materials and chemistry programme focused on expanding the set of high temperature general corrosion data, evaluating the effect of coatings on the corrosion of zirconium and titanium alloys in SC-water (500°C), and developing SC-water test facilities.

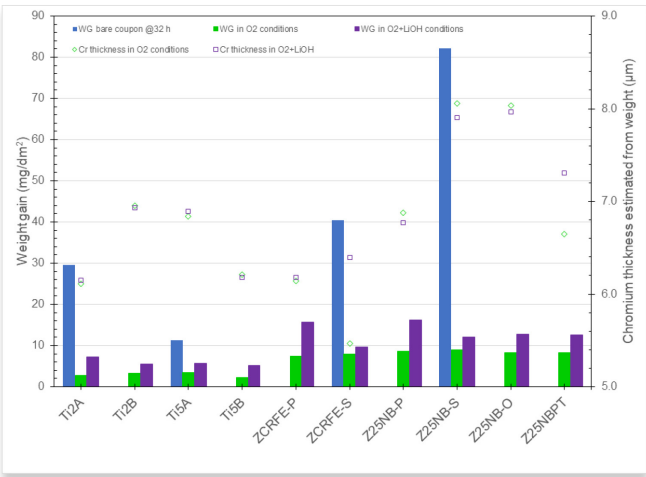
Figure SCWR 6. **Box plot of descaled weight loss data after 1 000 h exposure to deaerated water at 550°C and 25 MPa.**



Alloy 625 is a candidate fuel cladding for the benchmark Canadian SCWR concept with superior corrosion resistance. Experiments were conducted in 2019 that aimed to examine the effects of added hydrogen peroxide on the corrosion of Alloy 625 at temperatures between 650 and 700°C. These experiments also collected thermocouple data from a heated section of the loop that may be helpful in determining the effects of oxide growth on heat transfer. The collected data will be analyzed in 2020.

Corrosion testing of chromium-coated zirconium and titanium alloys was conducted in 2019 in support of the Canadian Small SCWR, a 300 MWe small modular concept. In an effort to improve neutron economy to lengthen the fuel cycle, coated zirconium alloys are being re-evaluated along with titanium alloys; the latter is an unlikely candidate given its poor neutron economy, but a Ti-50 enriched alloy would have good neutron economy and superior corrosion resistance. Coupons of Zr-1.2Cr-0.1Fe (R60804), Zr-2.5Nb (R60901 and R60904), pure titanium (R50400) and Ti-6Al-4V (R56400) were coated with a uniform layer of 10 µm chromium and exposed to 500°C oxygenated SC-water for 150 h, with and without LiOH as a pH-control agent. Weight gain measurements indicate an eight-fold improvement in corrosion resistance of coated coupons compared to the as-received alloys, shown in Figure SCWR 7. Alkaline treatment resulted in much higher weight gains than was found in pure oxygenated SC-water. Microscopic analysis of the coupons, as well as hydrogen uptake measurements, will be performed in 2020.

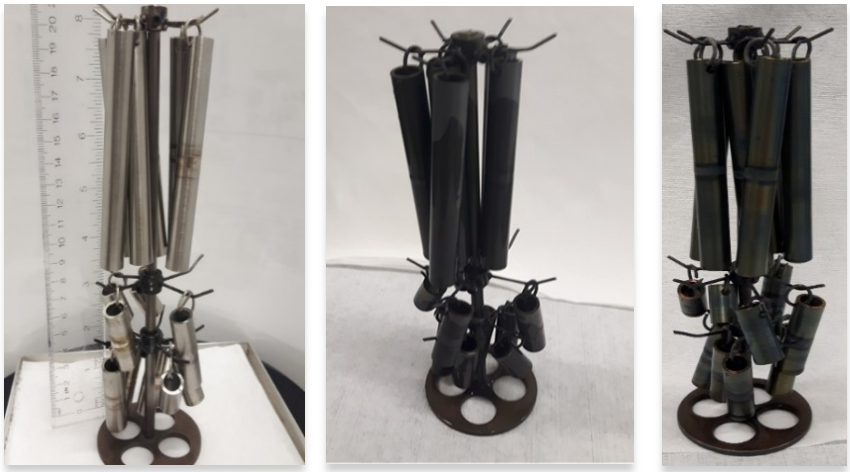
Figure SCWR 7. **Weight gain of chromium-coated zirconium and titanium alloys after exposure to 500°C SC-water for 150 h**



In addition, SC-water test facilities continued to be developed at CNL in 2019. A refreshed SC-water target system for a 2.5 MeV Van de Graaff electron accelerator was commissioned, and continues to be modified and improved. A SC-water hydrocyclone was designed with a MAWP of 29.25 MPa at 649°C (1 200°F) for future studies on activity transport and high temperature purification.

The corrosion and tensile behavior of both Alloy 625 and Alloy 800H welded specimens was examined by exposure to 575°C SC-water for 500 h at CNL. Tube specimens were butt welded autogenously by gas tungsten arc welding using a Swagelok® M200 orbital welding system. For Alloy 800H, post-exposure analysis indicated a 20% increase in the weight gain of welded specimens compared to unwelded specimens. Corrosion and aging of Alloy 800H reduced the ductility of both welded and unwelded specimens by 25% and increased the yield strength by 30%. For Alloy 625, which corrodes very little, welded specimens gained 60% more weight compared to unwelded specimens. Corrosion and aging of Alloy 625 reduced the ductility of both welded and unwelded specimens by 40% and increased the yield strength by 25%. Welded specimens that had been exposed to SC-water were observed to have 10% higher UTS and up to 15% lower ductility when compared unwelded specimens.

Figure SCWR 8. **Before Corrosion Test (left), 800H After 250 hours of Exposure (center) and Alloy 625 After 500 hours of exposure (Right)**

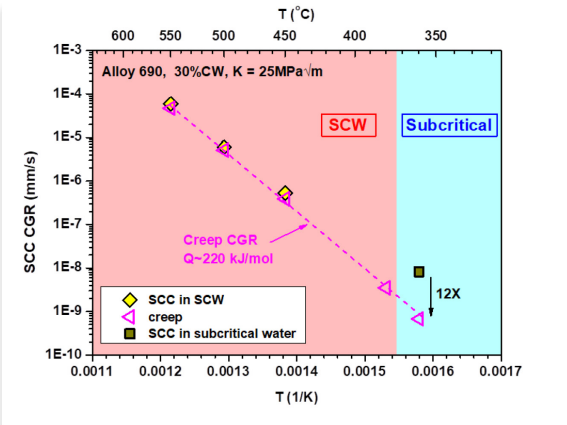


The SCC tests were conducted on 30% cold worked Alloy 690 in high-temperature pressurized water at temperatures between 360°C and 550°C in Shanghai Jiaotong University (SJTU). Creep induced cracking was measured at each testing temperature in inert argon environment to study the effect of creep on the overall crack growth behavior. The crack growth rates at each condition are summarized in **Figure SCWR 9.** Experimental results showed that creep contributed to more than 80% of the overall crack growth rate at temperatures above 450°C in SC-water, while only 8% at 360°C in subcritical water. It was clearly proved that the dominant mode of cracking from subcritical to SC-temperatures are different for cold worked Alloy 690. Corrosion induced cracking controls the crack growth rate in subcritical water environment, while creep is the major factor that dominates the cracking in SC-water.

The effect of InterGranular (IG) carbides on the cracking behavior of cold worked Alloy 690 was also studied in both subcritical and SC-water environments. The crack growth rate was lower when IG carbides were removed by prior solution annealed (SA) treatment, indicating a detrimental effect of IG carbides. The presence of IG carbides enhances the local strain accumulation at the grain boundary due to the lattice mismatch, thus promotes the crack tip strain rate and increases the crack growth rate. The SCC CGRs of Alloy 690 were compared with those of 310S SS in SC-water, and results show that the SCC CGR of the 310S SS specimen in 550°C SCW was 1.4×10^{-7} mm/s, ~17 times higher than the Alloy 690 specimen (8.3×10^{-9} mm/s) at

the same testing condition. The degree of sensitization prior to and after the test was confirmed by both double loop electrochemical potentio-kinetic reactivation (DL-EPR) method and TEM analysis at the grain boundaries. It was found that the degree of sensitization increased dramatically for 310S than Alloy 690 after the SCC test, indicating severe in situ sensitization occurred in 310S during the SCW exposure.

Figure SCWR 9. **Comparison of the SCC and creep CGRs of Alloy 690 in subcritical and supercritical water at the temperatures ranging from 360°C to 550°C**



Two candidate alloys modified from 310S austenitic stainless steels were subjected to ionic irradiation to see their radiation damage effects in Nuclear Power Institute of China (NPIC). The major difference in these two alloys lies in their optimized minor alloying elements, addition of Mo, Nb, W and Ta in alloy SC1, and in SC2, Mo and Zr were added. Proton radiation tests were performed on an ion accelerator at Wuhan University, with implanting energy of 50 keV, and temperature at 290°C up to doses of 0.1 and 0.3 dpa by proton, and at 550°C up to 5, 15 and 30 dpa by Ar ion. Figure SCWR 10 shows the TEM micrographs of specimens irradiated at 550°C and Figure SCWR 11 shows the defect caused by irradiation. The irradiation tests showed that minor alloy elements added to the alloys played different roles after irradiation. At 290°C, Zr modified alloy SC2 showed lower density of void and dislocation loop defects than SC1, which contains Nb, W and Ta. However, at 550°C Zr caused void swelling in SC2 while Nb and Ta in SC1 reduced the density of voids.

Figure SCWR 10. **The TEM micrographs of ally SC1 and SC2 irradiated at 550°C**

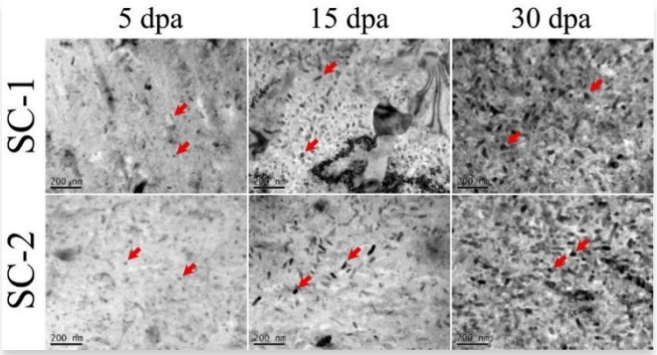
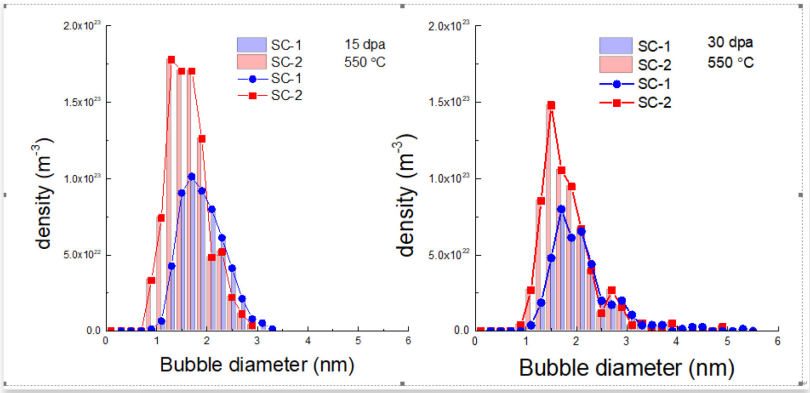
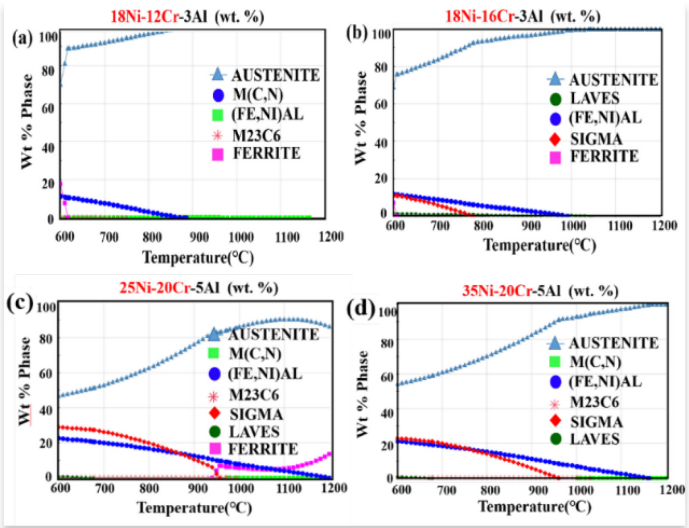


Figure SCWR 11. **The density of defects obtained by TEM of ally SC1 and SC2 irradiated at 550°C**



The major activity at the University of Science and Technology Beijing (USTB) has been focusing on developing of novel candidate alloys for cladding tube application. Alumina Forming Austenitic (AFA) alloy is proposed as a new grade SCWR candidate alloy in view of the existing reported research works, as well as results from the Round Robin corrosion test exercise between PMB partners (Canada, China and Euratom). One of the key challenges for the composition design of AFA alloys is to balance the corrosion resistance and maintain a single austenitic matrix phase for good creep strength. Figure SCWR 12 shows the matrix phases and fractional volume of precipitates in the materials with different Al% at temperatures between 600 and 1 200°C, which was calculated by using the computational thermodynamic calculation program, JMatPro. It is obvious that high Al, Cr content, but insufficient Ni content (25Ni20Cr5Al) will result in the formation of duplex $\gamma+\alpha$ matrix, as Al is a strong ferrite stabilization element. Therefore, the content of Ni should be carefully designed based on the content of Al, Cr and other minor ferritic forming elements to obtain a necessary single γ phase structure. ODS austenitic alloy is the other promising new grade material for the in-core structure application in SCWR. The microstructure stability of a 310 type ODS austenitic alloy after aging at 500°C for different length of time is investigated.

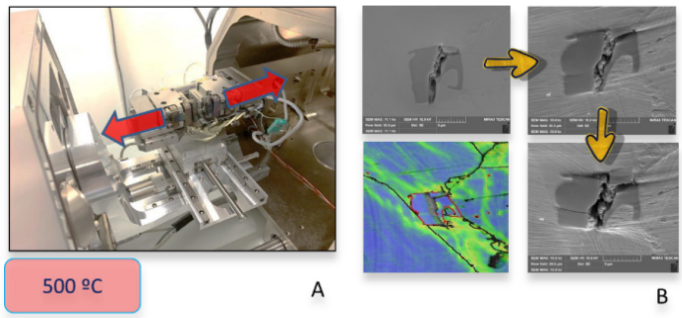
Figure SCWR 12. **Phases in AFA alloys with different Ni, Cr and Al contents based on Jmat- Pro calculations**



Along 2019 Ciemat has performed several in situ tensile tests with a nickel base alloy 690 TT previously tested in deaerated Supercritical Water (SCW) at 500°C. These tests were carried out in collaboration with CVR laboratories. The aim of this work was double, on the one hand, considering this was the first time the staff of CVR in Pilsen work with this material and this specimens geometry, the team tried to write a procedure to study the Alloy 690 TT by means in tensile in situ tests within a Scanning Electron Microscope (SEM) at high temperature (see **Figure SCWR 13**). On the other hand, the second objective was to follow the evolution of precipitates, defects, etc., in the microstructure by means the in situ test. As a result of this first attempt, both laboratories were able to follow the evolution of C,N(Ti) with temperature and strain. This test showed how the plastic deformation gathered around the carbides during the tensile test. This behavior, that has been reported by other groups, could have some meaning in the cracking processes of this material especially if the CNTi are near the surface in contact with SCW at high temperature. After these tests new tensile specimens were designed in order to avoid the problems detected during the first tests. These new specimens will be used to continue this work and to gain more in-depth knowledge into the mechanical behavior of some defects present in this alloy like the vacancies along the grain boundaries that appear as a result of cold work and high temperature. The main results from this work were presented in the EUROCORR congress that was held in Seville (Spain) and in the EPRI meeting EPRI Alloy 690/52/152 Primary Water Stress Corrosion Cracking Research Collaboration Meeting that was held in Tampa (United States).

In addition to this, Ciemat co-ordinates the European funded project MEACTOS in which there is a small task related to SCW. In this task the SCW will be used as an accelerating environment to produce stress corrosion cracks in an austenitic stainless steel 316 L type L.

Figure SCWR 13. **A) Photograph of the device used to perform the tensile in situ tests inside the chamber of a SEM: B) Evolution with temperature and stress of a C,N(Ti) found in a A690 TT specimen previously tested in SCW**



The activities at VTT in 2019 have mainly come through its participation in the EU MEACTOS project where SCW is used as accelerating method of the stress corrosion cracking processes in an austenitic stainless steel. Moreover, VTT participated in the writing of the ECC-SMART proposal for the Horizon 2020 call. In addition to this, VTT has prepared together with Aalto University a project proposal called TAMAT (Towards Advanced Materials for Energy Technologies: Multimetallic Layered Composites and Innovative Cladding Solutions for Nuclear and Beyond) to the Academy of Finland where one work package is dealing with experimental testing and oxide film modelling in supercritical water conditions. The Academy of Finland decision is expected by June 2020.

The M&C in CVR focused on microstructural evaluation of candidate materials for SCWR internals and fuel claddings. First three materials – 800H, T505 (uquivalent of T91) and 08Cr18Ni10Ti (equivalent of AISI 321) were exposed in the supercritical water loop(SCWL) in 2018. In the end of the 2019, next exposure in SCWL started with another three candidate materials (Nimonic 901, Nitronic 60 and In 718). First corrosion exposure up to 550 h and second exposure up to 1 000 h duration were carried out at 400°C/ 25 MPa with deoxygenated water, pH 6, conductivity under 2 µS/cm, Fe < 100 µg/l.

Materials from first exposure were analyzed by SEM technique in combination with EDX for chemical composition and EBSD for crystallography. Final Raman and XRD analysis confirmed compounds of magnetite oxides (Fe_3O_4) on all surfaces. No significant oxide layer occurred on 800H and 08Cr18Ni10Ti, only random not compact oxide particles. Double spinel (3-7 µm) layer occurred on T505: inner passivation layer of chromite $FeCr_2O_4$/trevorit $NiFe_2O_4$ and outer layer of magnetite Fe_3O_4. Other investigations in SCW were development works on two autoclaves with parameters: volume 137 ml, 600°C/25 MPa and volume 850 ml, 700°C/30 MPa. These autoclaves are supposed to work from 2020. One more autoclave is supposed to be developed in hot cells next year, to expose irradiated materials to SC-Water.

Yanping Huang

Chair of the SCWR SSC and all Contributors

Sodium-cooled fast reactor (SFR)

Main characteristics of the system

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

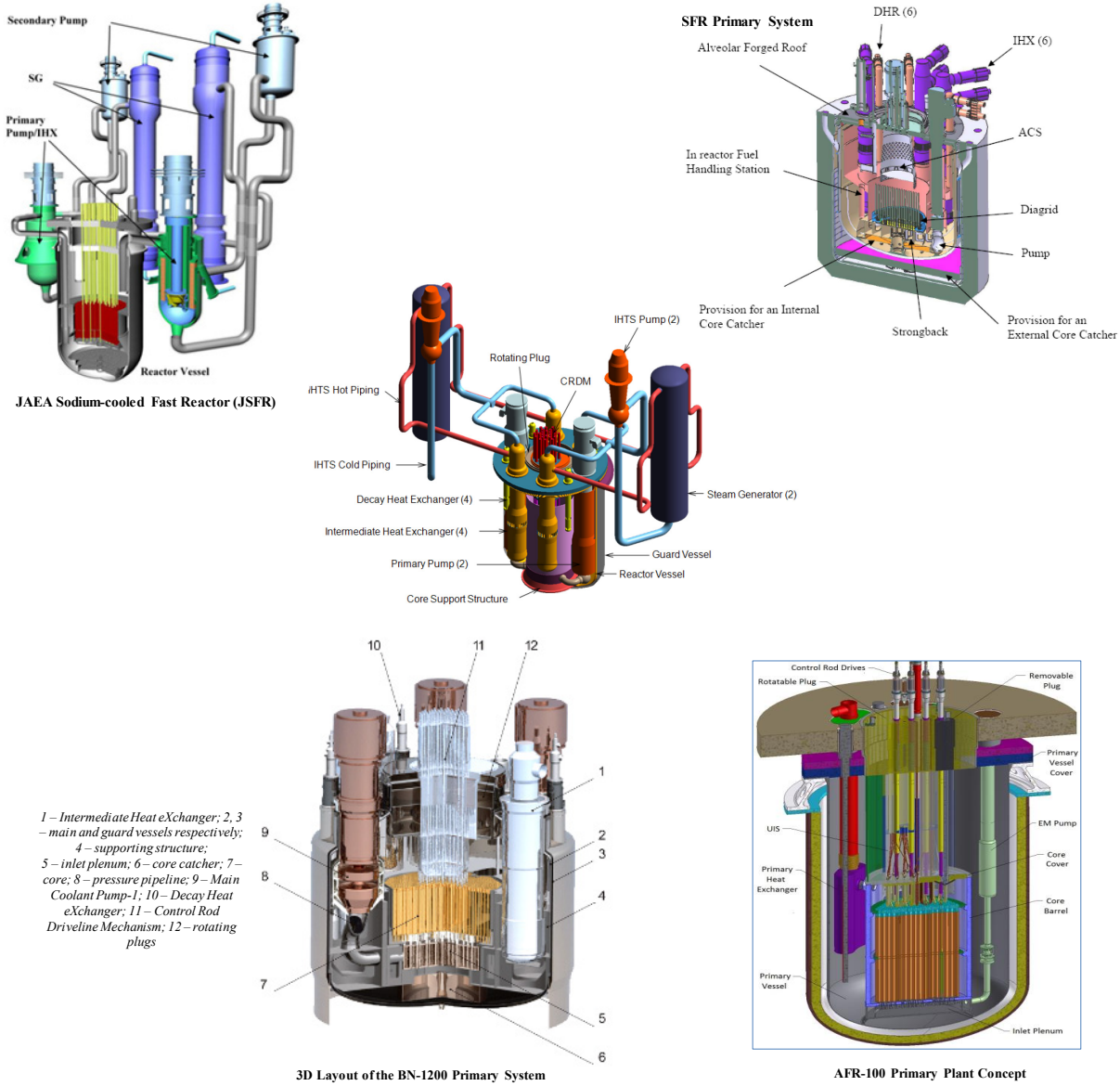
The SFR system uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction. Because of advantageous thermo-physical properties of sodium (high boiling point, heat of vaporization, heat capacity, and thermal conductivity) there is a significant thermal inertia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

Table SFR 1. **Typical Design Parameters for the Generation IV SFR**

Reactor parameters	Reference value
Outlet Temperature	500-550°C
Pressure	~1 Atmosphere
Power Rating	50-2000 MWe
Fuel	Oxide, metal alloy, others
Cladding	Ferritic-Martensitic, ODS, others
Average Burn-up	150 GWD/MTHM
Breeding Ratio	0.5 -1.30

There are many sodium-cooled fast reactor conceptual designs that have been developed worldwide in advanced reactor development programs. In particular, the BN-800 Reactor in Russia, the European Fast Reactor in the EU, the Advanced Liquid Metal Reactor (PRISM) and Integral Fast Reactor Programs in the United States, and the Demonstration Fast Breeder Reactor in Japan, have been the basis for many SFR design studies. For Gen-IV SFR research collaboration, several *system options* that define general classes of SFR design concepts have been identified: loop configuration, pool configuration, and small modular reactors. Furthermore, within this structure several *design tracks* that vary in size, key features (e.g. fuel type) and safety approach have been identified with pre-conceptual design contributions by Gen-IV SFR Members: JSFR (Japan), KALIMER (Korea), ESFR (Euratom), BN-1200 (Russia), and AFR-100 (United States) (see **Figure SFR.1**). The Gen-IV SFR design tracks incorporate significant technology innovations to reduce SFR capital costs by a combination of configuration simplicity, advanced fuels and materials, and refined safety systems; thus, they are utilized to guide and assess the Gen-IV SFR R&D collaborations.

Figure SFR 1. **The five Gen-IV SFR design tracks**



Status of co-operation

The system arrangement for the Gen-IV international R&D collaboration for the SFR nuclear energy system became effective in 2006 and was extended for a period of ten years in 2016. Several new Members were added to the original agreement and United Kingdom was welcomed to the system arrangement in 2019. The present signatories are: Commissariat à l'énergie atomique et aux énergies alternatives, France; US Department of Energy; Joint Research Centre, Euratom; Japan Atomic Energy Agency; Ministry of Science and ICT, Korea; China National Nuclear Corporation; Rosatom, Russia; UK Department of Business, Energy and Industrial Strategy.

Based on international R&D plans, the Gen-IV SFR research activities are arranged by the SFR Signatories into four technical Projects: System Integration and Assessment (SIA), Safety and Operations (SO), Advanced Fuels (AF) and Component Design and Balance-of-Plant (CDBOP).

Three Project Arrangements (PAs) were signed in 2007: Advanced Fuel (AF), Component Design and Balance-of-Plant (CDBOP), and Global Actinide Cycle International Demonstration (GACID). The PA for Safety and Operation (SO) was signed in 2009, and the PA for System Integration and Arrangement (SIA) was signed in 2014. The Project Arrangements were agreed for a ten-year term with annual updates of the Member contributions. The PA for AF and the PA for GACID expired in 2017. A new PA (Phase II) for AF for next ten years was entered into force in 2018. The PA for CDBOP and SO extended for another ten years in 2017 and 2019 respectively.

R&D objectives

SFR designs rely heavily on technologies already developed and demonstrated for sodium-cooled reactors and associated fuel cycle facilities that have successfully been built and operated in several countries. Overall, approximately 400 reactor years of operating experience have been logged on SFRs, including 300 years on smaller test reactors and 100 years on larger demonstration or prototype reactors. Significant SFR research and development programs have been conducted in the United States, Russia, Japan, France, India[1] and the United Kingdom. The only SFR power reactors in operation are the BN-600 (Russia) which has reliably operated since 1980 with a 75% capacity factor, and the BN-800 which started commercial operations in 2016. Currently operating test reactors include BOR-60 (Russia), and CEFR (China). The JOYO (Japan) test reactor is in licensing process for restart. New SFR test reactors MBIR (Russia) and VTR (United States) are expected in the next decade. In addition, SFR technology development programs are being pursued by all members of the GIF SFR System Arrangement.

A major benefit of previous investments in SFR technology is that the majority of the R&D needs that remain for the SFR reactor technology are related to performance rather than viability of the system. Accordingly, the Generation IV collaborative R&D focuses on a variety of design innovations for actinide management, improved SFR economics, development of recycle fuels, in-service inspection and repair, and verification of favorable safety performance.

System integration and assessment project (SIA): Through systematic review of the Technical Projects and relevant contributions on design options and performance, the SIA Project will help define and refine requirements for Generation IV SFR concept R&D. The Generation IV SFR system options and design tracks are identified and assessed with respect to Generation IV goals and objectives. Results from the technical R&D projects will be evaluated and integrated to assure consistency.

Safety and operation project (SO): The SO project is arranged into three work packages (WPs) which consist of WP SO 1 “Methods, models and codes” for safety technology and evaluation, WP SO 2 “Experimental programmes and operational experience” including the operation, maintenance and testing experience in the experimental facilities and SFRs (e.g. Monju, JOYO, Phénix, BN-600, BN-800 and CEFR), and WP SO 3 “Studies of innovative design and safety systems” related to the safety technology for the Gen-IV reactors such as inherent safety features and passive safety systems.

Advanced Fuel project (AF: presently expired and phase II project is under preparation): The Advanced Fuel Project aims at developing and demonstrating minor actinide-bearing (MA-bearing) high burn-up fuel for SFRs. The R&D activities of the Advanced Fuel Project include fuel fabrication, fuel irradiation and core materials (e.g. cladding materials) development. The advanced fuel concepts include non-MA-bearing driver fuels for reactor start-up as well as MA-bearing fuels as driver fuels and targets dedicated to transmutation, in order to address both homogeneous and heterogeneous ways of MA transmutation as a long-term goal. Fuels considered include oxide, metal, nitride and carbide. Currently, cladding/wrapper materials under consideration include austenitic as well as ferritic/martensitic steels but aim to transition in the longer term to other advanced alloys, such as ODS steels.

1. India is not belonging to GIF.

Component design and balance-of-plant project (CD&BOP): The project includes the development of advanced Energy Conversion Systems (ECS) to improve thermal efficiency and reduce secondary system capital costs. The project also include R&D on advanced in-service inspection and repair (in sodium) technologies, small sodium leak consequences, and new sodium testing capabilities. The main activities in energy conversion systems include: (1) development of advanced, high reliability steam generators and related instrumentation; and (2) the development of advanced ECS based on a Brayton cycle with supercritical carbon dioxide or nitrogen as the working fluid. In addition, the significance of the experience that has been gained from SFR operation and upgrading is shared.

Main activities and outcomes

In this Section, recent Member contributions to the Gen-IV SFR collaboration are highlighted.

System integration and assessment (SIA) Project: In 2019, five trade and assessment studies were contributed. CIAE contributed a study that evaluates the CFR1200 design main heat transfer parameters. Key factors that significantly influence the thermal performance were identified (e.g. primary/secondary circuit temperatures). They performed sensitivity analyses for these main factors and quantified the impacts on system efficiency and component design.

During the design phases of the ASTRID demonstrator (2010-2019), CEA continuously assessed and improved the design of ASTRID to enhance its safety. It is a good example of how SFR safety can be improved by design with a core showing favorable natural behavior under multi-failure accident conditions, and with added devoted complementary safety devices to prevent or mitigate severe accidents. In 2019, based on the ASTRID design evolutions, CEA is providing its feedback experience on the SFR safety enhancement by design.

Within the ESFR-SMART project, various safety improvements for ESFR have been proposed, taking into account the safety objectives envisaged for Gen-IV reactors and the recommendations following the Fukushima Daiichi accident. The Euratom contributions provide overviews of the improved ESFR safety approach including the safety requirements for the evaluation of the innovative design options, the assessments of the proposed system safety measures and recommendations for further developments. Safety approaches assessments were performed using the GIF RSWG ISAM methodology relevant tools: Qualitative Safety features Review (QSR) and Objective Provision Tree (OPT). The focus of the contribution for this year is the use of ISAM QSR including a short description of the QSR approach for SFR and a checklist of recommendations developed for a generic SFR concept. The contribution discusses the assessment of the checklist for the ESFR-SMART considering compliance with defence-in-depth, safety objectives, ALARA principle and need for harmonization of safety and security architecture. Included are recommendations and conclusions on the QSR application.

JAEA contributed a study on countermeasures against sodium-water reactions. A single tube helical coil steam generator was evaluated as a design alternative to the JSFR double-wall type. Failure propagation and leak detection behavior was compared for the two concepts. Future work will include a detailed evaluation of sodium-water pressure and system impacts.

KAERI performed a deployment scenario study of large size TRU burners to estimate spent fuel accumulation from PWR operation and to evaluate radiotoxicity reduction of spent fuels by introducing TRU burners. The spent fuel accumulation from PWR operation was estimated based on domestic plans for long-term electricity demand and supply. The spent fuel accumulation of TRU recycle was compared to that of direct disposal, and the radiotoxicity of finally disposed high-level wastes of TRU recycle reaches natural uranium level after about 5 000 years.

Safety and Operations Project: As the topic of the SO project, the common project that consists of two benchmark analyses (EBR-II test and PHÉNIX Dissymmetric tests) have been started in the SO project last quarter of 2019. The first phase of the benchmark analysis ("blind phase") will take two years.

The SO project is structured in three work packages (WPs): WP SO 1 "Methods, models and codes", WP SO 2 "Experimental programmes and operational experiences" and WP SO 3 "Studies of innovative design and safety systems".

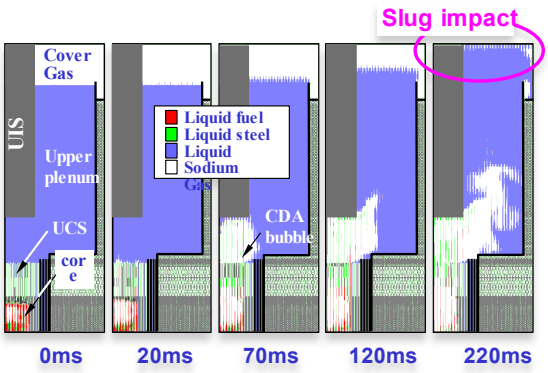
WP SO 1: Methods, models and codes

CIAE conducted benchmark analysis for EBR-II Shutdown heat removal tests SHRT-17 and SHRT-45R, as a part of joint work with ANL from 2017 to 2019.

In order to improve the simulation of an Unprotected Loss of Flow transient, CEA studied the way to calculate the reactivity coefficients to be used in the system code point kinetics module. MACARENa and APOLLO3 Codes were used for this study. As a result, it was confirmed that the point kinetics parameters (in particular the sodium-void reactivity worth) were strongly affected by 3D angular effects. Meanwhile, the transient simulation results were not very different. Hence, the most significant progress is probably to be found in improving the neutronic/thermo-hydraulic coupling. In order to demonstrate that a severe accident is sufficiently unlikely, CEA provided the deliverable which presents the safety demonstration methodology, measurement systems, and reactor protection sub-systems corresponding with each core meltdown initiating events.

JAEA developed the evaluation method of the consequence of energetics in Post-Disassembly Expansion (PDE) phase during an unprotected loss-of-flow accident. They performed the preliminary evaluation of mechanical energy and reactor vessel response using the developed method, see **Figure SFR 2**. As a result, no Slug Impact nor residual strain of the reactor vessel was predicted in the case of realistic temperature condition. Therefore, they obtained the perspective of the robustness of prototype SFR against the energetics in severe accident conditions.

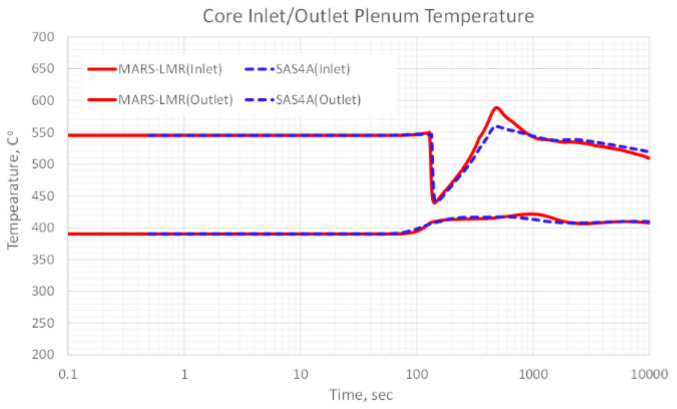
Figure SFR 2. **Material distribution calculated by JAEA (pessimistic temperature condition)**



In order to obtain licensing approval for the developed code (MARS-LMR), KAERI carried out the comparative safety analysis with SAS4A/SASSYS-1 code. In **Figure SFR 3**, sample results from the comparison of the Transient Over Power, Loss Of Flow, and Loss Of Heat Sink cases are shown. The safety analysis results from each code were found to be in good agreement.

IPPE (Rosatom) continued developing 3D severe accident analysis code COREMELT3D. The 3D model of the reactor gas system (from the gas volume under the sodium level in the reactor through the expansion tank up to the ventilation system) has been developed and implemented into the code. This model has been integrated with the primary circuit 3D thermo-hydraulic model, it is necessary for simulating transport of gaseous fission products from disintegrated fuel pins up to ventilation system, and consequently into the environment. IPPE has performed integral analysis of the consequences of the severe accidents in BN-1200. There have been used the following codes: COREMELT3D (core, primary and intermediate circulation loops, emergency system of heat removal, reactor gas system), KUPOL-BR (ventilation system), VYBROS-BN (transport of radioactive products in the environment under different meteorological conditions, doses). IPPE has performed preliminary experiments with thermite compositions to obtain melt of stainless steel with high temperature. This technique will be used in a facility (which is being designed now) to simulate transport of melt in the SFR conditions.

Figure SFR 3. **Comparison between MARS-LMR and SAS4A/SASSYS-1 codes on Loss Of Flow case**



Safety analysis results of core temperatures for LOHS accident

WP SO 2: Experimental programmes and operational experiences

CIAE conducted the experimental research and the code development for Heat Transfer Analysis of CEFR Damaged Spent Fuel Assemblies in Closed Space. The experiment simulated the spent fuel assemblies during transportation and the heat transfer characteristics were investigated.

The Project ESFR-SMART aims to evaluate the safety of a low-void Sodium Fast Reactor (SFR) core design, in particular the analysis of an unprotected loss-of-flow (ULOF) accident. Recent studies on the low-void SFR core show the occurrence of a stabilized chugging sodium boiling regime that can be classified as a new safety measure acting as a level of defence preventing severe accidents. In order to better understand and simulate the chugging boiling regime condition and to gather new experimental data, the ESFR-SMART project envisaged the construction of a new simple facility named CHUG (see **Figure SFR 4)**, designed using water as simulant. The Euratom contribution describes the pre-test calculation results, as well as the facility layout for the first phase of the test, including the main parts and the instrumentation. Preliminary results and main outcomes of first phase of experiments are summarized. Results of analytical simulations of the experiment conducted using the thermal hydraulics code TRACE to assess the validity of the code for the simulation of chugging boiling are shared.

Figure SFR 4. **Layout of the CHUG facility**

Euratom discusses design guidelines for sodium loops. Using liquid sodium at high temperatures in test facilities requires defining rules specific to this technology to ensure that operations are safe and reliable. The purpose of this contribution is to explain the safety rules to be incorporated by the designer during the definition of a project to build a facility implementing liquid sodium. The recommendations take into account European feedback on safety issues related to the design of sodium facilities. However, they do not under any circumstances replace the regulations in force applicable to each subject discussed.

WP SO 3: Studies of innovative design and safety systems

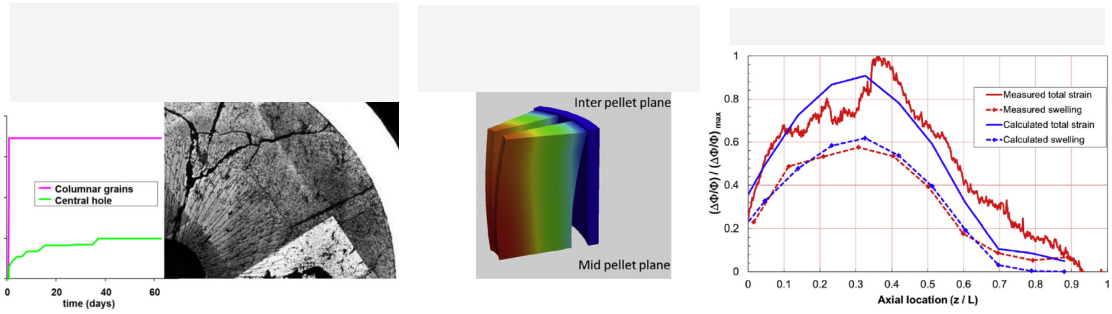
In order to confirm the applicability of Self-Actuated Shutdown System (SASS) in conditions of low power operation, JAEA carried out design modification and 3D thermal-hydraulic analysis and calculated the response time of SASS. By introducing the improvement of design and the required temperature difference, core damage was prevented by the SASS in the case of LOF type ATWS event from low power operation.

Advanced Fuels Project: The AF project consists of three work packages: WP 2.1 "SFR Non-MA-bearing Driver Fuel Evaluation, Opt. & Demo.", WP 2.2 "MA-bearing Transmutation Fuel Evaluation, Opt. & Demo." and WP 2.3 "High-burn-up Fuel Evaluation, Opt. & Demo."

WP 2.1: SFR Non-MA-bearing Driver Fuel Evaluation, Optimization & Demonstration

CEA presented the current capabilities of the GERMINAL fuel performance code, part of the PLEIADES simulation platform, used for (U, Pu) mixed oxide fuel pins calculations. The modelling of GERMINAL and its validation by comparison between calculations and measurements have been shared (see **Figure SFR 5**).

Figure SFR 5. **Examples of calculations and validation studies for GERMINAL code**



DOE continued to develop simulation tools for the evaluation of metallic fuel performance. Additional models were added and improved in the BISON fuel performance code to enhance its ability to model both U-Pu-Zr and MOX fuel for sodium fast reactors. Also, DOE successfully fabricated novel geometries of Pu bearing metal fuel to support accelerated testing. This includes both small diameter samples needed for high fission rate testing as well as more complex fuel geometries that can explore alternative methods to accommodate swelling that eliminate the need for sodium bonding.

JAEA measured the physical properties of non-stoichiometric (U, Pu)O_2 as function of Pu content, Am content, O/M ratio and temperature.

WP 2.2: MA-bearing Transmutation Fuel Evaluation, Optimization & Demonstration

CEA has performed a preliminary thermo-mechanical design of a MA-bearing oxide pin loaded with 10% of americium in UO_2 matrix. The behavior of the pin has been calculated with GERMINAL fuel performance code with specific developments for MA-bearing fuels.

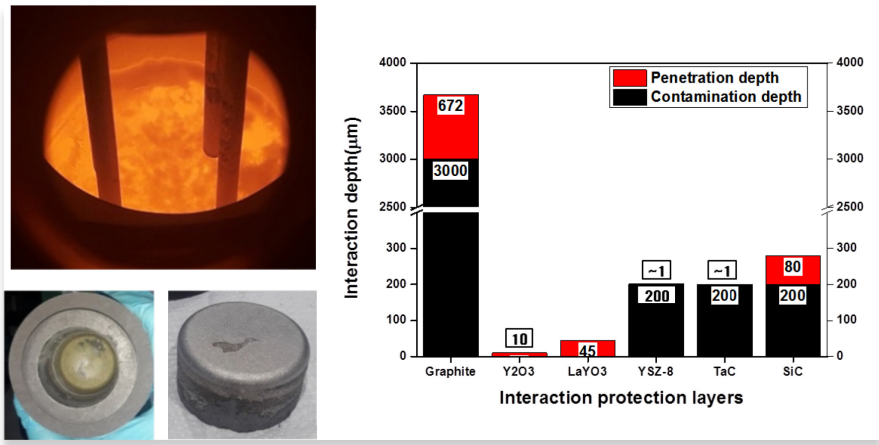
DOE investigated the effect of minor actinide additions to metallic fuel through post-irradiation examination (PIE) and micro-chemical analysis of minor actinide bearing transmutation fuel. Minor actinide bearing fuel irradiated in several different reactors including true fast spectrum reactors (EBR-II, Phenix) and pseudo-fast spectrum tests in the Idaho National Laboratory Advanced Test Reactor were all compared. The fuel performance across these different conditions was fundamentally the same, and the addition of minor actinides did not significantly change the performance of the fuel. The "Metallic Fuel Handbook" which documents the fundamental thermo-physical properties of metallic fuel alloys and their constituents was updated with a significant revision to the U-Zr and U-Pu-Zr system information.

Euratom JRC installed the Cold Finger Apparatus (KüFA) installed into the JRC hot cell facilities for out-of-pile safety transient testing. A temperature transient up to 1 800°C will be applied and the release of gaseous and solid fission products will be quantitatively determined as a function of time. Additionally, JRC studied on Synthesis of Am-bearing MOX fuel for the homogeneous recycling concept ($(U,Pu,Am)O_2$), containing circa 5% americium, 20% plutonium and 75% uranium. The proposed synthesis method synthesize $(U,Th)O_2$ nanopowder (particle size is about 5 nm).

JAEA evaluated effects of Am on MOX fuel temperature using an irradiation behavior analysis code, it was suggested Am-MOX fuel could be irradiated with the same conditions as conventional MOX fuel.

KAERI completed fuel rod fabrication for the 2nd Fuel irradiation test in HANARO (SMIRP-2 test), which will be started from 2020. They also conducted the development of reusable crucible and mold for metal fuel fabrication, see Figure 3.10. In order to reuse casting parts, various new materials were tested for the coating material, which confirmed the effectiveness of the Y_2O_3 coating.

Figure SFR 6. **Development of reusable crucible and mold**



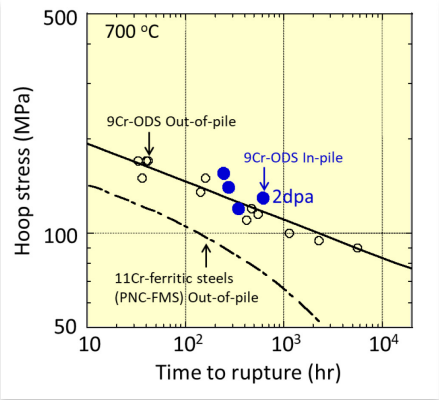
Rosatom created the experimental installation of low temperature sintering of mixed nitride uranium and plutonium fuel by the HVPC-process. Through the test using titanium nitride, they confirmed good repeatability of the HVPC-process. And they started the testing of the installation via uranium nitride.

WP 2.3: High-burn-up Fuel Evaluation, Optimization & Demonstration

CIAE updated the oxide fuel performance code, FIBER, to analyze up to 10at% burn-up fuel. They conducted the verification of the FIBER code and the benchmark analysis with the past CEFR calculation.

JAEA has developed 9Cr-ODS tempered martensitic steel (TMS) as prospective material for high burn-up fuel cladding tube. JAEA confirmed that there was no remarkable degradation in the in-pile creep mechanical strength of 9Cr-ODS TMS cladding tubes irradiated in JOYO (see **Figure SFR 7**). Furthermore, on the basis of knowledge on 9Cr-ODS TMS having prominent mechanical strength and irradiation resistance, JAEA started developing a new type of high Cr-ODS TMS: the 11Cr-ODS TMS for improving corrosion resistance.

Figure SFR 7. **Comparison of In-pile and Out-of-pile creep rupture strength of 9Cr-ODS TMS**



KAERI conducted the parametric study and sample manufacturing for Cr electroplating for nuclear cladding applications. And they also conducted several performance tests (Out-of pile diffusion couple test, mechanical test).

Rosatom have planned the Post-Irradiation Examination (PIE) for the specimens made of EP823 ODS steel in order to increase the burnout level of the fuel. Rosatom fabricated the samples for this PIE and conducted the pre-reactor tests. Additionally, the first stage of irradiation was completed and they are conducting the PIE.

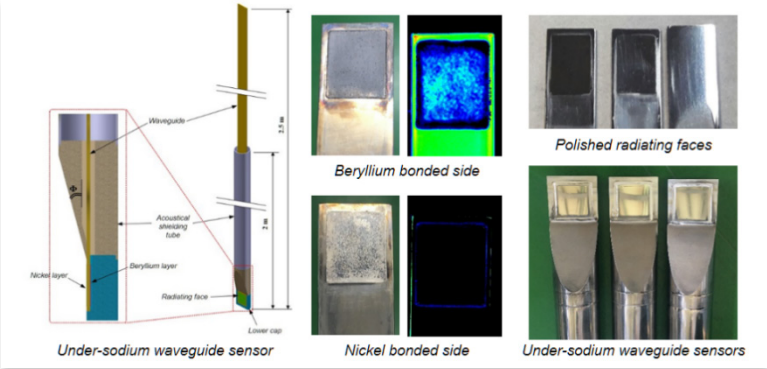
Component Design and Balance-Of-Plant Project: Activities within the CD&BOP project include experimental and analytical evaluation of advanced In-Service Inspection, Instrumentation & Repair technologies (ISI&R), development of Advanced Energy Conversion Systems (AECS), study of sodium Leakages and Consequences (SL), advanced Steam Generator technologies (SG), and study of sodium Operation technology and new sodium Testing Facilities (O&TF).

ISI&R technologies

CEA have studied the capability of the Leaky Lamb Waves on the non-destructive testing from outside of the main vessel. They developed the model to represent the behavior of Leaky Lamb Waves in plates, and validated it by comparing with the literature results. Additionally, for the further validation, they prepared experimental devices consisting of immersed plates, emitter and receiver.

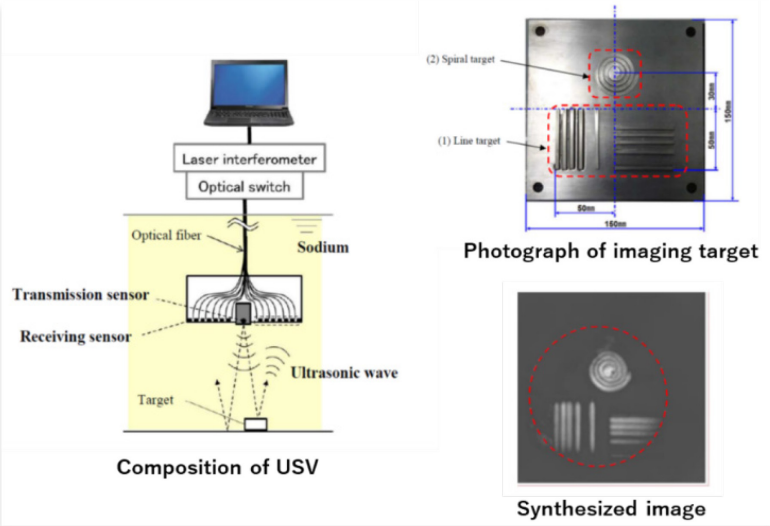
KAERI investigated the sodium-wetting property of the plate-type ultrasonic waveguide sensor under various conditions to improve the sensor performance. KAERI fabricated waveguide sensors with different surface roughness of radiating faces, and prepared a sodium-wetting test facility. They also constructed a new sodium test facility for the verification of under sodium viewing and ranging capabilities of the plate-type ultrasonic waveguide sensor as shown in Figure SFR.8.

Figure SFR 8. **Waveguide sensors fabricated for sodium-wetting tests**



JAEA developed an improved imaging under sodium viewer for a middle distance (see Figure SFR.9). The transmission sensor provided better profile of the wave, and the receiving sensor successfully reduced the noise of the wave profile. The imaging experiment in water showed that higher resolution can be obtained through the improvement of the imaging under sodium viewer.

Figure SFR 9. **Imaging experiment in water for the improved imaging under sodium viewer**

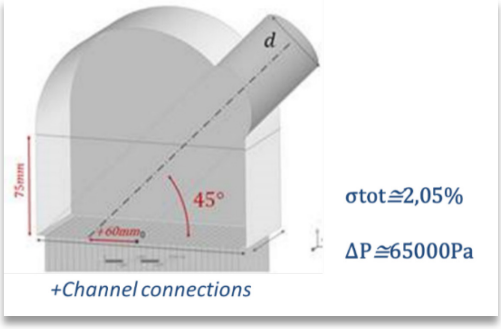


Supercritical CO_2 Brayton cycle

CEA is developing the method to detect the bubbles in the sodium flow by using eddy current flowmeter (ECFM). In 2019, CEA proposed a numerical representation of an ECFM and a bubble. Through the comparison study with experimental result, it was confirmed that this method can detect the bubble effectively.

CEA also conducted the parametric study and the optimization of the design of the heat exchanger's header. Their CFD models validated by the experimental result was used for this optimization. As the result of the optimization of the header and the channels bundle (see **Figure SFR 10**), the maldistribution level was reduced from 25% to about 2%, as compared to the design objective of 5%.

Figure SFR 10. **Design of heat exchanger's optimized header**



Sodium leakages and consequences

In 2019, no specific activity was conducted in this work package.

Steam generators

JAEA studied on the heat transfer coefficient inside tube in case of sodium-water reaction. The overheating tube rupture is one of the considerable failure mode derived from the sodium-water interaction. For the evaluation of the possibility of this mode, heat transfer coefficient of water side is important factor. JAEA conducted the rapid heating experiments for the tube containing water flow and estimated the heat transfer characteristics of inner surface. Based on this experimental results, correlations for RELAP5 code was modified and their conservativeness were confirmed.

KAERI has upgraded the signals analysis software as well as the combined SG tube inspection sensor and signal acquisition device. The upgraded software newly employs several signal transformation functions for MFL image processing, and an automatic defect detection algorithm. They conducted performance tests of the upgraded prototype combined SG tube inspection sensor system, and confirmed its defect detection performance.

Sodium operation technology and new sodium testing facilities

KAERI has completed the installation of the STELLA-2 test section in 2019, and remaining works for a cold test will be finished in early 2020. The first test data for sodium integral effect test using STELLA-2 will be obtained no later than the end of 2020 as well. Besides the sodium thermal-hydraulic test program, KAERI is constructing new test facilities for sodium leak detection and simulation, which are called WALSUM (Water-mock-up test for Advanced Leak Simulation and Upgraded Monitoring system) and SELAAD (Sodium Experimental Loop for Advanced Aerosol Detection). The objectives of the new facilities are to develop highly reliable sodium leak detection and monitoring system as well as performance evaluation of advanced sodium leak detectors.

Bob Hill

Chair of the SFR SSC and all Contributors

Very-high-temperature reactor (VHTR)

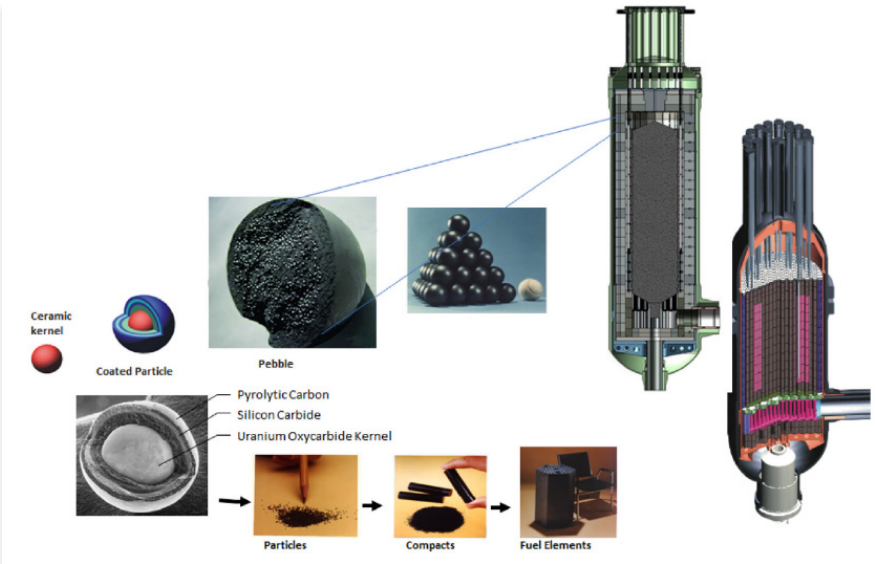
Main characteristics of the system

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

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

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

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



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

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

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

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

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

Status of co-operation

The VHTR system arrangement was signed in November 2006 by Canada, Euratom, France, Japan, Korea, Switzerland and the United States. In October 2008, China formally signed the VHTR SA during the Policy Group meeting held in Beijing. South Africa, formally acceded to the GIF Framework Agreement in 2008, but announced in December 2011 that it no longer intends to accede to the VHTR SA. Canada withdrew from the SA at the end of 2012 but is again an observer and remained active in the Hydrogen Production Project. The new members of the system arrangement was subsequently signed by Australia (December 2017) and the United Kingdom (January 2019).

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

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

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

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

R&D objectives

Even if the VHTR development is mainly driven by the achievement of very-high-temperatures providing higher thermal efficiency for new applications, other important topics are driving the current R&D: demonstration of inherent safety features and high fuel performance (temperature, burn-up), coupling with process heat applications, cogeneration of heat and power, and the resolution of potential conflicts between those challenging R&D goals.

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

- Fuel and fuel cycle (FFC) investigations are focusing on the performance of the TRISO coated particles, which are the basic fuel concept for the VHTR. R&D aims to increase the understanding of standard design (UO_2 kernels with SiC/PyC coating) and examine the use of uranium-oxicarbide UCO kernels and ZrC coatings for enhanced burn-up capability, best fission product confinement and increased resistance to core heat-up accidents (above 1 600°C). This work involves fuel characterization, post-irradiation examination, safety testing, fission product release evaluation, as well as assessment of chemical and thermo-mechanical materials properties in representative service and accident conditions. The R&D also addresses spent fuel treatment and disposal, including used-graphite management, as well as the deep burn of plutonium and minor actinides (MA) in support of a closed cycle.
- Materials (MAT) development and qualification, design codes and standards, as well as manufacturing methodologies, are essential for the VHTR system development. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in the operating environments. For core coolant outlet temperatures up to around 950°C, it is envisioned to use existing materials; however, the stretch goal of 1 000°C, including safe operation under off-normal conditions and involving corrosive process fluids, requires the development and qualification of new materials. Improved multi-scale modelling is needed to support inelastic finite element design analyses. In addition to other high-temperature heat exchangers, additional attention is being paid to the metal performance in steam generators, which reflects the current interest in steam-based process applications at somewhat lower core outlet temperature of 750 to 850°C. Structural materials are considered in three categories: graphite (for core structures, fuel matrix, etc.), very/medium-high-temperature metals, and ceramics & composites. A materials handbook has been developed and is being used to efficiently store and manage VHTR data, facilitate international R&D co-ordination, and support modelling to predict damage and lifetime assessment.
- For hydrogen production (HP), two main processes for splitting water were originally considered: the sulphur/iodine thermo-chemical cycle and the high-temperature steam electrolysis process. Evaluation of additional cycles has resulted in focused interest on two additional cycles with lower temperature requirements: the hybrid copper-chlorine thermo-chemical cycle and the hybrid sulphur cycle. R&D efforts in this PMB address feasibility, optimization, efficiency and economics evaluation for small and large-scale hydrogen production. Performance and optimization of the processes will be assessed through integrated test loops, from laboratory scale through pilot and demonstration scale, and include component development such as advanced process heat exchangers.

Hydrogen process coupling technology with the nuclear reactor will also be investigated and design-associated risk analysis will be performed covering potential interactions between nuclear and non-nuclear systems. Thermo-chemical or hybrid cycles are examined in terms of technical and economic feasibility in dedicated or cogeneration hydrogen production modes, aiming to lower operating temperature requirements in order to make them compatible also with other Generation IV nuclear reactor systems dealing with a lower temperature range.

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

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

Milestones

In the near term, lower-temperature demonstration projects (from 700°C to 950°C) are being pursued to meet the needs of current industries interested in early applications. Future operation at higher temperatures (1 000°C and above) requires development of high-temperature alloys, qualification of new graphite types and development of composite ceramic materials. Lower temperature version of VHTR (from 700°C to 950°C) will enter the demonstration phase around 2020, based on HTR-PM experience in China which is scheduled to operate in 2020. A future higher temperature version (1 000°C and above) will require more research.

Main activities and outcomes

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

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

measurement of chemical and thermo-mechanical material properties in representative conditions will feed a fuel material data base. Further development of physical models enables assessment of in-pile fuel behavior under normal and off normal conditions.

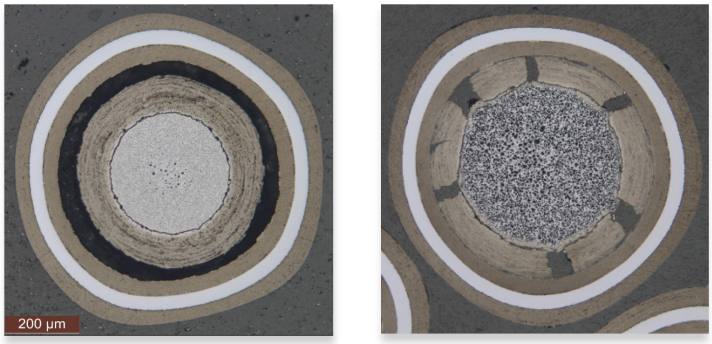
Fuel cycle back-end encompasses spent fuel treatment and disposal, as well as used graphite management. An optimized approach for dealing with the graphite needs to be defined. Although a once-through cycle is envisioned initially, the potential for deep burn of plutonium and minor actinides in a VHTR,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

Irradiation and PIE

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

Post-irradiation examination on the AGR-2 fuel (including both UCO and UO_2 TRISO particles) has continued in the United States. This includes destructive examination of fuel compacts and particles. Up to this time, 11 UCO and 2 UO_2 compacts have been examined, providing information on fission product retention in the particles and compacts during irradiation and detailed microstructural information on the condition of the coating layers and the migration of fission products in the layer (see **Figure VHTR 2**).

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



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

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

Fuel attributes and material properties

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

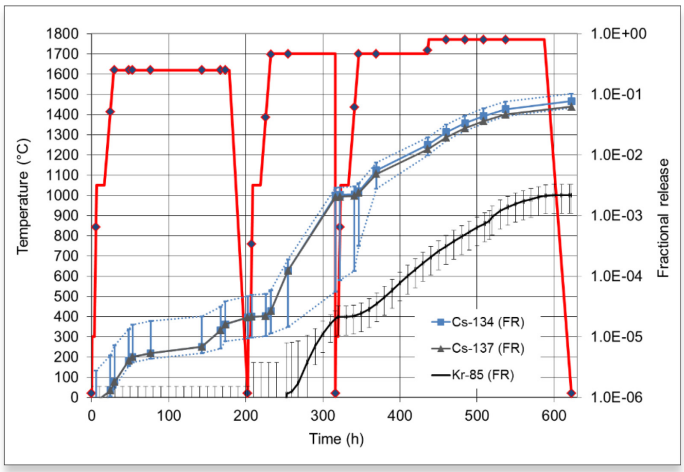
The United States, China, and Korea are completing the final stages of an as-fabricated fuel characterization "round robin" experiment. The work involves performing leach-burn-leach analysis on well-characterized particle specimens to detect defective SiC coatings and complete through-layer coating defects.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

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

Safety

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

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



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

The United States is also performing PIE on the AGR-3/4 irradiation experiment components and heating tests on AGR-3/4 TRISO fuel compacts. These compacts contain about 1 900 TRISO fuel particles, and 20 "designed-to-fail" particles that experience coating failure during the irradiation. Some of these compacts have been re-irradiated in the NRAD reactor to generate short-lived ^{131}I prior to the heating tests. These tests are therefore being used to assess fission product release from exposed kernels.

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

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

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

Enhanced and advanced fuel fabrication

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

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

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

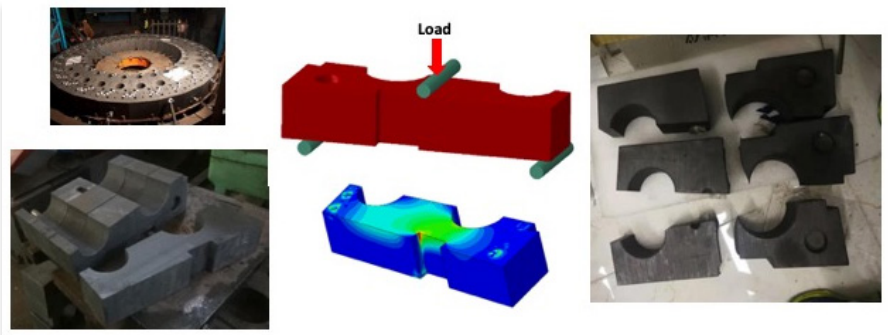
Materials project

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

In 2019, research activities continued focused on near- and medium-term projects needs (i.e. graphite and high-temperature metallic alloys) with limited activities on longer-term activities related to ceramics and composites.

Additional characterization and analysis of selected baseline data and its inherent scatter of candidate grades of graphite was performed by multiple members. Mechanical, physical, and fracture properties behavior were examined for numerous grades. Graphite irradiations and post-irradiation examinations & analysis continued to provide critical data on property changes, while related work on oxidation examined both short-term air and steam ingress, as well as the effects of their chronic exposure on graphite. One area of significant interest among signatories is the validation of the anticipated multi-axial loading response of graphite from dimensional changes and seismic events. A figure illustrating large-scale experiments on graphite blocks to validate design models is shown in **Figure VHTR 4.**

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

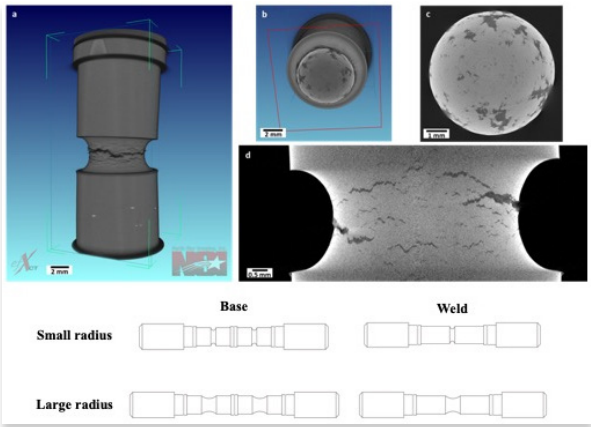


Courtesy of Institute of Nuclear and New Energy Technology.

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

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

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



Courtesy of Idaho National Laboratory.

In the near/medium term, metallic alloys are considered as the main option for control rods and internals in VHTR projects, which target temperatures below about 850°C. However, future projects are considering the use of ceramics and ceramic composites where radiation doses, environmental challenges, or temperatures (up to or beyond 1 000°C) will exceed capabilities of metallic materials. This is especially true for control rods, reactor internals, thermal insulation materials, and fuel cladding. Limited work continues to examine the thermo-mechanical properties of SiC and SiC-SiC composites and oxidation in C-C composites. Studies of fabrication, architecture, and processing on the properties and fracture mechanisms of the composites is being investigated. The results of this work is being actively incorporated into developing testing standards and design codes for composite materials, and to examine irradiation effects on ceramic composites for these types of applications. A significant milestone in this area occurred in 2019 with drafts on all articles related to General Requirements and Design Rules for Ceramic Components of ASME Code Section III Division 5 (Rules for Construction of High Temperature Reactor Components) having been completed and submitted for ballot.

Hydrogen production project

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

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

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

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

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

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

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

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

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

Figure VHTR 7. Systems Integration Laboratory at INL

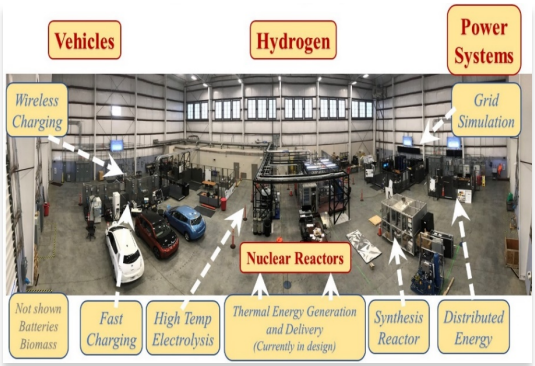


Figure VHTR 8. 25 kW HTSE Test Facility in DETAIL within the INL Energy Systems Laboratory

Computational methods validation and benchmarks Project

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

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

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

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

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

Figure VHTR 9. HTR-10 reactor coupled with RCCS

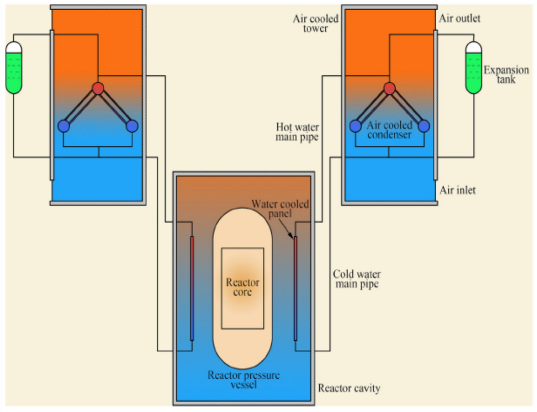
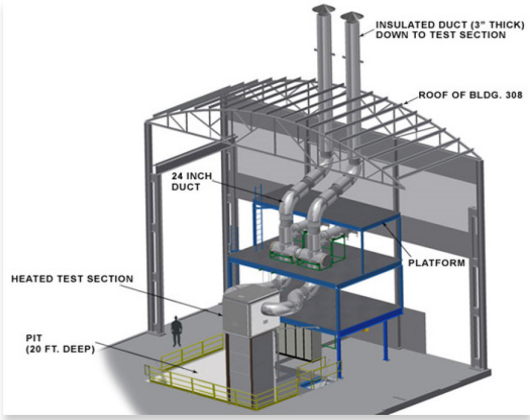


Figure VHTR 10. Natural Circulation Shutdown Heat Removal System at Argonne National Laboratory



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

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

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

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

Michaël Fuetterer

Chair of the VHTR SSC and all Contributors