GEN IV International Forum

Expertise | Collaboration | Excellence

ANNUAL REPORT 2021

www.gen-4.org

Foreword from the Chair

It is a privilege to compose this foreword for the Generation IV International Forum (GIF) 2021 Annual Report, which outlines last year's progress in Generation-IV (Gen-IV) reactor systems' collaboration and developments. GIF completed its second decade as the sole international organization dedicated to collaborative research and development (R&D) on Gen-IV systems. Since 2001, GIF has been promoting international R&D collaboration for six types of Gen-IV reactor systems using sodium, lead, gas, molten salt and supercritical water coolants. These systems follow the common development goals established by GIF: safety and economics, together with sustainability and proliferation resistance and physical protection.

The year 2021 also marked the end of Chair Hideki Kamide's highly successful three-year term. I would like to express my sincere gratitude for the great leadership that Chair Kamide provided during his term. Under his management, we have seen significant progress made within our planned projects even during the difficult times caused by the Coronavirus-19 (COVID-19) pandemic around the world.

During 2021, one of the significant activities and efforts that GIF addressed was substantive engagement with the private sector through a series of workshops. As we work towards commercialization, this engagement will become even more important. We also deepened GIF's role in climate change initiatives, including the Clean Energy Ministerial. Nuclear energy can be a major tool to meet the ambitious initiatives that each country has set. With plans to expand the use of advanced nuclear energy, we have elevated GIF's education and training efforts from a task force to a working group to ensure that our future workforce will be ready to support the vast technical needs of our Gen IV systems.

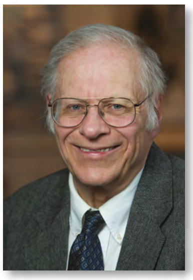
We also made significant strides in advancing the safety framework of Gen-IV systems, including strengthened collaborations with the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency's (NEA) Working Group on the Safety of Advanced Reactors (WGSAR) to ensure consistency of approaches and to avoid duplication of work. A well-coordinated framework will ensure that our Gen-IV systems include safety as part of the design. To streamline the ability to share nuclear technology with nations as they develop their nuclear energy infrastructure, we have made substantial updates to the proliferation resistance and physical protection white papers for the six Gen-IV systems. Further expanding on the potential uses of nuclear energy, we also launched a new initiative on the non-electric applications of nuclear heat. Finally, we completed a much-needed refresh of the GIF brand and communication products.

With continued global leadership and support through the GIF framework, we can realize the goals of ensuring that sustainable nuclear energy is available in the future and that commercial deployment of advanced systems is started in 2030. Looking forward, GIF will advance this vision by continuing to strengthen Gen-IV system features to combat climate change, with a focus on flexible operations and non-electric applications. We will support transition from R&D to demonstration and deployment by ensuring technical readiness, regulatory readiness and improved economics. GIF relevance to industry will be strengthened by furthering industry engagement and the guiding expertise of the GIF Senior Industry Advisory Panel; and looking to the future, we will continue to support the Gen-IV talent pipeline.

As I assume the role of GIF Chair, I look with great optimism at the potential of nuclear energy and its role in addressing our changing energy needs. I hope that this year's annual report will be a great resource in your efforts to coordinate upcoming plans for nuclear energy development.

Alice Caponiti
GIF Chair

A tribute



It was with great sorrow that the United States Department of Energy (DOE) conveyed the passing of Dr James Sienicki, a long-standing member of the Generation IV sodium-cooled fast reactor component design and balance-of-plant (CD&BOP) project; he was part of GIF and CD&BOP from their initiation, and held several terms as Project Management Board Chair, hosting many project meetings. He joined Argonne National Laboratory in August 1976 after receiving his PhD in physics from the University of Illinois. During his years of service, he conducted engineering design and safety analysis research on nuclear reactors, facilities and other engineering systems to support the development of advanced concepts. Jim also participated in the start of Gen-IV collaborations on the lead-cooled fast reactor technology. He was extensively involved in the development of the STAR line of lead-cooled fast reactors and advanced energy conversion with the supercritical carbon dioxide Brayton cycle. In more recent years, Jim was a key contributor to the design and evaluation of the versatile test reactor (VTR) and the assessment of designs for mobile nuclear power plants. Jim was a highly valued contributor to GIF and the nuclear engineering community for over 45 years, and he will truly be missed.

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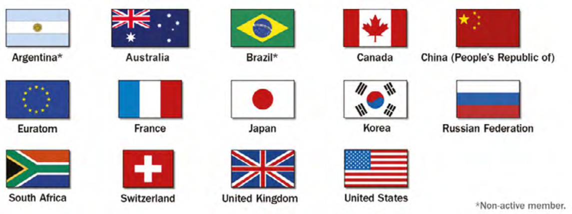
GIF membership, organization and R&D collaboration

In 2021, the structure and organization of the Generation IV International Forum (GIF) remained the same. GIF is composed of 14 member countries, all of which are signatories of its founding document.

GIF membership (2021)



The present chapter will focus only on major changes that occurred in 2021. More detailed information on the GIF membership and organization can be found on the GIF website[1].

GIF organization

GIF global governance in 2021 is summarized below in Figure 1-1.

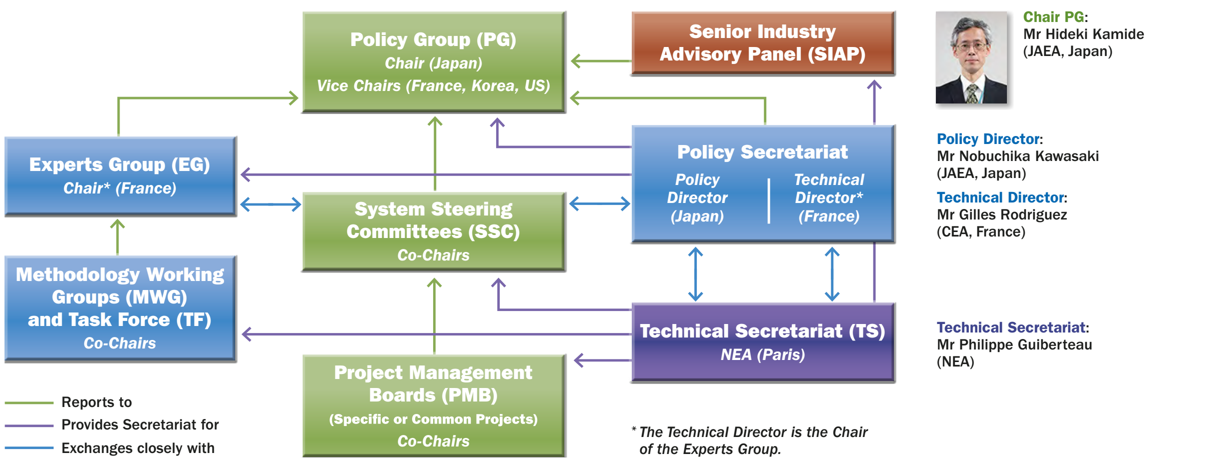
At the end of October 2021, the GIF Policy Group members thanked the outgoing Chair Hideki Kamide from the Japan Atomic Energy Agency (JAEA) for the notable accomplishments of the past three years under his leadership. Alice Caponiti, Deputy Assistant Secretary for Reactor Fleet and Advanced Reactor Deployment in the Office of Nuclear Energy at the United States Department of Energy (DOE) was appointed as the new GIF Chair, and she has proposed a new support structure for the GIF Chair effective 1 January 2022.

Concerning the GIF Technical Secretariat structure, the main changes and objectives from 2021 are:

- Discussion at the highest level of GIF occurred in 2021 about the creation of the molten salt reactor (MSR) system (i.e. the System Research Plan and System Arrangement, Project Plans and Project Arrangement for three new Project Management Boards) from the provisional System Steering Committee (SSC). The final decision was made to maintain the MSR under the current memorandum of understanding (MoU), with a possible enhancement in the organization of activities.
- The possibility of undertaking activities within the GIF framework relative to the non-electric application of nuclear energy heat (NEaNH) was also examined in 2021. A roadmap and a new organization were proposed in 2021, and these were presented to the GIF Experts Group and Policy Group, which decided to create a dedicated task force at the end of October 2021.

With the aim of stabilizing and reinforcing the GIF Technical Secretariat, several changes in staff occurred in 2021; the resulting structure of the GIF Technical Secretariat as of November 2021 is shown in Figure 1-2.

Figure 1-1: GIF Governance in 2021



1. For further details, please visit the GIF website at: www.gen-4.org/gif/jcms/c_59452/governance-stucture.

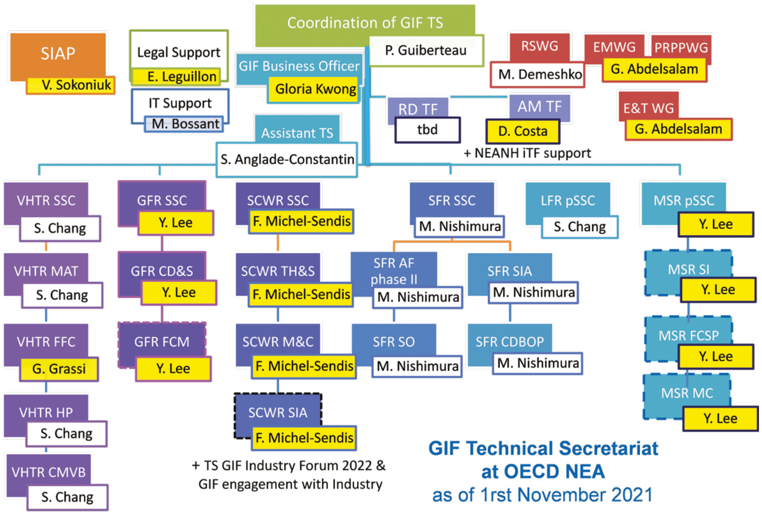


Figure 1-2: Structure of the GIF Technical Secretariat

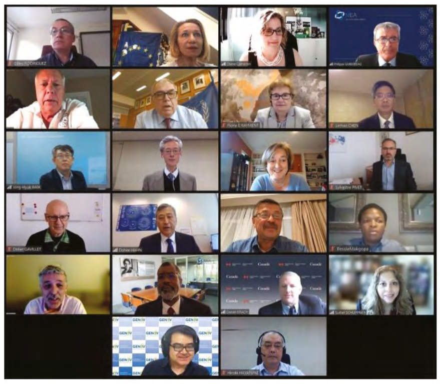


Figure 1-3: GIF Policy Group photo during the virtual Zoom meeting in October 2021

The GIF Technical Secretariat continued in 2021 to improve GIF communication so as to reinforce the GIF position as a leading, collaborative organization at the international level, with technical expertise focused entirely on 4th generation nuclear energy systems, and with the key theme for the coming years of accelerating the readiness of Generation-IV (Gen-IV) systems to meet net-zero goals. To reach these goals, GIF:

- enhanced the hard/soft contents, structure and style of the GIF website;
- disseminated 11 technical reports with new and specific covers and back pages;
- released the "Gifted" Newsletter on a regular basis (five newsletters were released in 2021) with articles from member countries;
- supported the GIF Education and Training Working Group (ETWG) in the organization of the "Pitch your Gen-IV Research" competition involving young researchers, as well as the "GIF Chair webinar" in the framework of the 20th anniversary of GIF, with the participation of the current and former GIF chairs in two panel discussions about "views on GIF progress" and "future prospects towards deploying Gen-IV reactors as advanced nuclear energy systems";
- participated in the 2021 World Nuclear Exhibition (WNE 2021) in Paris with specific GIF materials (e.g. kakemonos and flyers), presenting the GIF Economic Modelling Working Group's *Nuclear Energy: An ESG investable Asset Class* report in a special workshop and staffing a booth where a teaser announcement for the 2022 GIF Industry Forum was made.

The GIF logo for the GIF Industry Forum[2]

The objectives are to accelerate the readiness of Gen-IV systems to meet net-zero goals

Throughout 2021, despite the unprecedented challenges posed by the COVID-19 pandemic, GIF members managed to make progress in their work with a minimum of disruption. The GIF Technical Secretariat continued to provide support – in virtual mode (see Figure 1-3) – to GIF technical bodies in charge of the development of the different systems, to the methodology working groups and to SIAP, as well as to the task forces.

More than ever, we are all looking forward to meeting face to face again, and having the opportunity to have technical exchanges in person.

In conclusion, 2022 perspectives and objectives are to continue to assume and improve the GIF Technical Secretariat despite difficulties related to the COVID-19 situation while:

- stabilizing the organization of the GIF Technical Secretariat and enhancing its work;
- Improving the secretariat links and improving support to the GIF Chair, GIF members and systems, working group, task force and SIAP chairs;
- Developing cross-cutting projects within GIF and with the Nuclear Energy Agency (NEA);
- Improving GIF links with industry and other non-profit international organizations (e.g. the World Nuclear Association [WNA] and the International Atomic Energy Agency [IAEA]), in particular through the organization of the GIF Forum with Industry;
- Continuing to develop GIF communication (technical, external and internal).

Philippe Guiberteau
Head of the GIF Technical Secretariat

2. To download the flyer, please visit www.gen-4.org/gif/jcms/c_196903/download-the-flyer-for-details.

2 GIF outlook and current initiatives

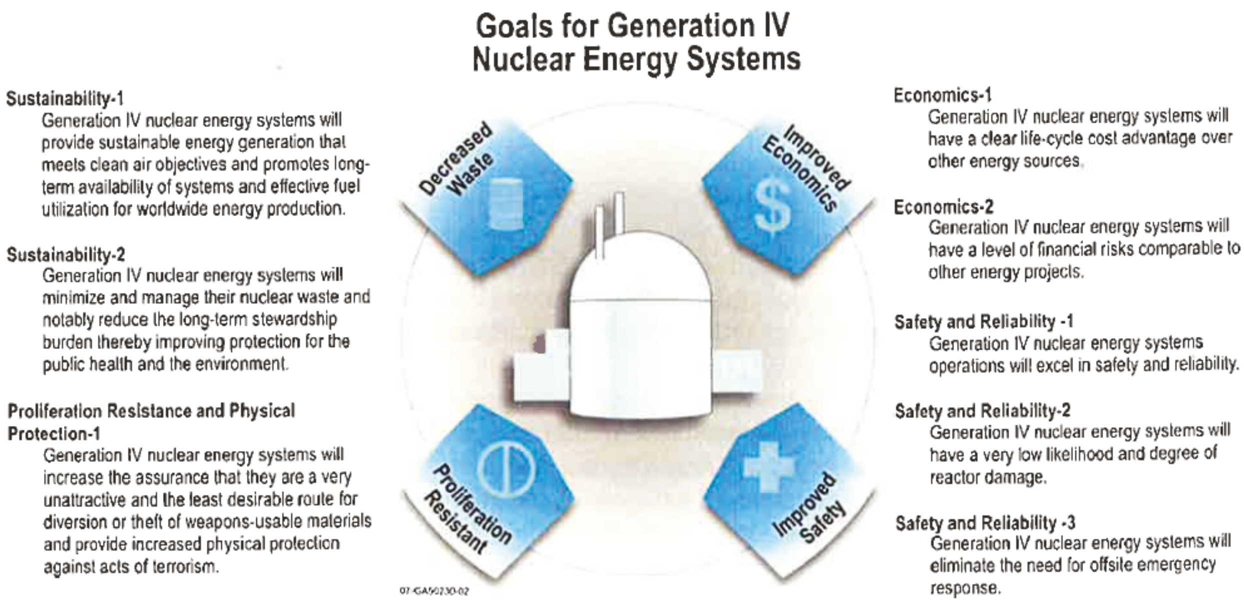
The Generation-IV (Gen-IV) goals originally defined in the Generation IV International Forum (GIF) Charter[1] have continued to motivate research and development (R&D) on advanced reactor technology options and guide GIF collaboration activities. Gen-IV goals are defined in four broad areas:

- Sustainability is the ability to meet present needs without compromising the ability of future generations to meet their own needs. Sustainability requires conserving resources, protecting the environment and preserving capabilities. In the GIF Technology Roadmap (GIF, 2002), sustainability goals are defined with a focus on waste management and resource utilization.
- Economic competitiveness is a requirement of the marketplace and is essential for Generation-IV nuclear energy systems. Future nuclear energy systems should accommodate a range of plant ownership options and anticipate a wider array of potential energy supply roles. Generation-IV nuclear energy systems may be utilized for a broader range of energy products beyond electricity.
- Safety and reliability are essential priorities in the deployment and sustained operation of nuclear energy systems, while competitiveness requires a very high level of reliability and performance. Generation-IV nuclear energy systems reinforce the defense-in-depth approach and utilize innovative features to provide inherent safety (i.e. passive decay heat removal pathways).
- Proliferation resistance and physical protection are also essential priorities for the expanded deployment of nuclear energy systems. In addition to the ready application of international safeguards, Generation-IV advanced reactor technologies promote the integration of safety, security and safeguard requirements into the design of new fuel cycles and reactors.

For these four areas, the eight specific goals utilized in the GIF Roadmap (GIF, 2002) are shown below:

The GIF Roadmap also identified three successive phases for advanced reactor development:

- the viability phase, when basic concepts are tested under relevant conditions and all potential technical showstoppers are identified and resolved;
- the performance phase, when engineering-scale processes, phenomena and materials capabilities are verified and refined under prototypical conditions;



1. For more information on the GIF Charter, please visit www.gen-4.org/gif/jcms/c_40243/gif-charter.

- the demonstration phase, when the detailed design is finalized, and licensing, construction and operation of the system is carried out, with the aim of bringing Gen-IV reactors to commercial deployment.

The current technology status and timeline for the phases were evaluated in 2014, in the GIF Technology Roadmap update (GIF, 2013). Today, several of the six Gen-IV systems are entering the demonstration phase, which presents several challenges and opportunities for the GIF community.

In general, the same infrastructure (both in terms of expertise and facilities) that is needed for R&D on viability and performance remains useful in the demonstration and deployment phases. However, the topics that need to be addressed are identified and prioritized by operational and construction experience. An intimate working relationship with the advanced reactor industry working to license, construct and operate advanced reactors is thus needed to ensure the continued relevance of Gen-IV contributions.

Another challenge relates to systems that are being demonstrated in the near term and utilize low-risk (high technical maturity) design choices. However, developers are often aware beforehand of favorable features that are not yet technically mature, and operational experience will identify specific technology challenges for optimal performance. Therefore, to support future innovation, a robust R&D infrastructure is needed to support technology development, where particular design features or subsystems can be quickly matured through the viability and performance phases (even while first-of-a-kind Gen-IV technology advanced reactors are operating in the demonstration phase). This provision for future refinement may prove to be critical for the widespread deployment of Gen-IV technology that will depend on the reliable, robust and high-performance operations of these advanced reactors.

Reference

GIF (2002), *A Technology Roadmap for Generation IV Nuclear Energy Systems*, Generation IV International Forum, Paris, GIF-002-00.

GIF (2013), *Technology Roadmap Update for Generation IV Nuclear Energy Systems*, Generation IV International Forum, Paris.

Robert Hill
GIF Technical Director

3 System summaries

Gas-cooled fast reactor

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

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

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

Main characteristics of the system

The GFR system features a high-temperature helium-cooled fast spectrum reactor that can be part of a closed fuel cycle. The GFR, cooled with helium, is being proposed as a longer-term alternative to liquid metal cooled fast reactors. The main advantages of GFRs, in addition to enabling the adoption of a closed fuel cycle, are:

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

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

Technical highlights – conceptual design and safety project

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

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

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

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

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

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

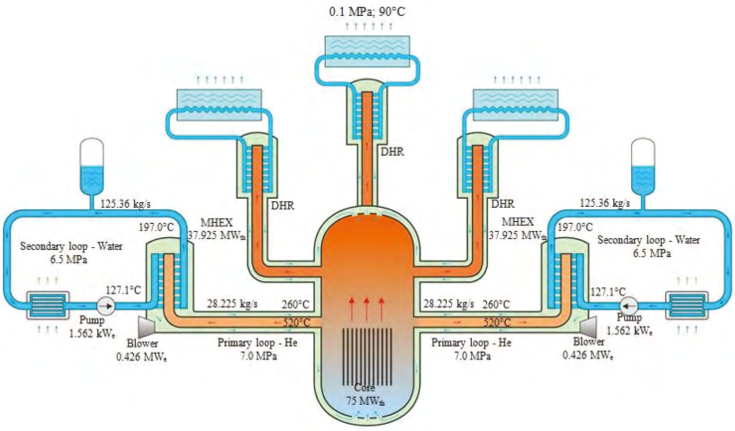


Figure GFR-1: ALLEGRO design overview

a demonstration reactor, hosting GFR technological experiments, but also as a test pad to:

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

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

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

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

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

The SafeG project

The SafeG project has received funding from the Euratom Horizon 2020 Nuclear Fission and Radiation Protection Research program (NFRP-2019-2020-06). The global objective of the SafeG project is to further develop GFR technology and strengthen its safety. The project will support the development of nuclear, low-carbon dioxide (CO_2) electricity and the industrial process heat generation technology through the following main objectives:

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

The main task of the project is to respond to the safety issues of the GFR concept and to introduce the key safety systems of the ALLEGRO reactor. An important part of the design is to acquire new experimental data using recent research from experimental devices and special computational programs to carry out safety analyses and the study of relevant physical phenomena. The SafeG project takes into account the most urgent questions and open issues concerning the GFR technology and the ALLEGRO demonstrator.

Technical highlights – fuel, core materials and fuel cycle project

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

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

The candidate fuel types already identified are:

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

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

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

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

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

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

The ion-irradiation effect on SiC composites will be investigated in order to evaluate the importance of the significant volume change observed for hydrogen (Hi)-Nicalon type-S fiber and C fiber coating. High-dose ion irradiation will be carried out with various temperature ranges, including GFR operating temperatures for SiC composites. The high-dose irradiation effect on SiC composites will be examined. The boron nitride (BN) particle dispersed SiC is candidate matrix material for the high-dose irradiation tolerant SiC composites. The BN particle dispersed SiC was irradiated with the reference chemical vapor deposition SiC at 800°C and up to 10 displacements per atom (dpa) under the SafeG project at the Dual-Beam Facility for Energy Science and Technology (DuET), Kyoto University. Significant differences in swelling were not observed. Regarding the behavior of the SiC brazing agent under irradiation, results of the CROCUS irradiation, performed in the OSIRIS MTR (CEA), have been shared. Several samples consisting of mini composites with brazed joint have been irradiated at dose levels ranging from 1 to 3 dpa. Some of them have been subjected to mechanical solicitation up to 150 MPa. Overall, the braze behaves well under irradiation. No creep has been observed on the samples subject to 150 MPa.

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

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

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

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

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

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

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

Lead-cooled fast reactor

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

Main characteristics of the system

Gen-IV LFR concepts include three reference systems: 1) a large system rated at 600 megawatts electric (MWe) – the European lead fast reactor (ELFR), intended for central station power generation; 2) a 300 MWe system of intermediate size – the Russian BREST-OD-300; and 3) a small, transportable system of 10-100 MWe in size – the US small secure transportable autonomous reactor (SSTAR), which features a very long core life (see Figure LFR-1). The expected secondary cycle efficiency of each LFR system is at or above 42%. GIF-LFR systems thus cover the full range of power levels from small and intermediate to large sizes. Important synergies exist among the different reference systems, with one of the key elements of LFR development being the co-ordination of efforts carried out among participating countries.

R&D objectives

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

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

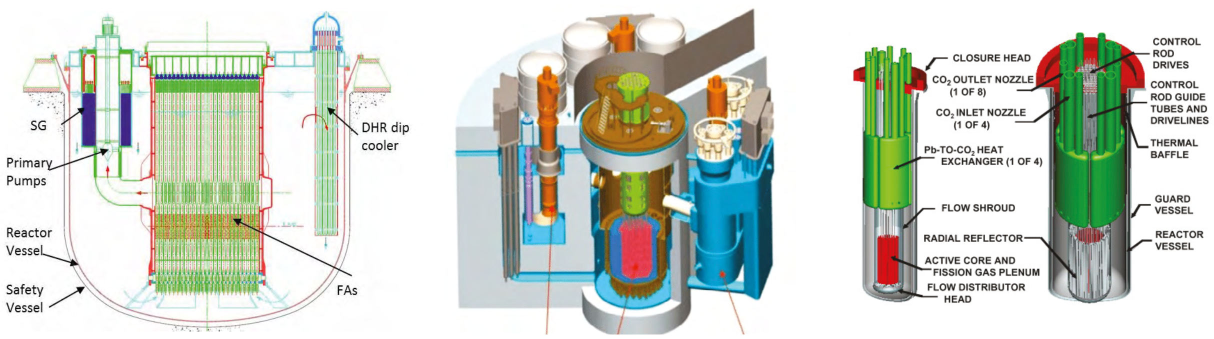
Parameters	ELFR	BREST	SSTAR
Core power (MWt)	1 500	700	45
Electrical power (MWe)	600	300	20
Primary system type	Pool	Pool	Pool
Core inlet T (°C)	400	420	420
Core outlet T (°C)	480	540	567
Secondary cycle	Superheated steam	Superheated steam	Supercritical CO_2
Net efficiency (%)	42	42	44
Turbine inlet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

Technical highlights – provisional System Steering Committee activities

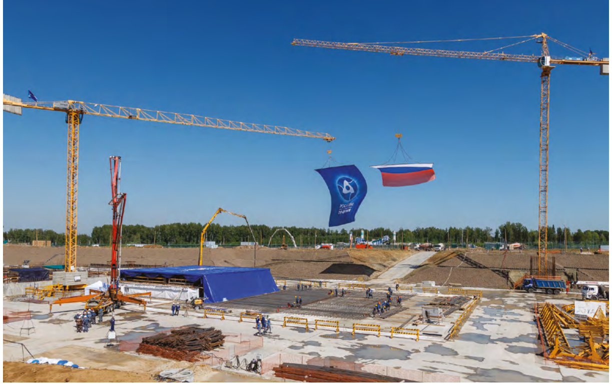
The pSSC has been very active on the following main lines of work in 2021:

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

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

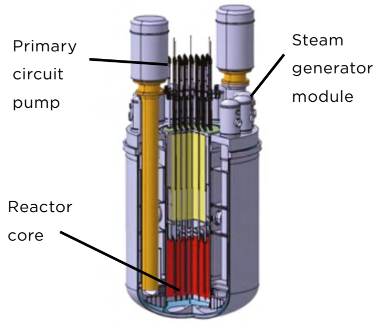


Source: Alemberti, A. et al. (2018).



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

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



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

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

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

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

National LFR demonstration and development highlights

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

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

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

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

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

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

The main activities in Europe related to liquid metal technologies are centered on three main projects: 1) the development of the Multi-purpose hybrid research reactor for high-tech applications (MYRRHA) research infrastructure, which is being carried out by SCK CEN in Mol (Belgium) and is aiming at the demonstration of an accelerator-driven system technology, also supporting the development of fast-neutron spectrum Generation-IV systems; 2) R&D activities for the construction of an LFR demonstrator in Romania, i.e. the Advanced Lead Fast Reactor European Demonstrator (ALFRED) project (see Figure LFR-4); and 3) R&D activities carried out in the United Kingdom in collaboration with several EU organizations in the framework of the Advanced Modular Reactor (AMR) Program (Phase 2), which supports the development of the Westinghouse LFR concept.

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

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

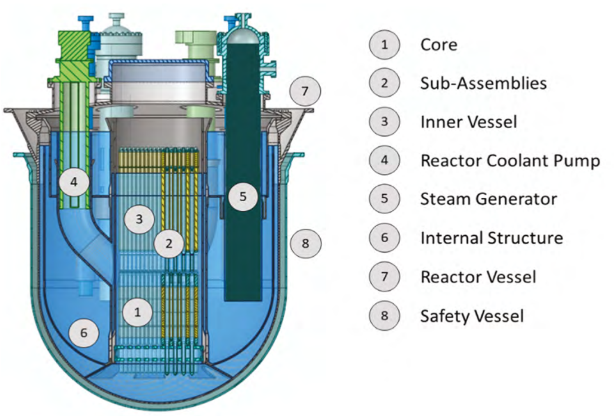
In the United Kingdom, the AMR Program is ongoing. It is funded by the UK Department of Business, Energy and Industrial Strategy and supports LFR development with GBP 10 million (British pounds). The program, led by Westinghouse, is supported by Ansaldo Nucleare (Italy), the Italian National Agency for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

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

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

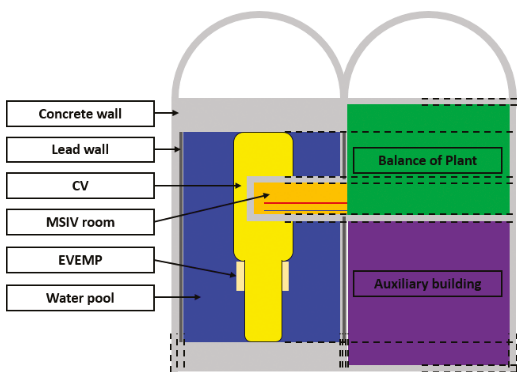
- A project led by the Massachusetts Institute of Technology (MIT) in the area of corrosion/irradiation testing in lead and LBE. The project seeks to investigate the "radiation decelerated corrosion hypothesis", relying on simultaneous exposure.

Figure LFR-4. ALFRED primary system configuration

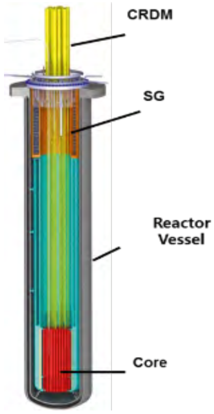


Source: Alemberti, A. et al. (2020); Frignani, M. et al. (2019).

Figure LFR-5. General layout of MicroURANUS 2.5



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



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

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

Source: IAEA LMFNS database.

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

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

In the industrial sector, ongoing LFR reactor initiatives include the continuing initiative of the Westinghouse Corporation to develop a new advanced LFR system (Westinghouse-LFR) and the efforts of Hydromine, Inc. to continue development of the 200 MWe LFR identified as LFR-AS-200 (i.e. amphora shaped), as well as several microreactor spin-off concepts identified as the LFR-TL-X series (where T refers to transportability, L refers to the long-lived core, and X is a variable identifying power options ranging from 5 to 60 MWe). During 2021, the Hydromine design activities were transferred to a newly incorporated European entity, Newcleo, with activities in Italy and the United Kingdom.

In China, the China lead-based mini reactor (CLEAR-M) project, with the 10 MW-grade CLEAR-M10 as a representative concept for a small modular energy supply system, has been launched (see Figure LFR-6). The main purpose of the project is to provide electricity as a flexible power system for wide application, for example in the case of islands, remote districts or industrial parks. In order to support CLEAR projects, as well as to validate and test the key components and integrated operating technology of the lead-based reactor, a multi-functional lead-bismuth experiment loop platform (i.e. KYLIN-II) was built and has operated for more than 30 000 h. Various tests have been conducted, including corrosion tests, LBE thermal-hydraulic experiments and prototype component proof tests. In addition, three integrated test facilities have been built and were commissioned in 2017, leading to the lead-based engineering validation reactor CLEAR-S, the lead-based zero power critical/subcritical reactor CLEAR-0, coupled with the High Intensity D-T fusion Neutron Generator (HINEG) for reactor nuclear design validation, as well as the lead-based virtual reactor, CLEAR-V.

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

References

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

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

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

Molten salt reactor

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

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

Main characteristics of the system

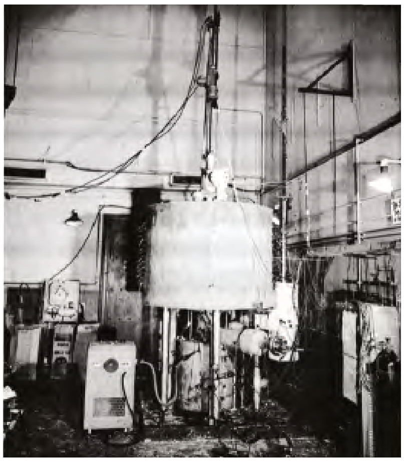
Liquid-fuel MSRs are a type of nuclear fission reactor in which a halide salt serves as the nuclear fuel and may also serve as the coolant. In solid-fuel molten salt reactors, the halide salt serves as the coolant for solid phase nuclear fuel. MSRs were originally conceived in the 1940s. The Oak Ridge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

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

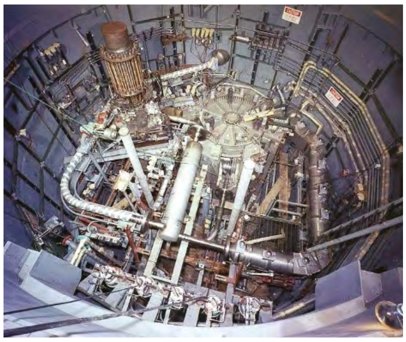
Figure MSR-1: Photos of MSR-related test facilities and reactor at ORNL

ORR in-pile fuel salt loop

ART 650°C critical assembly

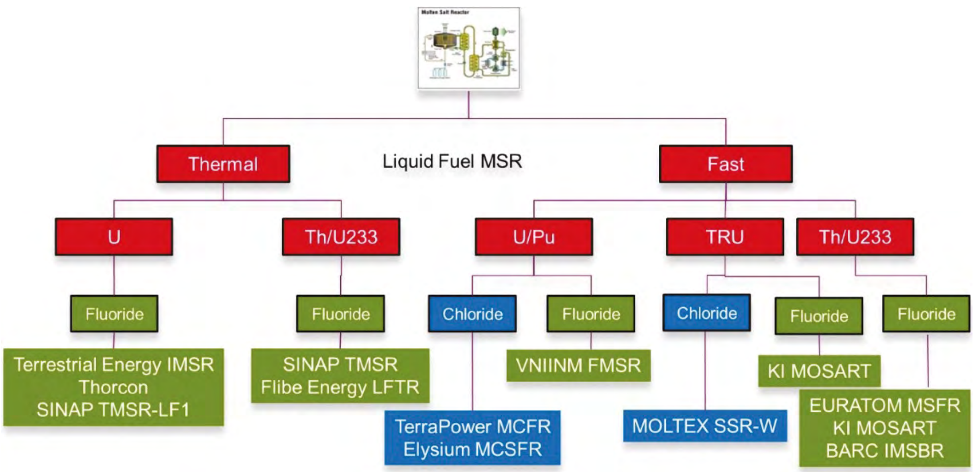


MSRE reactor



Source: ORNL.

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



Source: 39th GIF Experts Group Meeting.

Source: SALIENT-03 Project, JRC.

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

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

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

R&D objectives

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

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

National MSR demonstration and development highlights

On 1 October 2019, the Euratom-funded severe accident modeling and safety assessment for fluid-fuel energy reactors (SAMOSAFER) project started with the aim of developing new simulation models and tools, and designing new safety barriers for the MSR. The goal of this project is to develop and demonstrate new safety barriers for more controlled behavior of MSRs in severe accidents, based on new simulation models and tools validated by experiments. The overall objective is to ensure that the MSR can comply with all expected regulations in 30 years' time. SAMOSAFER is coordinated by TU Delft and will run until 2023.

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

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

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

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

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

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

In Russia, Rosatom continued preliminary MSR design development for:

- a 10 MWt test and large-power 2.4 gigawatt thermal (GWt) units with homogeneous core;
- its fuel salt clean-up unit at the Mining and Chemical Combine site (Zheleznogorsk) in order to demonstrate control of the reactor and fuel salt management with different long-lived actinide loadings, drain-out, shut down, etc.

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

- development and demonstration of key technological solutions for Li,Be,An/F and Li,Na,K,An/F MSRs with circulating fuel for the transmutation of long-lived actinides from used LWR fuel;
- development of a preliminary design for the Li,Be,An/F MSR test and required materials to obtain a license for its placement.

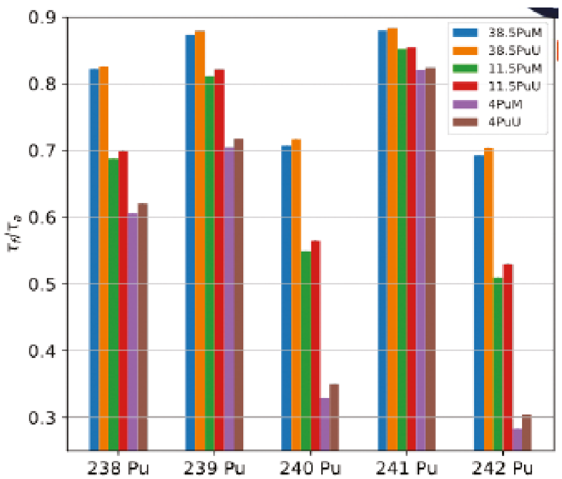
The main R&D efforts in 2021 were focused on the following topics:

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

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

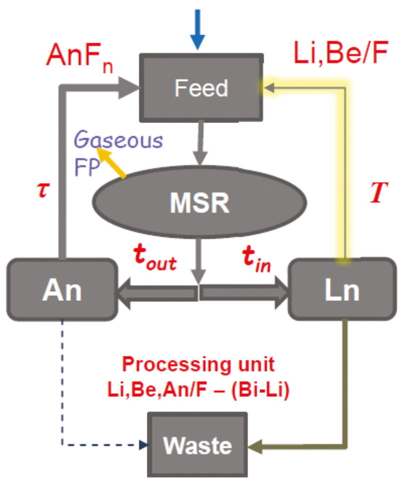
The Abilene Christian University-led effort to construct a university research reactor continues to make progress.[2] Also, a DOE-sponsored integrated research project to design, construct and operate a near core molten salt test loop at the MIT Research Reactor (MITRR) continues to make anticipated

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



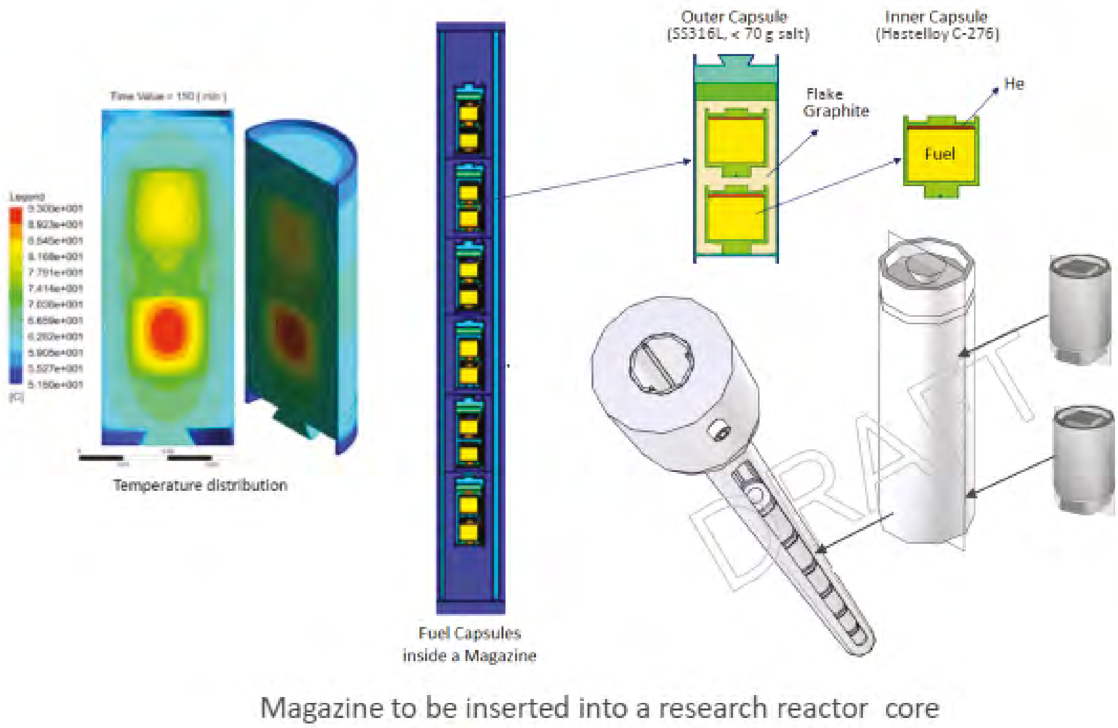
Source: Czech national program on MSR technology development.

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



Source: 31st MSR pSSC meeting, CEA.

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



Source: Laboratory of Subatomic Physics & Cosmology, CNRS.

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

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

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

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

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

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

3. See: https://neup.inl.gov/FY%202020%20CINR%20Abstracts/IRP-20-22026.
4. https://arpa-e.energy.gov/technologies/projects/generation-critical-irradiation-data-enable-digital-twinning-molten-salt.
5. See: https://sc.edu/study/colleges_schools/engineering_and_computing/research/research_centers_and_institutes/general_atomics_center/molten_salt_working_group/index.php.

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

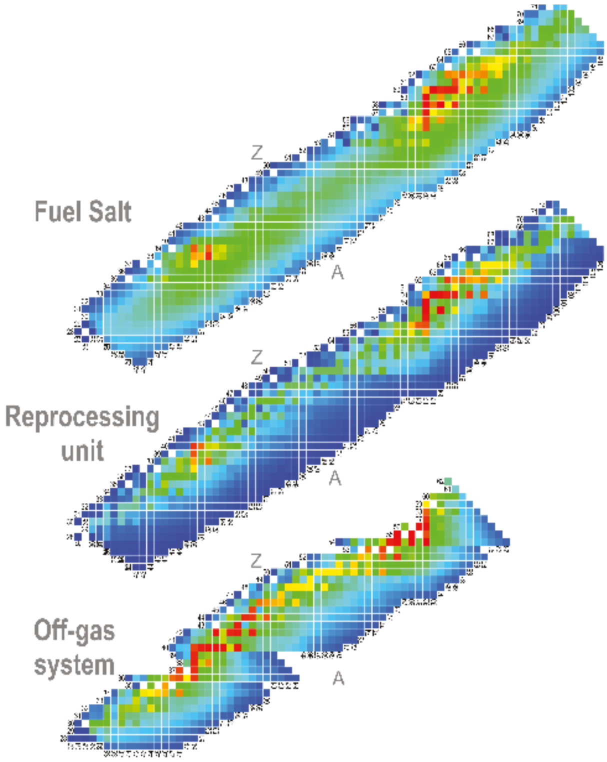
The system behavior study with Open-FOAM based solver ATARI was finalized in 2021 (De Oliveira, 2021). It focused on the assessment of the freezing phenomena in printed circuit heat exchangers and the conceptual design of an MCFR core with tube-in-tube and baffles options.

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

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

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

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



Source: Rosatom Project, Kurchatov Institute.

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

Figure MSR-8: FLiNaK test loop

Source: Sukegawa Electric.

Figure MSR-9: Freeze-valve experiment

Source: Aji.

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

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

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

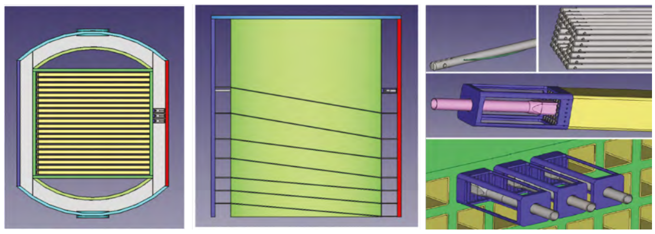
6. FLiNak is a salt mixture of lithium fluoride, sodium fluoride and potassium fluoride.

Super-critical water reactor

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

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

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



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

Main characteristics of the system

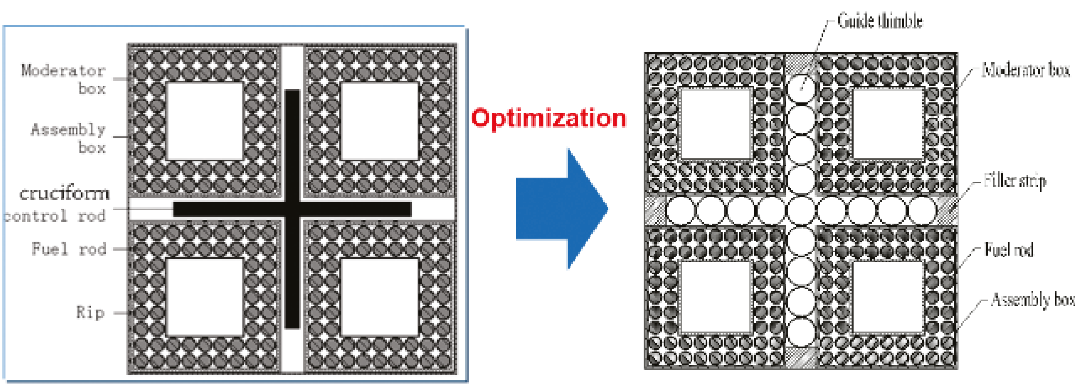
The SCWR is a high temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point (374°C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: pressure-vessel concepts proposed first by Japan and more recently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

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

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

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

Figure SCWR-2: FA optimization design of SCWR 1000



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

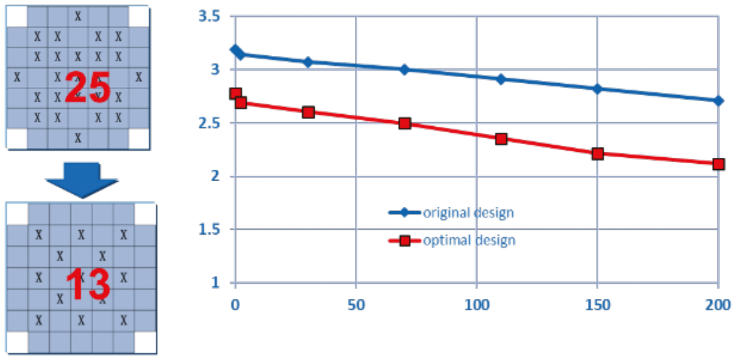
Technical highlights – system integration and assessment project

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

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

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

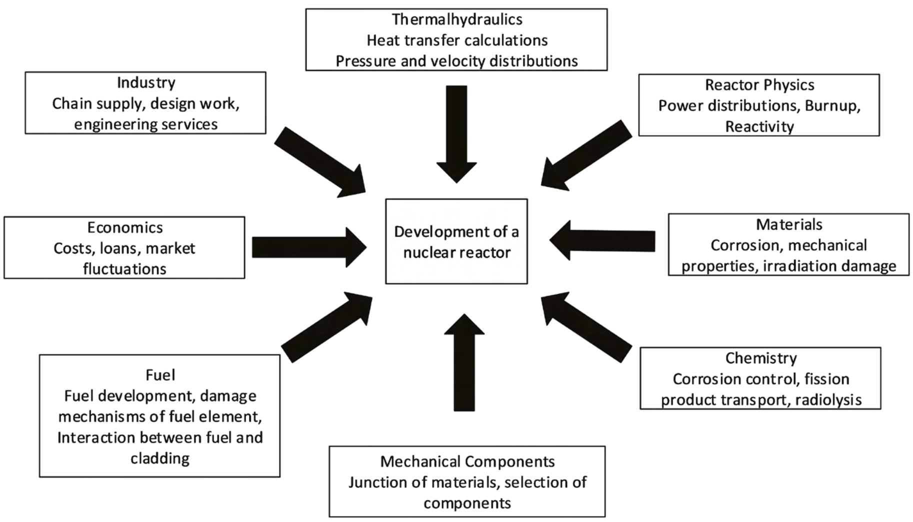
Figure SCWR-3: Optimization design of CSR150



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

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

Figure SCWR-4: Nuclear reactor design interrelation



Source: Canadian Nuclear Laboratories project on SCWRs.

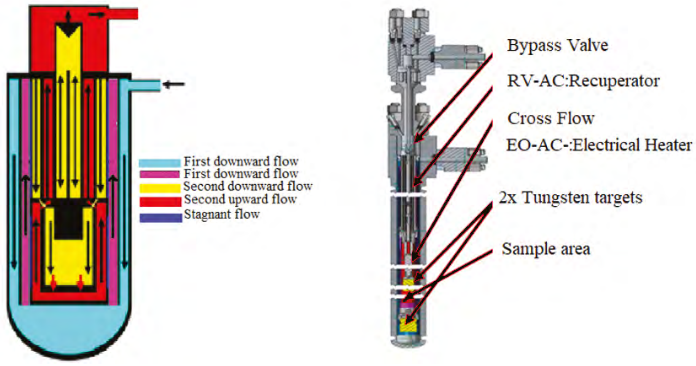
the radiation safety of the turbine plant at the VVER-1200 level; 3) the fast-resonant neutron spectrum in the reactor core provides good neutron-thermal hydraulics stability; and 4) the reactor installation has a compact layout for the primary equipment, and the high quality of the secondary SCW steam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

Technical highlights – thermal hydraulics and safety project

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

Using an algebraic heat flux model developed in the STAR-CCM+ code, the University of Pisa has in recent years been analyzing via Reynolds-averaged Navier-Stokes (RANS) model CO_2 data, water data and other data produced by several researchers. The RANS model was assessed and improved through a variety of experimental data, obtaining good results in comparison with experimental data. Based on these results and on data provided through direct numerical simulation (DNS) studies, the subject of a fluid-to-fluid similarity theory for heat transfer at

Figure SCWR-5: Active channel thermocouples map

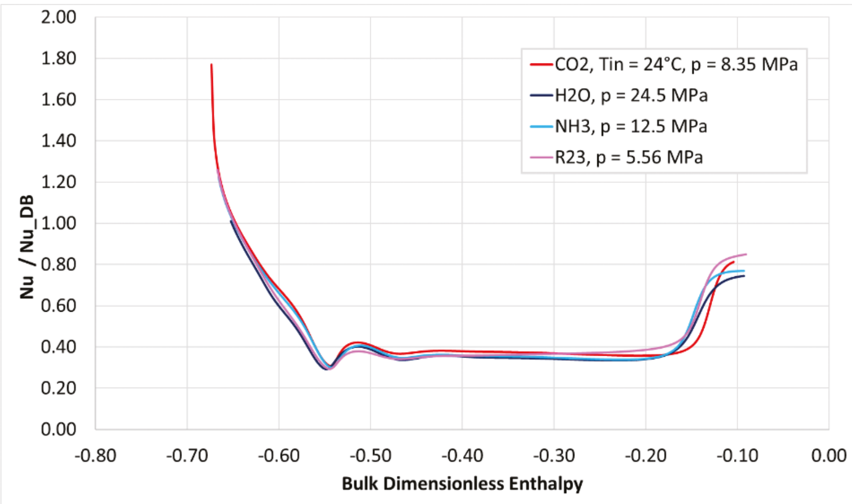


Source: SCWR SA Original work.

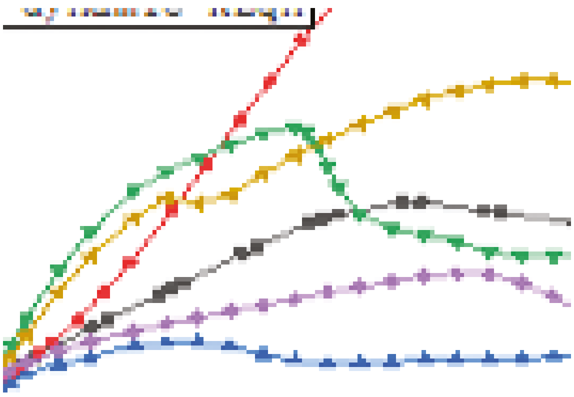
supercritical pressure was further developed. Figure SCWR-6 shows one of the cases addressed from Kline's database (i.e. the trend of the ratio between the computed Nusselt number for four different fluids) and one predicted by the classical Dittus-Boelter correlation.

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

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

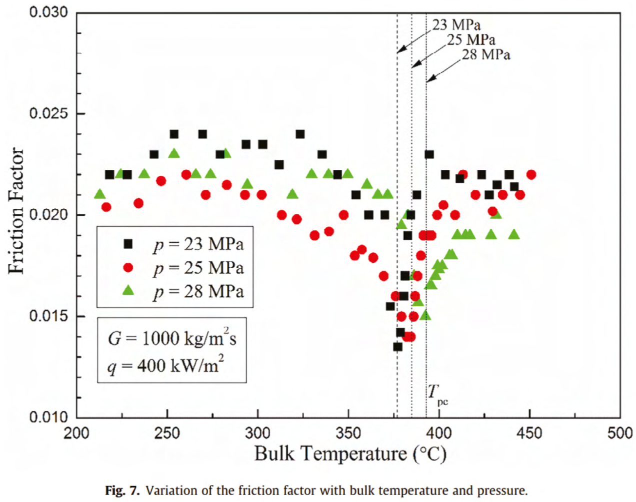


Source: Sara Kassem, Andrea Pucciarelli, Walter Ambrosini, Insight into a fluid-to-fluid similarity theory for heat transfer at supercritical pressure: Results and perspectives, International Journal of Heat and Mass Transfer 168 (2021) 120813.



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

Figure SCWR-7: Wall temperature distribution comparison



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

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

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

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

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

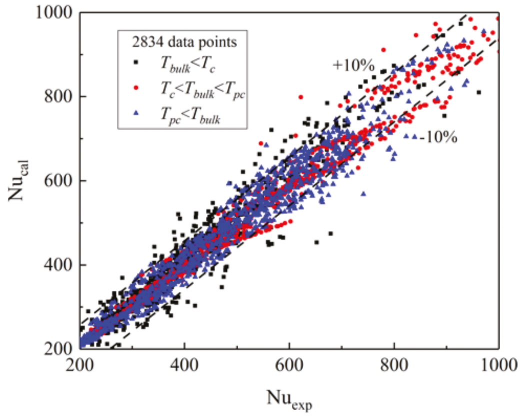
XJTU developed the SCTRAN V2.0 by introducing the two-phase flow model. A wide range of heat transfer models, from subcritical to supercritical, are being realized in this tool. The capabilities of the developed SCTRAN have been verified by calculating the rapid blowdown phenomenon and heat transfer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

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

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

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

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



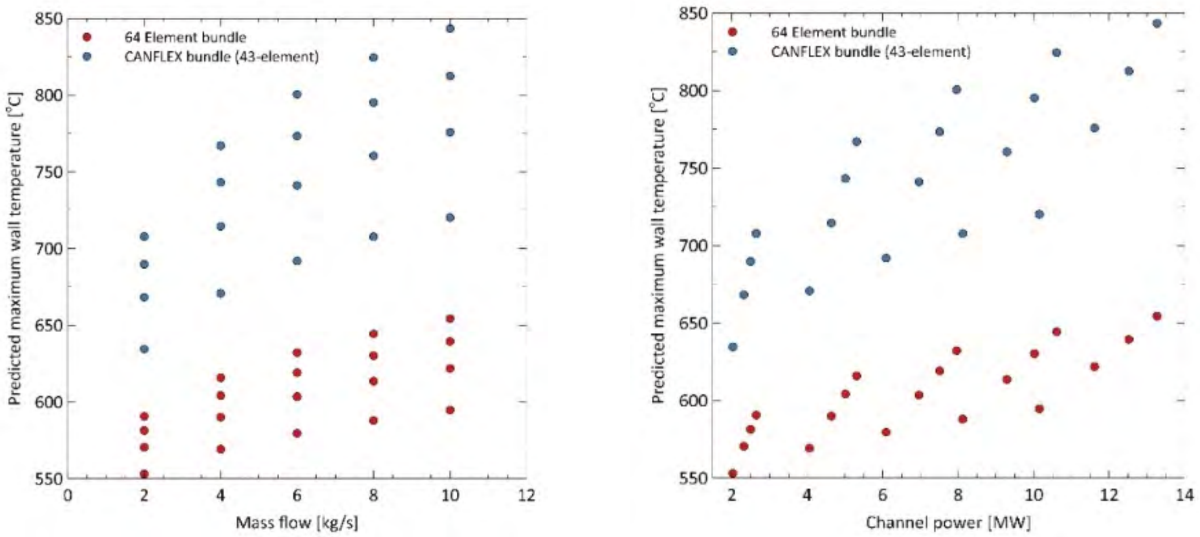
Source: Presented at the TH&S SCWR PMB meeting.

Technical highlights – materials and chemistry project

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

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

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



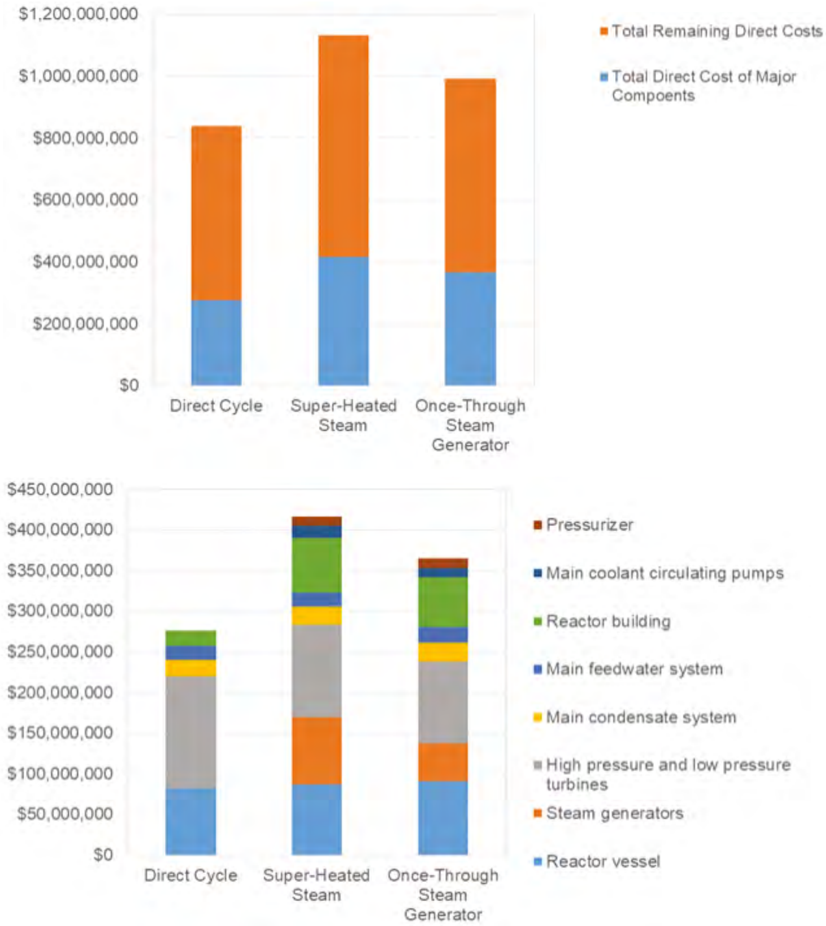
Note: The inlet coolant temperature ranges from 350°C to 380°C with an increase of 10°C. The outlet coolant temperature is fixed at 400°C.

Source: Canadian Nuclear Laboratories project on SCWRs.

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

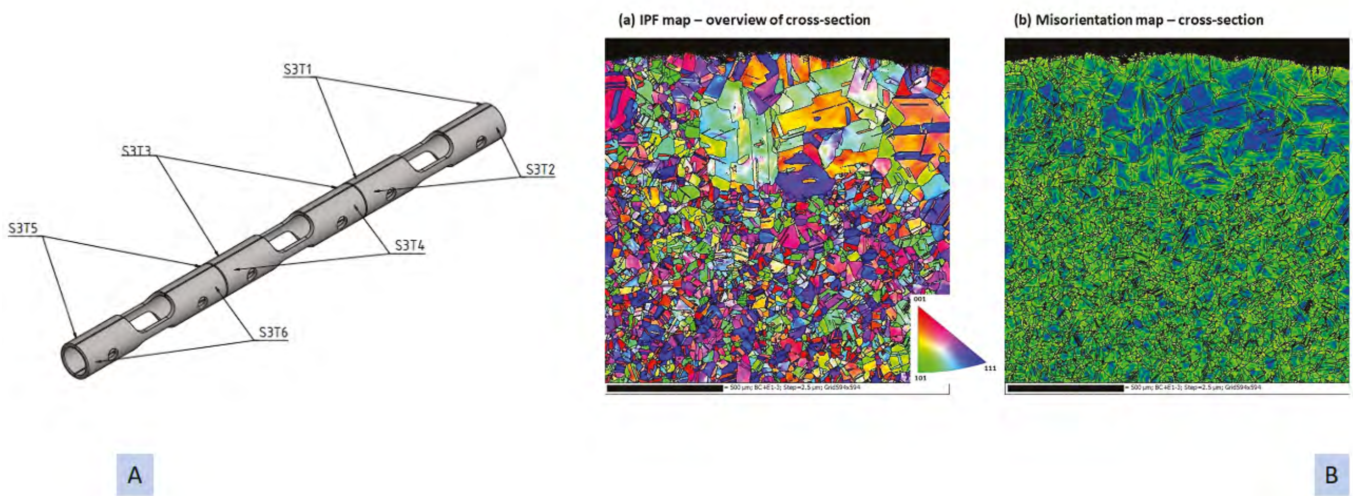
In China, materials and chemistry activities focused on the design and preparation of novel candidate materials for the SCWR fuel cladding tube. Alumina-forming austenitic (AFA) and oxide dispersion-strengthened (ODS) steels have been recognized as potential candidates for SCWR applications. Based on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

Figure SCWR-11: Economic analysis of major components



Source: Canadian Nuclear Laboratories project on SCWRs.

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



Source: SCWR SA original work.

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

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

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

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

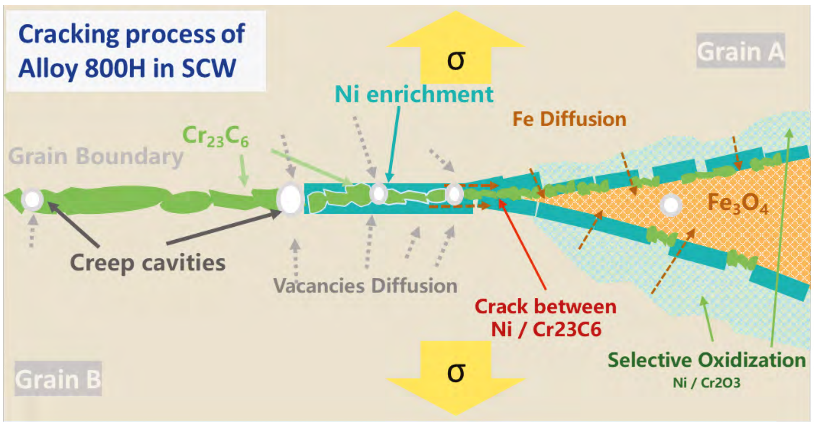
Corrosion studies were conducted on potential fuel cladding materials for small SCWR applications. Figure SCWR-14 shows the autoclave oxidation results from three groups of materials (Cr-coated Zr2.5Nb, Cr coated ZrCrFe and FeCrAl) tested at the operating conditions of scaled-down SCWRs. The operating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

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

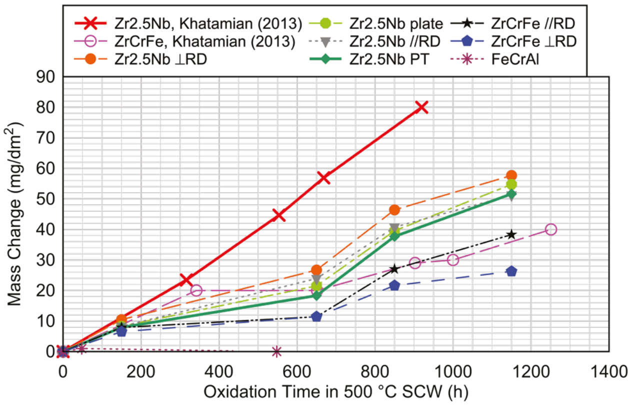
Fuel cladding material	Zr-2.5Nb + 10 μm Cr	Alloy 800HT	Type 310 stainless steel
5-year dpa	5.52	5.92	5.94
5-year He generation	0	62.2	39.7

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

Figure SCWR-13: Cracking mechanisms of alloy 800H

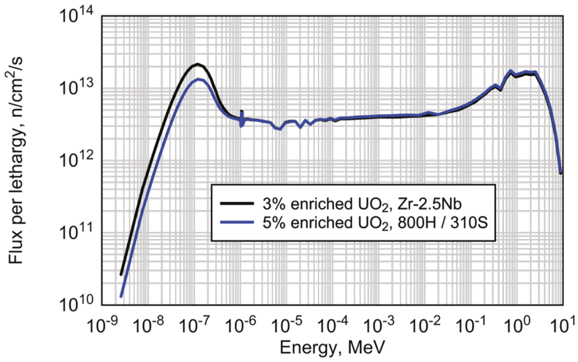


Source: SCWR SA original work.



Source: SCWR SA original work.

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



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

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

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

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

Sodium-cooled fast reactor

The System Arrangement for Gen-IV international R&D collaboration on the sodium-cooled fast reactor (SFR) nuclear energy system became effective in 2006 and was extended for a period of ten years in 2016. Several new members were added to the original agreement and the United Kingdom was welcomed to the System Arrangement in 2019. The present signatories are: the CEA, France; the DOE, United States; the JRC, Euratom; the Japan Atomic Energy Agency (JAEA), Japan; the Ministry of Science and Information and Communication and Technology (ICT), Korea; the China National Nuclear Corporation (CNNC), China; Rosatom, Russia; and the Department for Business, Energy and Industrial Strategy, the United Kingdom. Four technical projects have been established for GIF collaborations:

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

Main characteristics of the system

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

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

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

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

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

Technical highlights – system integration and assessment project

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

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

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

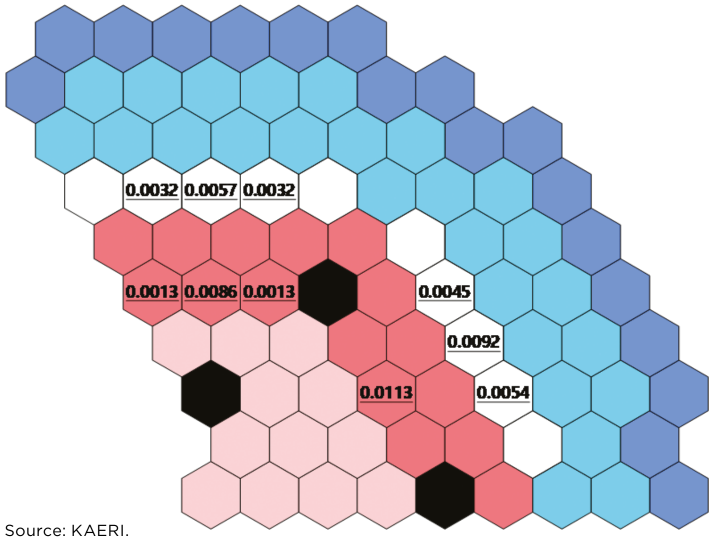
Figure SFR-1: ASTRID reactor view

Source: CEA.

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

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

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



Source: KAERI.

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

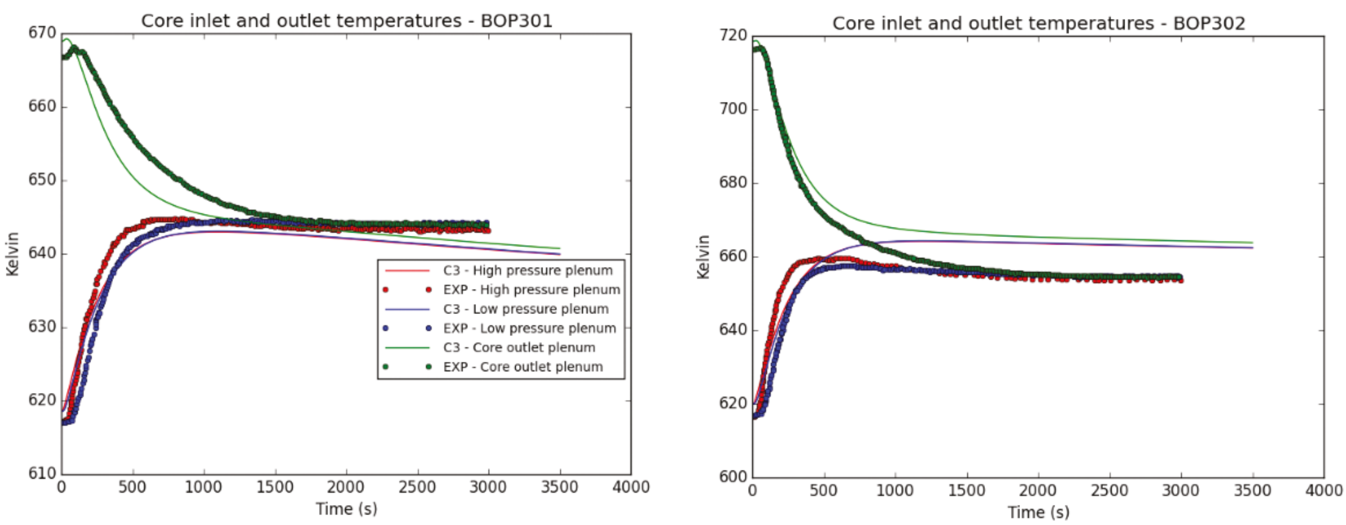
Technical highlights – safety and operations project

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

The project is also pursuing a "common" project that consists of two benchmark analyses (the EBR-II test and Phenix dissymmetric tests), which started from the last quarter of 2019. The first phase of the benchmark analysis is a "blind phase", which will take two years to complete. Argonne National Laboratory (ANL), the JAEA and KAERI completed the blind phase of the EBR-II test study at the end of 2020. In 2021, the CEA started the "open phase".

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

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



Source: CEA.

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

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

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

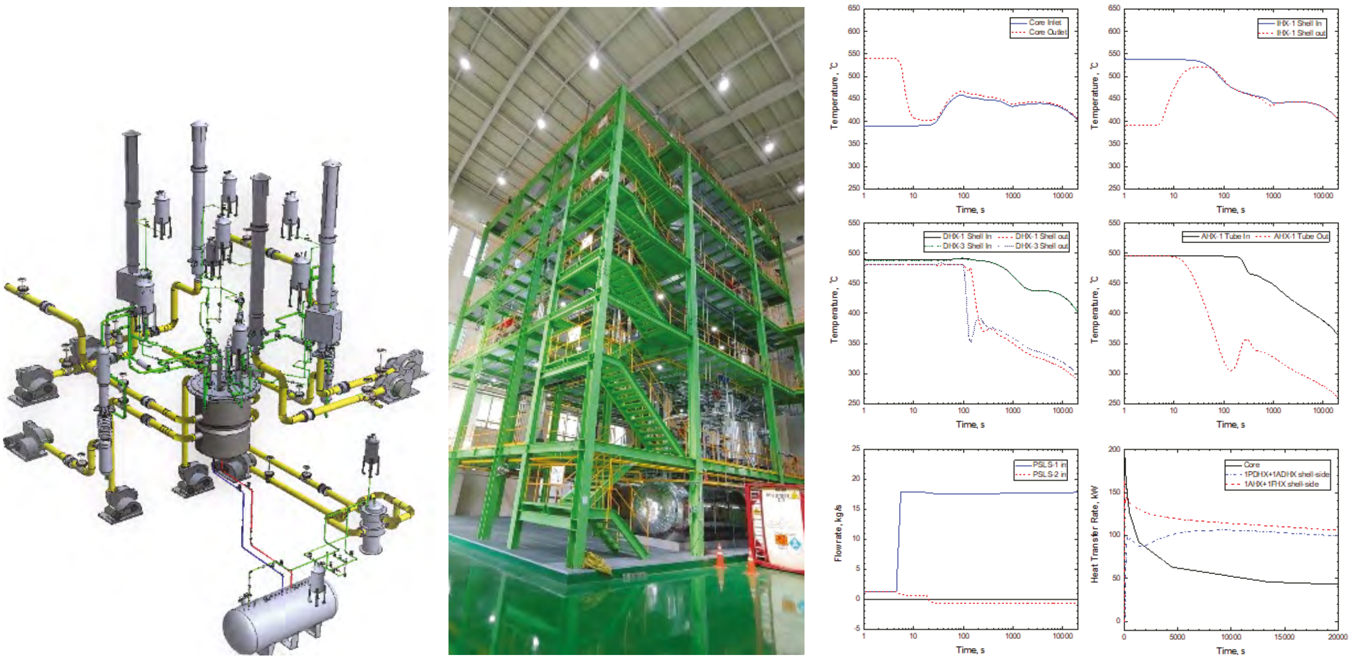
The JAEA also presented the results of the assessment during a postulated, unprotected loss-of-flow (LOF) accident for small-scale SFRs to show that the decay heat generated from the relocated fuels would be stably removed in the post-accident-material-relocation/post-accident-heat-removal phase, where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

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

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

Figure SFR-4: The JIMEC-2 test setup

Source: JRC.



Source: KAERI.

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

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

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

Innovative design and safety systems

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

Technical highlights – advanced fuels project

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

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

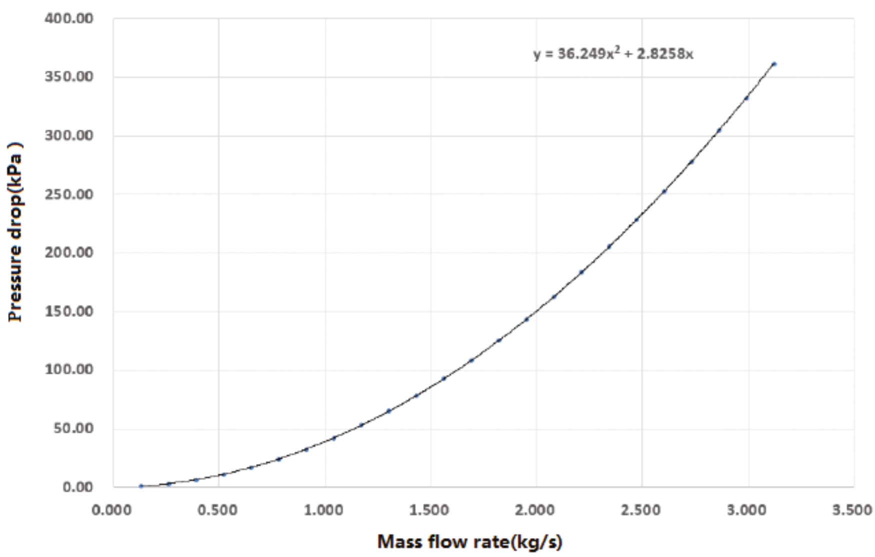
The China Institute of Atomic Energy (CIAE) is preparing to undertake some irradiation tests. It has finished the fabrication of dummy irradiation assemblies and out-of-pile hydraulic tests (hydraulic characteristic experiments). In 2021, out-of-pile hydraulic and mechanical supplementary tests were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

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

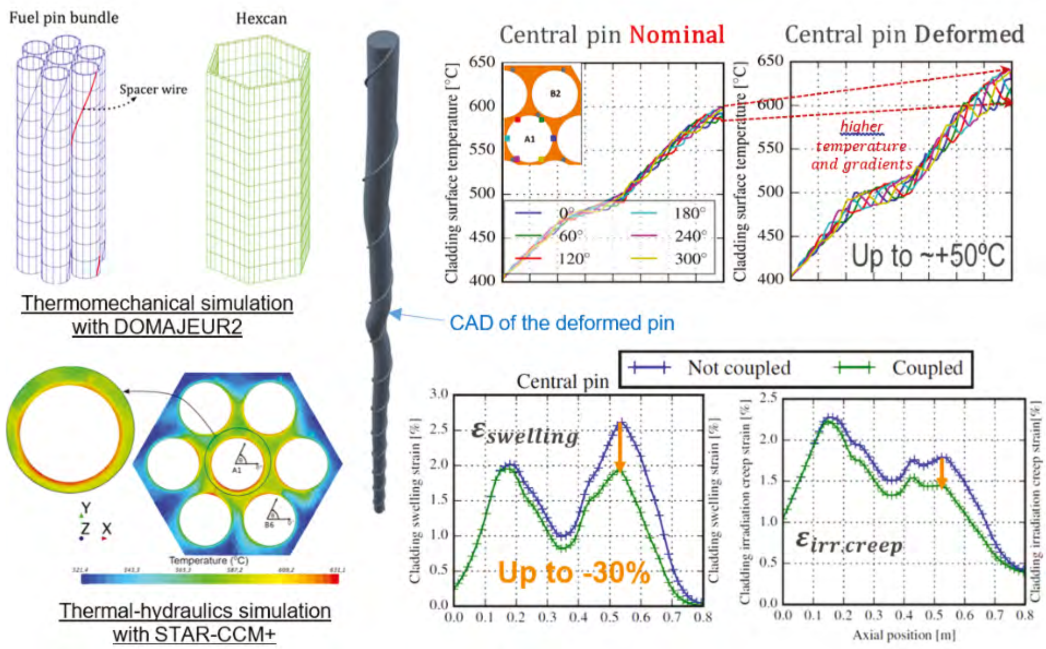
The Euratom Joint Research Centre recently proposed an innovative approach for the safe and secure synthesis of oxide based nuclear fuels. The new method is based on the hydrothermal decomposition of oxalates at low temperature, and it significantly reduces the required energy and number of process steps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

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

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



Source: CIAE.



Source: Acosta et al., 2019 and 2021.

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

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

The CIAE will conduct a program to undertake CN-1515 and CN-FMS material irradiation tests in the CEFR in the coming years. The R&D and fabrication of CN-1515 and CN-FMS has been completed. The design of the irradiation rig and the fabrication of irradiation assemblies has also finished (Figure SFR-8). The mechanical properties of CN-1515 and CN-FMS have been tested and have been used to evaluate the irradiation assemblies.

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

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

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

The component design and balance-of-plant (CD&BOP) project includes the development of advanced energy conversion systems (ECS) to improve thermal efficiency and reduce secondary system capital costs. The project also includes R&D on advances in sodium ISI&R technologies, small sodium leak consequences, new sodium testing capabilities and decommissioning. The main activities in ECS include: 1) the development of advanced, high-reliability steam generators and related instrumentation; 2) the development of advanced ECS based on a Brayton cycle with supercritical CO_2 or nitrogen as the working fluid; and 3) the study of decommissioning of sodium facilities for guidance on future plant design.

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

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

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

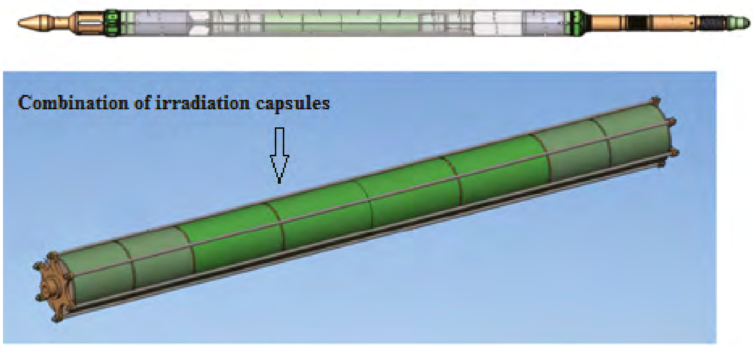
The CEA worked on the SOVEXP model, it was developed to represent the sodium-water reaction (SWR) as a fast dynamic process that generates pressure waves carrying part of the reaction energy. This model applies to configurations that can be explosive, especially when water is in excess compared to the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

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

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

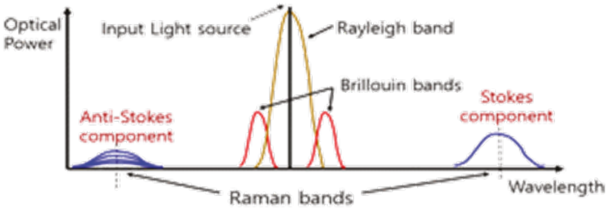
Figure SFR-8: Cladding tubes of CN-1515 and the irradiation assembly of CN-1515 cladding tubes and CN-FMS hexagonal ducts

a) Cladding tubes of CN-1515.

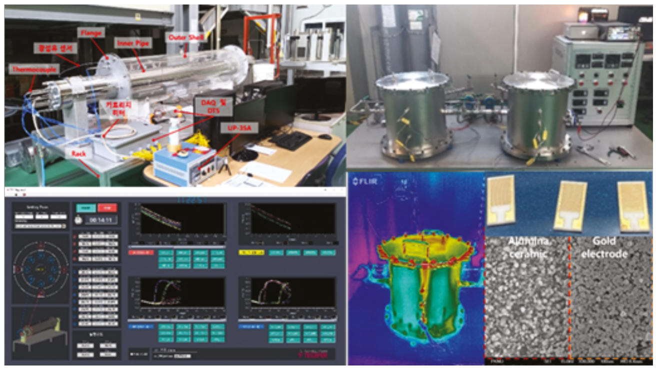


(b) Irradiation assembly of CN-1515 and CN-FMS.

Source: CIAE.



Elastic	Rayleigh Scattering	Distance & Loss Measurement	λ_1
Inelastic	Brillouin Scattering (11GHz:~0.09nm)	Strain & Temperature Measurement	$\lambda_1 \pm \lambda_B$
	Raman Scattering (13THz:~104nm)	Temperature Measurement	$\lambda_1 \pm \lambda_R$



Source: KAERI.

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

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

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

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

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

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Chair of the SFR SSC,
with contributions from SFR members

Very-high-temperature reactor

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

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

Main characteristics of the system

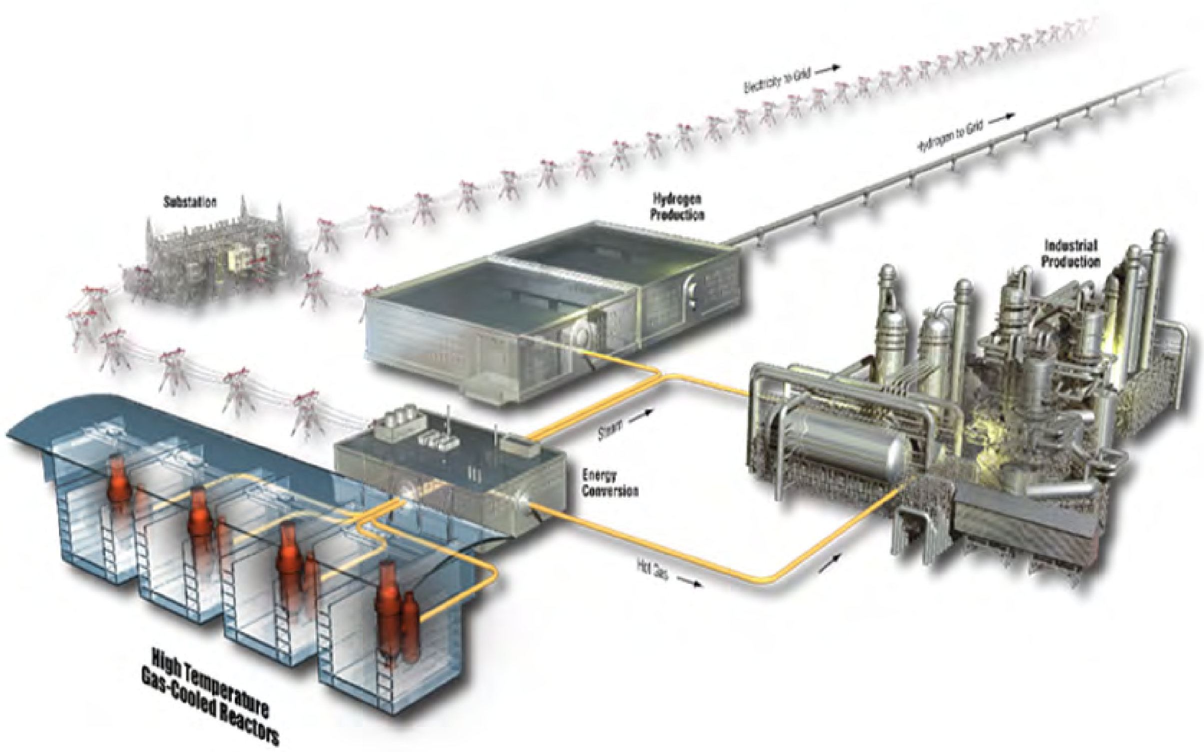
High-temperature gas-cooled reactors (HTRs or HTGRs) are helium-cooled graphite-moderated nuclear fission reactors using fully ceramic tri-structural isotropic (TRISO) coated particle-based fuels. They are characterized by inherent safety features, excellent fission product retention in the fuel, and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

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

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

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

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



Source: Gougar, H., (2014).

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

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

R&D objectives

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

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

Technical highlights – fuel and fuel cycle project

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

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

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

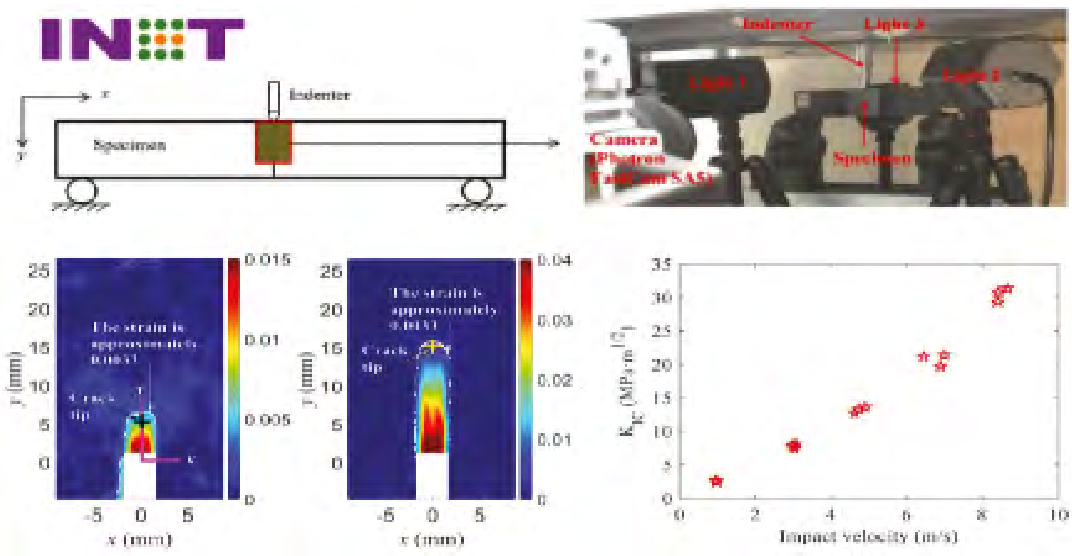
PIE of fuel pebbles discharged from the Chinese HTR-10 reactor is in progress at the newly commissioned hot cells at the Institute of Nuclear and New Energy Technology (INET). It includes burnup measurement and pebble deconsolidation. Hot testing of the new KÜFA furnace at INET, and subsequent accident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

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

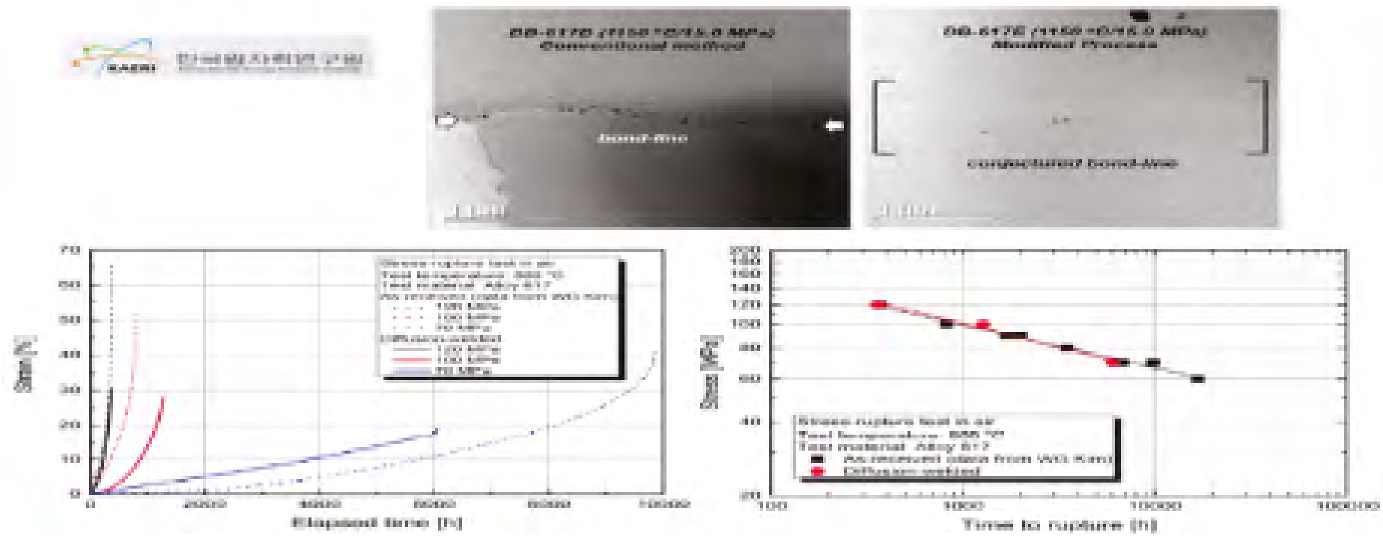
Several PMB members are pursuing the development of advanced TRISO fuel. In China, studies on UCO fuel kernel processing and fabrication of ZrC coatings are in progress. KAERI is currently developing fabrication methods for large (~800-µm-diameter) UO_2 kernels (with potential application in accident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

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

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



Source: INET.



Source: KAERI.

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

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

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

Technical highlights – materials project

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

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

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

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

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

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

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

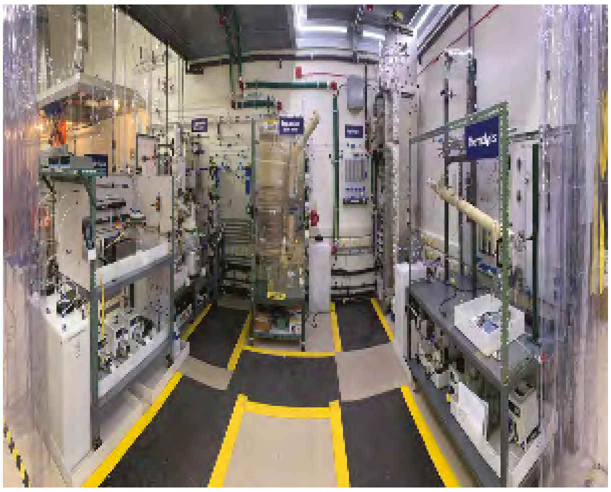
In the near/medium term, metallic alloys are considered as the main option for control rods and internals in VHTR projects, which target outlet temperatures below about 850°C. However, future projects are considering the use of ceramics and ceramic composites where radiation doses, environmental challenges,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

Technical highlights – hydrogen production project

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

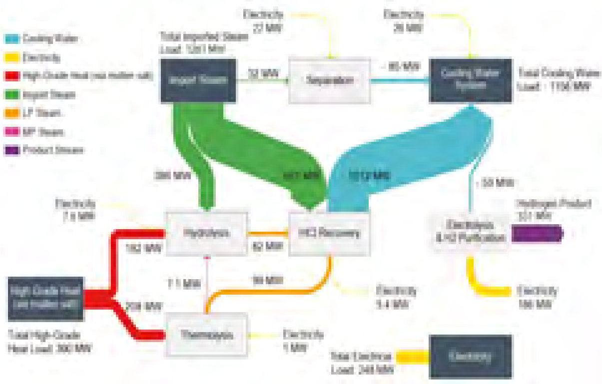
Over the last year, the Canadian effort has focused on setting up in laboratory a 50 NL/h hydrogen production system based on the copper-chlorine cycle. Figure VHTR-4 below shows the four steps of the process set up for integrated continuous operation.

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



Source: CNL.

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



Source: CNL.

Source: INET.

Figure VHTR-6: Prototype of the sulfuric acid decomposer

Source: INET.

Figure VHTR-7: Installation of the helium loop

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

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

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

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

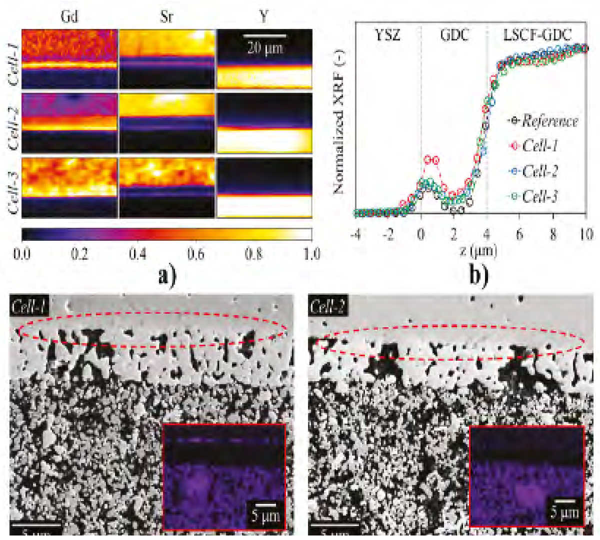
Comparative SO_3 splitting screening experiments with catalyst-coated SiSiC honeycombs, for periods between 100-950 hours onstream, demonstrated that platinum-based catalysts did not exhibit satisfactory conversion and suffered deactivation with prolonged on-stream exposure times at 650°C. This was attributed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

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

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

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

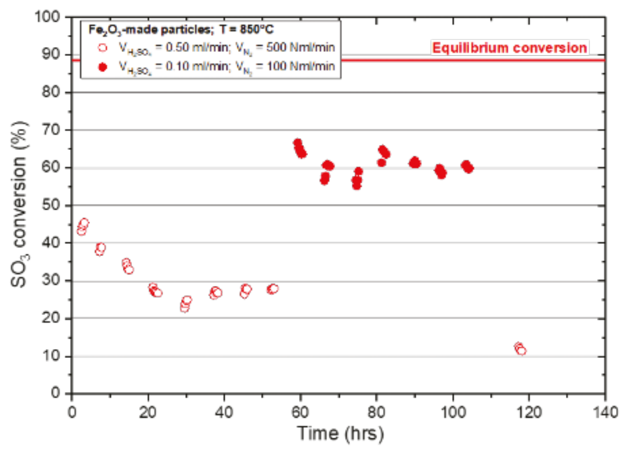
Figure VHTR-8: Analysis of operated cell



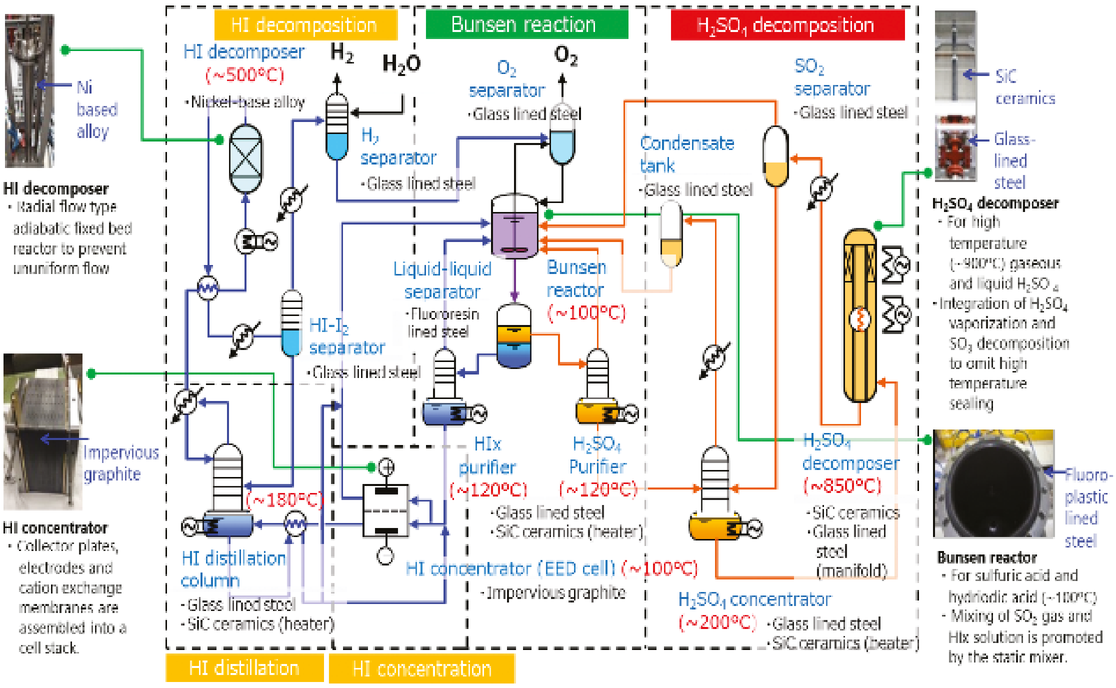
a) Normalized 2D element distribution obtained from the WRF signal at the electrode/barrier layer/electrolyte interface for cell-1, cell-2 and cell-3; b) Comparison of the integrated XRF signal of Sr between the pristine cell and the aged samples; and c) SEM images of the LSCF-GDC/GDC/8YSZ region for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

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

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



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



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

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

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

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

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

Technical highlights – computational methods validation, and benchmarks project

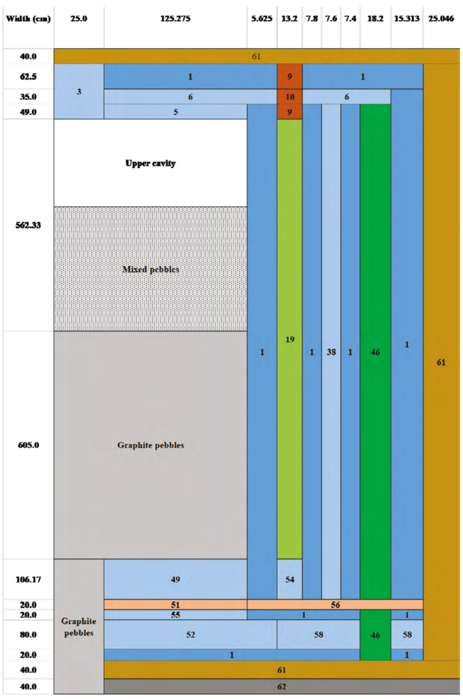
Validation of new computational methods and codes in the areas of thermal hydraulics, thermal-mechanics, core physics and chemical transport are needed for the design and licensing assessment of reactor performance in normal, upset and accident conditions. Code validation needs will be carried out through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

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

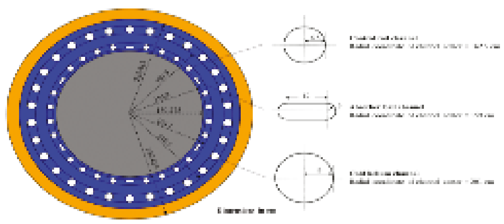
In Japan, the JAEA restarted the operation of the HTTR, a 30 MW research reactor on 30 July 2021, representing another major milestone for international HTGR activities in 2021. Safety demonstration tests under the framework of a OECD/NEA project will be carried out in 2022. There are also plans to perform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

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

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



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



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

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

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

4 Working group reports

Economic Modelling Working Group

The Economic Modelling Working Group (EMWG) was established in 2003 to provide a methodology for the assessment of Generation IV (Gen-IV) systems against two economic-related goals:

- to have a life-cycle cost advantage over other energy sources (i.e. to have a lower levelized unit cost of energy);
- to have a level of financial risk comparable to other energy projects (i.e. to have a similar total investment cost at the time of commercial operation).

The EMWG published *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* (GIF, 2007) and released the Excel-based software package, G4ECONS v2.0, providing the EMWG the means to calculate the levelized cost of energy and the total investment cost so as to evaluate Gen-IV systems against GIF economic goals. These resources were made available to the public through the GIF Technical Secretariat, resulting in subsequent publications[1] that demonstrate the EMWG methodology for the economic assessments of Gen-III and Gen-IV systems, as well for cogeneration applications, such as hydrogen production.

G4ECONS v2.0 was also benchmarked against economic models developed by the International Atomic Energy Agency (IAEA), including the Nuclear Economics Support Tool (NEST) and the Hydrogen Economic Evaluation Programme (HEEP), and the results were published in peer-reviewed publications. Lessons learned from the benchmarking exercise and from the feedback of users has informed the refinement of the G4ECONS tool. The EMWG released the latest version, G4ECONS v3.0, with an improved user interface, in October 2018. In 2022, the EMWG launched a survey to collect user feedback on G4ECONS v3.0 and identify potential model improvements.

In 2016, the EMWG started to investigate challenges and opportunities for the deployment of Gen-IV systems in emerging energy markets with an increasing share of renewable energy resources. The terms of reference for the EMWG were amended in 2018 to incorporate the expanded mandate so as to inform the GIF Policy Group and the Experts Group on the policies and R&D needs for the future deployment of Gen-IV systems.

Since October 2016, the EMWG has worked collaboratively with the GIF Senior Industry Advisory Panel (SIAP) to investigate challenges and opportunities for deployment of Gen-IV systems in electricity markets with a significant penetration of renewable energy resources and to produce a position paper for the Policy Group. An abridged version of the EMWG position paper on the impact of increasing shares of renewables on the deployment prospects of Gen IV systems was presented at the 4th GIF Symposium (2018) and an executive summary was posted on the GIF website.[2] The study found that Gen-IV systems need to be more flexible compared to current reactors for deployment in low-carbon energy systems, and such requirements are already being proposed by the utilities. Large-scale energy-storage and cogeneration applications, for example, would allow flexible dispatch while ensuring high-capacity utilization. Nuclear/renewable hybrid energy systems with adequate energy-storage and cogeneration applications could, in this way, meet flexible demands from the grid while operating power generators at full capacity to ensure overall economically viable operation. However, such flexibility considerations impose additional requirements on the R&D of Gen-IV systems.

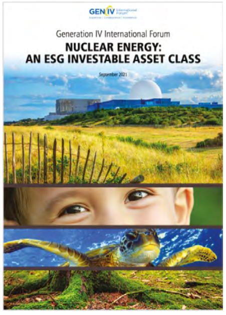
In 2022, the EMWG will survey modeling requirements for the G4ECONS software, develop new cost reduction strategies in accordance with the advanced nuclear technology cost reduction strategies and systematic economic review (ANTSER) framework and extend work on nuclear financing. The EMWG is engaging with the IAEA and the GIF Proliferation Resistance and Physical Protection Working Group (PRPPWG) on economic topics, including safeguards and security, cost modeling, and the economics of emerging and existing reactors.

1. All EMWG publications are available on the GIF website at www.gen-4.org/gif/jcms/c_9364/economics.
2. See: www.gen-4.org/gif/jcms/c_117864/2018-gif-symposium-proceedings.

EMWG activities in 2021

In 2021, the EMWG issued two major reports:

Nuclear Energy: An ESG investable Asset Class:

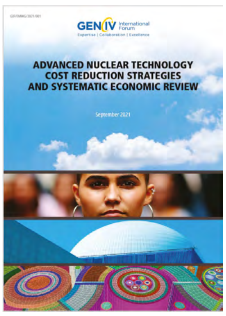


This report was produced by a finance industry taskforce set up by the EMWG in 2020 to consider the nuclear industry's ability to report against environmental, social and governance data collection and accounting metrics (ESG). Reporting well against ESG allows nuclear energy to be considered as an investable asset class, thereby enabling nuclear companies and projects to access climate finance.

The report has been produced by the finance community for the finance community. It establishes not only how nuclear energy, as an asset class, has the potential to report well against a wide range of ESG, but also highlights the importance of wide ranging, consistent and standardized ESG reporting to determine the credentials of all energy companies across their lifecycles and throughout their supply chains. The report discusses how ESG fit within international frameworks, including the UN Framework Convention on Climate Change (1992), the Kyoto Protocol (1997) and the Paris Agreement (2015), and how ESG are linked to the Green Bond Principles, while examining the relationship between ESGs and the various taxonomies and other policy documents being developed around the world.

The report can be viewed as a blueprint on how to set up a project and enable it to access financing. It received significant press coverage (e.g. from Standard & Poor's, Reuters, World Nuclear News) and was successfully presented in a dedicated workshop at the 2021 World Nuclear Exhibition.

Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review:



This report provides a process to produce a methodological framework for evaluating nuclear cost reduction strategies. Key areas for nuclear cost reduction strategies and technologies are categorized under design, construction/production and project management. Application of the framework is illustrated on reactor designs that are based on cost reduction through "functional confinement," followed by a presentation of a more rigorous application of the methodology.

The report has been written with three goals in mind: first, to create a methodology for creating impactful cost reduction strategies that cut across a broad set of advanced nuclear reactor technologies; second, to generate an example cost reduction strategy using the methodology; and third, to provide a path forward to improve the methodology and produce additional cost reduction strategies. The EMWG can lead future cost reduction strategy studies, but the range and impact of this body of work is greatly enhanced by direct contributions from the GIF methodology working groups, task forces and system steering committees for the six Gen-IV reactors, as well as SIAP. In 2022, the EMWG will investigate other cost reduction strategies, such as "modularity at scale".

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David Shropshire
Co-Chair of the EMWG

Megan Moore
Co-Chair of the EMWG

Fiona Reilly
Co-Chair of the EMWG

Education and Training Working Group

The Education and Training Working Group (ETWG) was launched in 2015 to enhance open education and training, as well as communication and networking among people and organizations supporting GIF. The ETWG has partnered with numerous university professors, scientists and engineers around the world to create and deliver 60 webinars from September 2016 to December 2021, all of which are archived on the GIF website.[1] The webinars have focused on advanced reactor systems, as well as on numerous other areas such as nuclear fuels, structural materials and the nuclear fuel cycle. The webinars presented in 2021 (highlighted in blue) are archived and retrievable in digital format (see Table ETWG-1).

Table ETWG-1: GIF webinar series from September 2016 to December 2021

	GIF webinars from September 2016 to December 2021 – Presented and archived
Introduction **3 webinars**	**Atoms for Peace** - John Kelly, United States (2016) **Introduction to Nuclear Reactor Design** - Claude Renault, France (2016) **European Sodium Fast Reactor: An Introduction** - Konstantin Mikityuk, Switzerland (2019)
Gen-IV systems **17 webinars**	**Sodium-cooled Fast Reactor** - Bob Hill, United States (2016) **Lead Fast Reactor** - Craig Smith, United States (2017) **Gas-cooled Fast Reactor** - Alfredo Vassile, France (2017) **Very-High-Temperature Reactors** - Carl Sink, United States (2017) **Supercritical Water Reactors (SCWRs)** - Laurence Leung, Canada (2017) **Fluoride-cooled High-Temperature Reactors** - Per Peterson, United States (2017) **Molten Salt Reactors** - Elsa Merle, France (2017) **MYRRHA: An Accelerator Driven System Based on LFR Technology** – H.A. Abderrahim, Belgium (2018) **Molten Salt Actinide Recycler & Transforming System with and without thorium-uraniumSupport: MOSART** - Victor Ignatiev, Russia (2018) **Lead containing mainly Isotope Pb-208: New Reflector for Improving Safety of Fast Neutron Reactors** - Evgeny Kulikov, Russia (2019) **Gen-IV Coolants Quality Control** - Christian Latge, France (2019) **Czech Experimental Program on MSR Technology Development** - Jan Uhlir, Czech Republic (2019) **GIF VHTR Hydrogen Production Project Management Board** - Sam Suppiah, Canada (2020) **Thermal Hydraulics in Liquid Metal Fast Reactor** - Antoine Gerschenfeld, CEA, France (2020) **Micro-reactors: A Technology Option for Accelerated Innovation** - D.V. Rao, United States (2020) **Overview of Small Modular Reactor Technology Development** - Frederik Reitsma, IAEA (2020) Experimental R&D in Russia to Justify Sodium Fast Reactors - Iuliia Kuzina, Russia (2021)
Operational experience **13 webinars**	**Feedback Phénix and Superphénix** - Joel Guidez, France (2017) **Design, Safety Features and Progress of HTR-PM** - Yujie Dong, China (2018) **ASTRID Lessons Learned** - Gilles Rodriguez, France (2018) **Advanced Lead Fast Reactor European Demonstrator, ALFRED Project** – A. Alemberti, EC (2018) **Russia BN 600 & BN 800** - Ilya Pakhomov, Russia (2018) **Safety of Gen-IV Reactors** - Luca Ammirabile, EC (2019) **The ALLEGRO Experimental Gas-Cooled Fast Reactor Project** - Ladislav Belovsky, Czech Republic, (2019) **Passive Decay Heat Removal** - Mitchell Farmer, ANL United States (2019) **Molten Salt SFR Safety Design Criteria (SDC) and Safety Design Guideline (SDG)** - S. Kubo, Japan 2020) **Reactor Safety Evaluation** - A U.S. Perspective - David Holcomb, United States (2020) Introducing New Plant Systems Design Code - Nawal Prinja, UK (2021) Experience of HTTR Licensing for Japan's New Nuclear Regulation - Etsuo Ishitsuka, Japan (2021) In Service Inspection and Repair Developments for SFRs and Extension to Other Gen-IV Systems - François Baqué, France (2021)
Gen-IV cross cutting topics **13 webinars**	**Energy Conversion** - Richard Stainsby, United Kingdom (2017) **Estimating Costs of Gen-IV Systems** - Geoffrey Rothwell, NEA/OECD (2017) **Materials Challenges for Gen-IV Reactors** - Stu Maloy, United States (2018) **Proliferation Resistance and Physical Protection of Gen-IV Reactor Systems** - Robert Bari, United States (2018) **Maximizing Clean Energy Integration: The Role of Nuclear and Renewable Technologies in Integrated Energy Systems** - Shannon Bragg-Sitton, United States (2020) **Global Potential for Small and Micro Reactor Systems to Provide Electricity Access** - Amy Schweikert, United States (2020) **Neutrino and Gen-IV Reactor Systems** - Jonathan Link, United States (2020) Overview of Waste Treatment Plant, Hanford Site - David Peeler, United States (2021) Opportunities for Generation-IV Reactor Designers through Advanced Manufacturing Techniques - Isabella Van Rooyen, United States (2021) Graded Approach: Not just Why and When, but How - Vince Chermak, United States (2021) Geometry Design and Transient Simulation of a Heat Pipe Micro Reactor - Jun Wang, United States (2021) Metal Fuel for Prototype Generation-IV SFR: Design, Fabrication and Qualification - Chan Bock Lee, Korea (2021) Development of an austenitic/martensitic gradient steel connection by additive manufacturing - Flore Villaret, France (2021)
Fuel types **5 webinars**	**General Consideration on Thorium as a Nuclear Fuel** - Franco Michel-Sendis, NEA/OECD (2017) **Metallic Fuels for SFRs** - Steven Hayes, United States (2017) **Advanced Gas Reactor TRISO Particle Fuel** - Madeline Feltus, United States (2019) **Performance Assessments for Fuels and Materials for Advanced Nuclear Reactors** - D. LaBrier, United States (2020) MOX Fuel for Advanced Reactors - Nathalie Chauvin, CEA France (2021)
Fuel cycle **4 webinars**	**Closing the Fuel Cycle** - Myeung Seung, Korea (2016) **Sustainability, A Relevant Approach for Defining Future Nuclear Fuel Cycles** - Christophe Poinssot, France (2017) **Scientific and Technical Problems of Closed Nuclear Fuel in Two-Component Nuclear Energetics** – Alexander Orlov, Russia (2019) **Comparison of 16 Reactors' Neutronic Performance in Closed Th-U and U-Pu Cycles** – Jerri Krepel, Switzerland, (2020)
Winners of Pitch Competition **5 webinars**	**Formulation of Alternative Cement Matrix For Solidification/Stabilization of Nuclear Waste** - Matthieu de Campos, France (2019) **Interactions between Sodium and Fission Products in case of a severe Accident in a Sodium-cooled Fast Reactor** - Guilhem Kauric, France (2019) **Security Study of Sodium Gas Heat Exchangers in Frame of Sodium-Cooled Fast Reactors** - Fang Chen, France (2019) **Development of Multiple Particle Positron Emission Tracking for Flow Measurement** - Cody Wiggins, United States (2020) Evaluating Changing Paradigms Across the Nuclear Industry - Jessica Lovering, United States (2021)

1. See: www.gen-4.org/jcms/c_84279/webinars

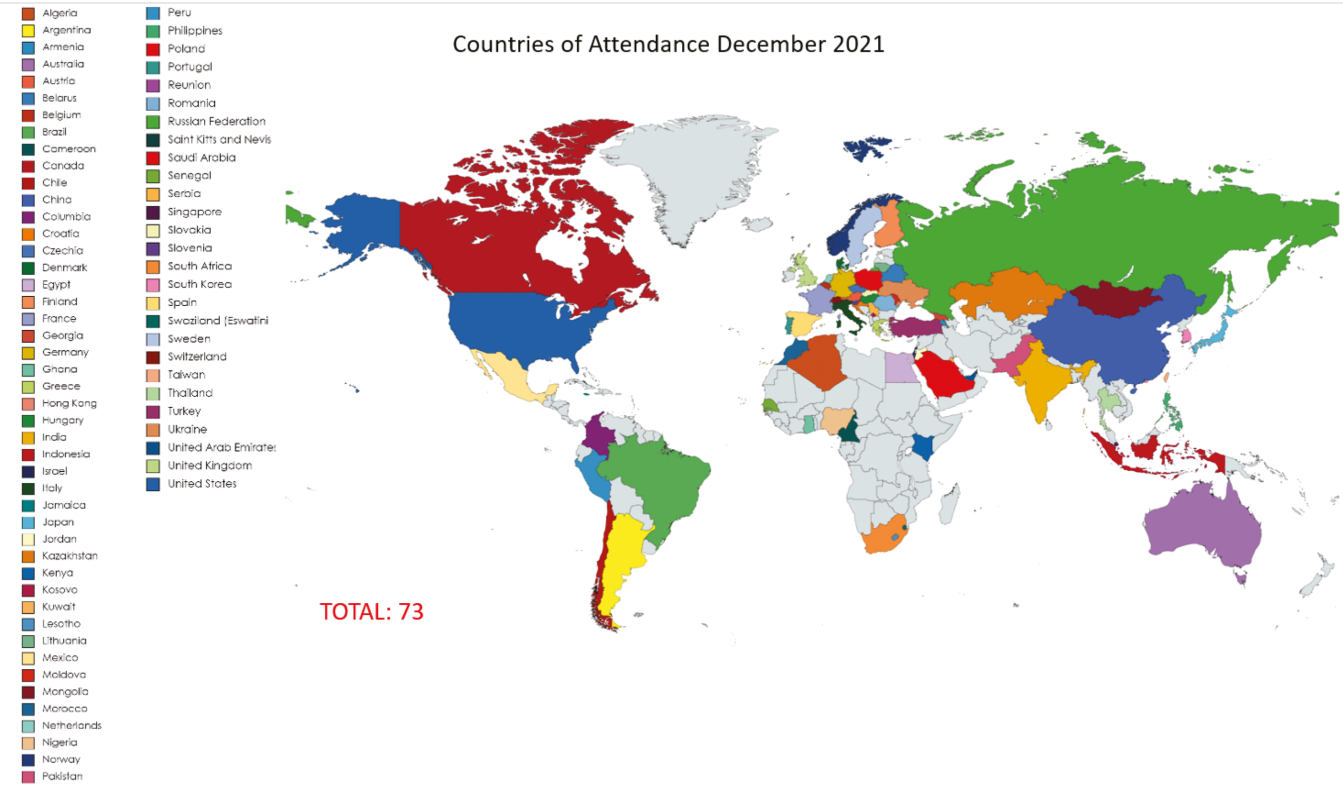


Figure ETWG-1. International participation in the GIF webinar series

As of December 2021, attendance during the live webcasts totaled 5 542. The number of viewings of recorded webinars in the online archive was 6 642. The total over a five-year period for webinar viewing is 12 084. These webinars have reached scientists and engineers across a total of 73 countries (Figure ETWG-1).

The GIF ETWG launched a virtual "Pitch your Gen-IV Research (PyGR)" competition in February 2021, for several reasons. The first reason was to involve junior researchers in the Gen-IV community by stimulating their interest. The second was to inform the public about advanced reactors and related topics, and at the same time, provide an opportunity to participate by voting for a preferred video/candidate/topic. Finally, the last reason was to have very short videos available on YouTube and Bilibili platforms that could be viewed any time to reach out to a large public and increase global awareness in the nuclear energy sector. The ETWG competition invited all junior researchers worldwide who were either PhD students or had completed their PhDs after 1 January 2019 to participate in the competition. Their research work needed to be related to Gen-IV advanced nuclear energy systems and could be either an independent research project or one working with a research mentor. Of the 51 submittals, the top 21 research projects were selected for the finals, in which proponents were invited to record a 3-minute video pitch. The six Gen-IV reactor systems (SFR, MSR, GFR, VHTR, SCWR, LFR) were represented with different topics. Three winners were selected for the 2021 PyGR competition, and each winner has been invited to present a webinar. In addition, the first prize winner will attend the next GIF Symposium planned in October 2022.

Also this year, an international panel discussion featuring the current and former GIF Chairs was organized and executed as a webinar in April 2021. The Chairs provided their perspectives on the progress of Gen-IV systems and prospects for the deployment of Gen-IV systems.

Patricia Paviet
Chair of the ETWG, with contributions from ETWG members

Proliferation Resistance and Physical Protection Working Group

The PRPPWG was established to develop, implement and foster the use of an evaluation methodology to assess Gen-IV nuclear energy systems with respect to the GIF PR&PP goal, whereby:

"*Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.*"

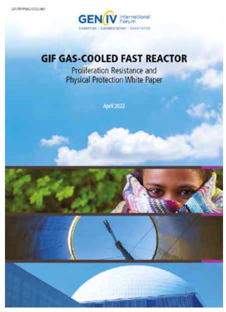
The methodology provides designers and policy-makers with a technology-neutral framework and a formal comprehensive approach to evaluate, through measures and metrics, the proliferation resistance (PR) and physical protection (PP) characteristics of advanced nuclear systems. As such, application of the evaluation methodology offers opportunities to improve the PR&PP robustness of system concepts throughout their development cycle. The working group released *Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems: Revision 6* (GIF, 2011) for general distribution,[1] and Japanese and Korean translations of the methodology report have been produced for national use.

Since 2018, the main focus of the PRPPWG has been on updating white papers on PRPP robustness of the six GIF design concepts. The past decade has seen several new advanced reactor vendors receiving funding from private and public investment, and these white papers provide recommendations to improve the safeguards and security of a variety of advanced reactor designs. This is a joint effort with System Steering Committees (SSCs) and the provisional System Steering Committees (pSSCs) of the six Gen-IV technologies. The first versions of these white papers were produced in the period from 2008 to 2011.[2] Currently, the papers are being updated according to a revised, common template. The current update reflects changes in the reactor designs with new tracks added and maturation of the designs of the six GIF systems, including through enhanced intrinsic PR&PP features.

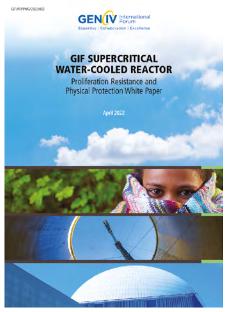
Individual white papers, after endorsement by both the PRPPWG and the responsible SSC/pSSC, will be transmitted to the Experts Group for approval and published as GIF reports. PR&PP aspects that transcend all six GIF systems are also being investigated. Cross-cutting topics include common themes, such as the fuel type, or topics not dealt with in the white papers, such as cybersecurity. The target is to complete and publish the reports on the GIF website during the course of 2022. Two white papers (i.e. on LFRs and SFRs) were finalized and made publicly available for download on the GIF website in October 2021 (GIF, 2021). Below is a summary status of the white papers still in the process of being updated as of the end of 2021.

- GFR – the draft has been updated, incorporating PRPPWG and GFR SSC recommendations. The PRPPWG is currently performing a final technical review and formatting prior to its release to the Experts Group.
- SCWR – the paper is in its final form, undergoing final review prior to its release to the Experts Group.
- MSR – the paper incorporated feedback from the pSSC and the PRPPWG in a new draft. The MSR pSSC is currently reviewing the latest draft.
- VHTR – the latest draft incorporated revisions by the VHTR SSC Chair after multiple exchanges. The PRPPWG will address additional SSC questions/comments in a modified draft of the white paper.

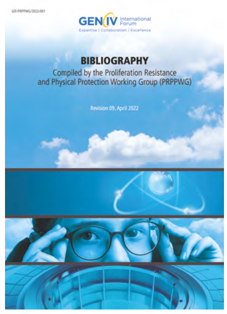
A paper outlining the status of updates to the six GIF reactor technology white papers was presented by the PRPPWG at the 2021 Institute of Nuclear Materials Management and European Safeguards Research and Development Association (INMM & ESARDA) Joint Virtual Annual Meeting, and is available in the meeting's online proceedings (Cheng, et al. 2021).



The PRPPWG maintains an annually updated bibliography of official publications, of publications referring to the PR&PP methodology and of relevant issues (GIF, 2021). The latest edition, revision 8, was published in April 2021. It is available on the GIF website.



The PRPPWG maintains regular exchanges with the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the agency's Department of Safeguards. An IAEA representative participates regularly in PRPPWG activities. The PRPPWG made a presentation at the 15th GIF-IAEA Interface meeting on 29-30 June 2021, highlighting collaboration on the INPRO PR methodology that the IAEA is updating, as well as on emerging safe-



1. See: www.gen-4.org/gif/jcms/c_40413/evaluation-methodology-for-proliferation-resistance-and-physical-protection-of-generation-iv-nuclear-energy-systems-rev-6.

2. See: www.gen-4.org/gif/jcms/c_40414/proliferation-resistance-and-physical-protection-of-the-six-generation-iv-nuclear-energy-system.

guards issues related to the deployment of small modular reactors (SMRs) and microreactors, and the interfaces between safety, security and safeguards.

Collaboration with the GIF Risk and Safety Working Group (RSWG) was strengthened through personal exchanges at each group's meetings. PRPPWG representatives attended the 32nd and 33rd meetings of the RSWG, and RSWG representatives attended the 32nd PRPPWG meeting. During the latter, it was agreed to expand collaboration on the investigation of interfaces between safety, security and safeguards, a topic that is also of interest to the IAEA. A dedicated meeting with participants from both working groups is foreseen in early 2022 to propose a path forward for this collaboration.

The PRPPWG is also engaged with the GIF Economic Modelling Working Group (EMWG) in exploring areas of potential collaboration. The EMWG co-chair joined one of the PRPPWG virtual meetings, and presented the advanced nuclear technology cost reduction strategies and systematic economic review (ANTSER) methodology. One area of mutual interest is the economic impact of safeguards by design.

The PRPPWG holds monthly teleconferences to report on the progress of group and member activities. The group held its 32nd annual meeting in virtual mode in three separate sessions (15 and 17 November, and on 6 December). Most member countries attended the meeting and delivered country reports. Representatives from the IAEA and the RSWG also participated. The meeting was dedicated to discussing the advancement of white papers, planning of new activities, such as the cross-cutting topics from the white papers, and developing the work plan for the period 2022-2023.

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Giacomo Cojazzi
Co-Chair of the PRPPWG

Lap-Yan Cheng
Co-Chair of the PRPPWG

Risk and Safety Working Group

The RSWG was formed in 2005 to promote a consistent approach to safety, risk and regulatory issues between the six Gen-IV systems, and to establish broad safety principles and attributes as input to R&D plans for specific design tracks based on high-level GIF safety and reliability goals. In 2021, RSWG membership included representatives from Canada, China, the European Union, France, Japan, Korea, South Africa, Russia, the United Kingdom and the United States. The IAEA Safety Department also participates as an observer.

The RSWG has developed the integrated safety assessment methodology (ISAM) as a technology-neutral toolkit to support design and evaluate risk and safety. After completion of the ISAM, the RSWG also supported its implementation by GIF System Steering Committees (SSCs) for specific Gen-IV design tracks as documented in a series of system-specific white papers. More recent emphasis has been placed on the preparation of system safety assessments as a summary of high-level safety design attributes and remaining R&D needs, also in close coordination with the respective GIF SSCs. Two ISAM documents, as well as the white papers and system safety assessment reports completed to date, are available on the GIF RSWG web page.[1]

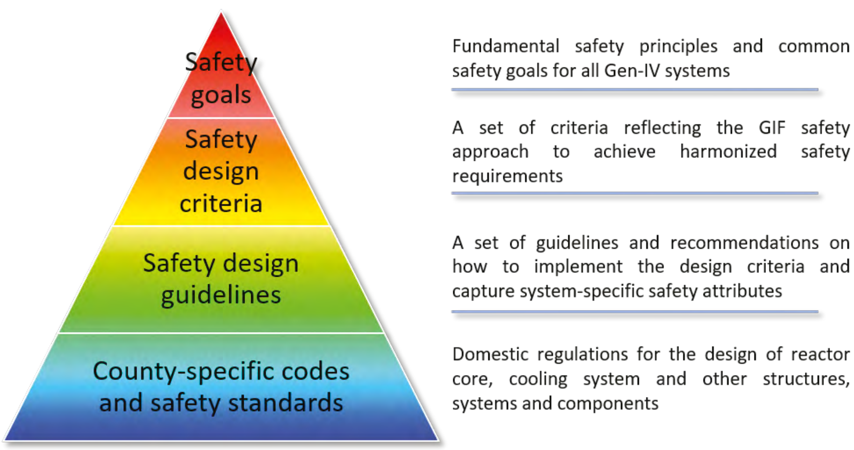
The first RSWG accomplishment in 2021 was the publication of the GIF report, *Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems* (GIF, 2021). As an update to the original 2008 version, the report aims to:

- capture the post-Fukushima recommendations and requirements to ensure a level of safety consistent with the regulatory expectations;
- provide common definitions for the plant states considered in a design and their alignment with the levels of defense in depth;
- clarify the concepts of design extension conditions and practically eliminated situations;
- achieve a balance between the prevention and mitigation of accidents, emphasizing the importance of reliance on inherent and passive safety;
- reinforce the independence of safety systems and features in different defense-in-depth levels.

The development of safety design criteria (SDC) and guidelines for specific systems is also an ongoing collaborative effort between the RSWG and SSCs to establish the basic requirements for design, fabrication, construction, inspection, testing and operation of Gen-IV prototypes. As shown in Figure RSWG-1, the SDC and guidelines are intended to fill the gap between the high-level GIF safety goals and country-specific codes and standards.

In 2021, the *Safety Design Criteria for Generation IV Lead-Cooled Fast Reactor System* report (GIF, 2021) was completed in collaboration with the LFR provisional SSC. Based on SFR SDC as the starting point, the LFR SDC introduces specific revisions that capture the impact of differences between sodium and lead as a very low Prandtl number liquid metal coolant with large thermal inertia and very high density, freezing and boiling points, induced coolant activity, and corrosion/erosion challenges posing elevated material compatibility concerns and requiring an elevated level of coolant purity control. Unique LFR thermo-fluid system designs with integrated primary coolant and power conversion systems (without an intermediate loop) are also addressed.

Figure RSWG-1: Hierarchy of safety standards



1. See: www.gen-4.org/gif/jcms/c_9366/risk-safety.

The LFR SDC mandates the robust assessment of accidents that may lead to core melt so as to demonstrate their practical elimination with a high degree of confidence.

The RSWG is currently collaborating with the VHTR SSC for SDC development. In 2021, an informal subgroup was created, gathering RSWG and VHTR SSC experts, who completed the first draft of SDC largely based on two previous IAEA efforts to revise the IAEA SSR 2/1 requirements for HTGRs. The draft VHTR SDC addresses consistency with the GIF basic safety approach and IAEA safety standards for design extension conditions, clarifies the requirements for confinement function versus the conventional containment structure, and introduces revisions and new requirements for the VHTR fuel forms based on coated fuel particles, as well as for unique coolant and decay heat removal system designs. The SDC development effort is expected to continue throughout 2022, before the report can be finalized.

The RSWG maintains technical interfaces with the OECD/NEA Working Group on the Safety of Advanced Reactors (WGSAR), which operates under the Committee on Nuclear Regulatory Activities (CNRA). In the past, the working group performed independent reviews of the SFR SDC and produced two SFR safety design guideline reports, providing extensive comments that led to significant revisions. Following its completion, the WGSAR is currently performing an independent review of the LFR SDC, and feedback is anticipated in 2022. In 2021, the RSWG also coordinated GIF contributions to the WGSAR report on "Fuel Qualification for Generation-IV Nuclear Energy Systems." GIF contributions provide high-level descriptions of SFR, VHTR, GFR, MSR, LFR and SCWR fuel types and forms, their role in the safety case and anticipated challenges for their qualifications.

Ongoing RSWG-WGSAR collaboration also includes the new joint initiative for the development of a risk-informed approach to the selection of licensing basis events and safety classification of systems, structures and components. In 2021, a position paper that introduces the foundational concepts and main elements of the approach was finalized by the RSWG and was distributed to WGSAR members. The paper establishes the event sequence categories considered in design to integrate the deterministic input and risk insights, defines a generic frequency-consequence target structure to categorize the event sequences against the regulatory requirements, outlines a process to classify the plant equipment based on their risk-significance and role in plant safety, and supports the selection of design-basis accidents and design extension conditions consistent with the safety classification of the responding plant equipment. Based on WGSAR comments and input, the position paper is expected to be finalized in 2023.

The RSWG also maintains technical interfaces with the IAEA and participates in the new IAEA initiative on the development of safety standards for SMRs and novel advanced reactors. Since these reactors have many technical similarities to GIF systems, selected RSWG experts have been sharing GIF experience while identifying system-specific safety features and supporting completion of the IAEA report on the applicability of IAEA safety standards to novel advanced reactors. The RSWG also contributes to the organization of joint IAEA-GIF technical meetings on the safety of liquid metal-cooled fast reactors that focus on the development of safety design guidelines (SDGs) for SFRs and SDC for LFRs.

Having completed most of its 2019 missions, the SFR Safety Design Criteria Task Force (SDC-TF) members have now joined the RSWG, and the remaining work on updating the SDGs for SFR systems, structures and components based on IAEA and WGSAR comments is being subsumed by the RSWG. IAEA and WGSAR comments are currently being addressed, and they are expected to be incorporated into the next version of the report in 2022. These new RSWG members (former SDC-TF members) will also support the pilot implementation of the proposed risk-informed approach for GIF SFR design tracks to demonstrate its applicability and ensure its consistency with the SFR SDC and SDG reports completed in earlier years.

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Tanju Sofu
Chair of the RSWG, with contributions from RSWG members

5

Task force reports

Advanced Manufacturing and Materials Engineering Task Force

Although some small modular reactors (SMRs) under development are based on simplified LWR technology, many are based on Generation-IV (Gen-IV) technology that utilizes the inherent safety capability arising from non-water-cooled reactors. A key to the successful mass deployment of the SMR approach is the assertion that the agile manufacturing of components and structures in factory-like environments enabled by large-scale production will substantially reduce the capital cost of new nuclear deployment. This reduction in capital cost requires innovation in the nuclear supply chain, particularly in the areas of advanced manufacturing and materials engineering, if they are to be delivered on time and on budget.

Nuclear design codes, however, typically dictate that only qualified materials and processes can be used, making qualification a potentially long and complicated process. Furthermore, developments in advanced manufacturing are occurring faster than our ability to introduce new materials and methods into design codes, potentially stifling innovation and hampering deployment.

The GIF Advanced Manufacturing Materials Engineering Task Force (AMME-TF) was formed to determine how collaborative R&D could be used to enable such advances so as to reduce the time to deployment of Gen-IV and comparable advanced reactors.

In 2019, the AMME-TF undertook a survey of advanced nuclear reactor vendors, supply chain specialists, regulators and other experts, discovering that most advanced manufacturing methods were considered opportunities by potential end users, and that 90% of respondents identified the greatest obstacle to their adoption as the creation and approval of the appropriate codes and standards.

In the first half of 2021, the task force undertook a second survey to investigate the "if and how" the community's activities and opinions on these issues had changed. The survey was sent to nuclear reactor vendors, supply chain specialists, regulators, national laboratories and potential end users. The distribution list was created based on recommendations from GIF member country representatives and similar experts. The intent, as in 2019, was to gather the opinions of people actively engaged in the design and manufacturing of advanced nuclear reactors. The size of the distribution list had doubled since 2019, providing further anecdotal evidence that the community working on the manufacturing and deployment of advanced reactors has continued to grow. Consequently, the number of survey responses also increased concomitantly. Replies were nearly equally divided from North America, Asia and Europe.

Turning to the survey itself, it is interesting to compare the responses from the 2019 survey to the 2021 survey. Overall, active interest in advanced manufacturing has expanded, with significant increases in reported efforts to support codes and standards and to secure regulatory approval, and with a substantial decrease in the number of respondents taking an interest, but adopting a "wait and see" attitude.

Regarding the application of advanced manufacturing, a change in the focus has been observed in terms of the reactor components being considered for advanced manufacturing, with an increase in interest in relation to reactor internals and heat transfer infrastructure (e.g. pipes, valves and heat exchangers) and a decrease in interest in relation to reactor vessels, fuel cladding and fuel assemblies. There was also an increased focus on the advanced manufacturing of traditional nuclear materials, with most efforts concentrated on stainless steels, low alloy steels and nickel alloys.

Overall, the results from the second survey in 2021 showed that both the size and the interest of the community had grown and become more focused. In addition, the community's focus on qualification, and modeling and simulation as a potential enabler for the qualification of advanced manufacturing and materials, was confirmed. Indeed, enthusiasm for engagement with the task force has also grown, with 54% of respondents reporting a *high* or *very high* interest in the idea of holding a workshop on modeling and simulation and 84% reporting a *high* or *very high* interest in holding a workshop on qualification.

In February 2020, before the COVID-19 pandemic began, the task force held an international workshop at the Nuclear Energy Agency (NEA) headquarters in Paris designed to investigate how collaborative R&D in the field of advanced manufacturing can be used to reduce the time to deployment of advanced reactor systems.

Details of the workshop, which was attended by over 70 representatives from conventional and SMR vendors, nuclear supply chain suppliers, regulators and researchers, can be found on the GIF website.[1]

1. See: www.gen-4.org/gif/jcms/c_115848/workshop-on-advanced-manufacturing.

The recommendation of the workshop was that collaborative activities should be actively encouraged in the following areas:

- qualification: including new modalities (e.g. real time process qualification);
- design and modeling: including the sharing of data for design optimization and modeling and simulation benchmarks.

The task force has thus developed an integrated plan for a series of workshops on both qualification and how modeling and simulation can be used to accelerate qualification. Since the outputs of a modeling and simulation workshop were perceived as representing a key input into a qualification workshop, it was decided to hold modeling and simulation workshop first on 8-9 November 2021.

The focus of the 2021 workshop was on how modeling and simulation can enable the qualification of advanced manufacturing. As was the case during the 2020 workshop, this virtual workshop contained several interactive small group sessions with peers, where attendees were asked to discuss and assess options and opportunities for the qualification of advanced manufacturing. Details of the workshop can be found on the GIF website,[2] and the workshop structure is shown in Figure AMME-1.

Session 1 of the workshop provided an overview of the potential for modeling and simulation to support the nuclear qualification of advanced manufacturing from both researcher and regulator's perspectives while session 2 showcased the activities of advanced reactor vendors and designers. Both sessions were designed to stimulate discussion among workshop attendees, with most of the time allocated to group sessions. In session 3, attendees were allocated to groups, and asked to discuss and identify opportunities for modeling and simulation that would help to accelerate qualification of advanced manufacturing. They were, then asked to prioritize, by likelihood of success, the best opportunities for deployment. The outputs of each of these deliberations were presented to the entire workshop during session 4, entitled "Comparing Group Outputs".

Some common themes emerged from the group presentations, enabling a number of distinct opportunities to be identified as areas for further analysis, and these include:

- collaborative modeling and simulation R&D to control the process (process qualification and quality assurance);
- collaborative modeling and simulation R&D to predict the microstructure and/or the material/component properties (enabling component design code compliance);
- collaborative R&D where modeling and simulation models, software and data are "shared" (including the idea of an advanced material property database);
- collaborative modeling and simulation R&D focused on codes and standards.

2 See: www.gen-4.org/gif/jcms/c_82829/workshops.

The task force is currently undertaking a detailed examination of the ideas and comments from the workshop and will present the full outcomes in 2022.

Figure AMME-1: Content and methodology of the AMME-TF Modelling and Simulation Workshop, 8-9 November 2021

AMME Workshop on Advanced Manufacturing (Virtual) Nov 2021, OECD, Paris, France

Mon 8 Nov	**GIF AMME Workshop on Advanced Manufacturing DAY 1**
Session 1 – Overview of workshop	
13:00 – 13.10	Welcome and Introduction, overview of AMME-TF, purpose of workshop! *Lyndon Edwards* ANSTO, AMME-TF Chair
13:10- 13:35	How Modelling and Simulation could be used to support the nuclear qualification of advanced manufacturing *Albert To*, University of Pittsburgh (20mins+5 min questions)
13:35 - 14:00	Challenges to the use of M&S to support the nuclear qualification of advanced manufacturing: a nuclear engineer's perspective *Pierre-François Giroux*, CEA (20mins+5 min questions)
14:00 – 14:25	NRC Action Plan for Advanced Manufacturing Technologies *Carolyn Fairbanks*, NRC (20mins+5 min questions)
Session 2 – Vendors and designers perspective: the view from the innovation front line	
14:30 - 14:50	*David Huegel & Clint Armstrong*, Westinghouse (15mins+5min questions)
14:50– 15:10	*Jean-Marie Hamy*, Framatome (15mins+5min questions)
15:10– 15:30	*George Jacobsen*, General Atomics (15mins+5min questions)
15:30– 16:00	Summary, Panel Discussion and Group Activity Briefing
Session 3 – Group activity 1	
16:00 – 17:30 **Includes break as necessary**	Attendees split into allocated groups, which undertake the following activities with the group Moderator/Rapporteur: a. Discuss and identify opportunities for Modelling & Simulation to accelerate qualification of Advanced Manufacturing b. Analyse each identified opportunity (SWOT analysis or similar) c. Prioritise, by likelihood of success, best opportunities for 2030 deployment d. Agree communication for Rapporteur to give to meeting at start of Day 2 (Can develop presentation overnight if necessary)
17:30	End of Day 1
Tue 9 Nov	**GIF AMME Workshop on Advanced Manufacturing DAY 2**
Session 4 – Comparing Group Outputs	
13:00 – 14:15	Whole workshop undertakes the following activities: a. Rapporteurs from each group presents group output (5 x 10 minutes) b. List agreed areas to be addressed and potential collaborative opportunities
14:15 — 14:45	Break
Session 5 – Group Activity 2	
14:45 – 16:00	Attendees split into new opportunity specific groups (those identified as having highest priority) and undertake the following activities: a. What needs to be done b. How can collaboration help? c. Identify and prioritize areas/ideas for AMME activities/projects d. Agree communication for Rapporteur to give to meeting after break
16:00 – 16:20	Break
Session 6 – Final Group Reporting and Meeting Outcomes	
16:20- 17:30	Each Group present their findings and recommendations
17:30 – 18:00	Summarise findings and consensus of meeting
18:00	End of Meeting

Lyndon Edwards
Chair of the AMME TF, with contributions from AMME members

Non-Electric Applications of Nuclear Heat Task Force

In November 2020 and February 2021, GIF held two, open brainstorming meetings out to exchange expert views on the position of Gen-IV systems (and particularly advanced modular reactors) regarding non-electric applications of nuclear heat (NEaNH). The outcomes from the brainstorming sessions suggested a path forward for the development of a NEaNH-related activity, and the following key areas were identified:

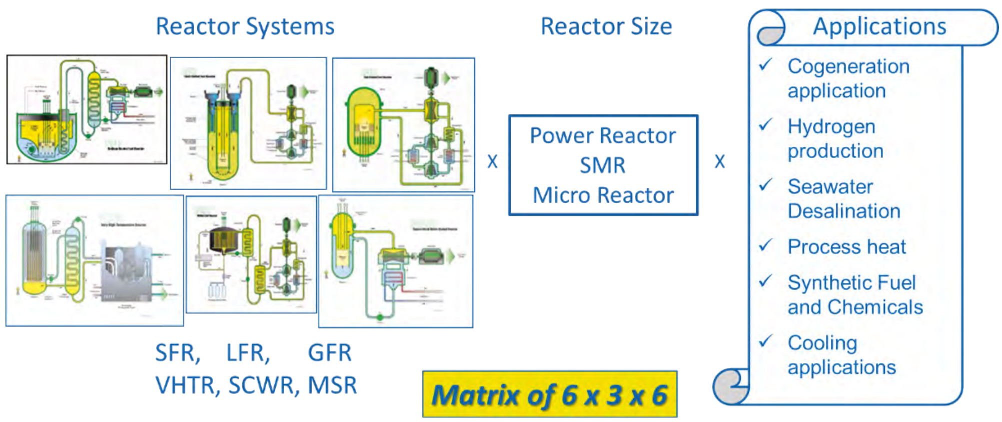
- GIF can play an important role in identifying the specific benefits that Gen-IV reactor technologies could bring to the NEaNH sector in the context of future energy markets.
- The development of a NEaNH-related activity launched under the auspices of GIF should be based upon the considerations listed below.
  - Organizations and members express an interest in contributing to the proposed activity and are able to share foundational knowledge and input data on the subject, such that a limited but relevant amount of useful documentation can be identified and produced to support the NEaNH objectives.
  - Overlaps with similar efforts undertaken within other nuclear organizations (e.g. IAEA, IEA, NEA, WNA) are avoided and opportunities for collaboration pursued.
  - An optimal balance is sought between addressing potential NEaNH configurations that can be achieved with Gen-III technologies (mainly pressurized water reactors) and those addressing specific high-temperature NEaNH processes, requiring that Gen-IV technologies be deployed. The complexity and variety of boundary conditions (e.g. local/global economy, geopolitics, government strategies) must be taken into account to determine the optimal combination of reactor technology, power level and NEaNH to fulfill specific requirements; this activity should aim at providing decision makers with the tools for pursuing optimal solutions which-depend on their specific needs rather than suggesting a technology down-selection.
  - In order to address the previous item, this activity should explore combinations arising from the "6 (NEaNH processes) x 3 (levels of power) x 6 (GIF reactor systems) matrix" (see Figure NEaNH-1) from different perspectives (geopolitical, economic, R&D, regulation). Use of such a matrix would enable the identification of potential technical and economic hurdles specific to the most promising combinations.
  - Efforts are needed to reach out to technical-economic communities beyond the nuclear power sector. In particular, efforts should be made to engage stakeholders in the high-temperature community (e.g. concentrated solar power, industrial processes, thermal energy storage).

Objectives of the GIF NEaNH Task Force

In order to fulfill the requirements outlined above, a dedicated, new task force was created in October 2021, the Non-Electric Applications of Nuclear Heat Task Force, or NEaNH TF (pronounced "NENH"). This task force is initially defined for a duration of 24 months, with a predefinition of the following items:

- Item 1: provide to GIF a position paper on its vision for NEaNH coupled to GIF systems (near term and transition to future).
- Item 2: upgrade and share the general level of knowledge to all GIF members by organizing an open workshop and establishing a shared database, implemented on the GIF website and updated regularly.

Figure NEaNH-1: Definition of the 6x3x6 matrix: 6 Gen-IV systems over 3 power ranges to address 6 major non-electric applications



- Item 3: highlight relevant solutions of the 6x3x6 matrix; creating a dedicated mind map to fill in potentially viable configurations.
- Item 4: implement within GIF a network that extends beyond the nuclear field and connects with the high temperature community.
- Item 5: analyze these systems in view of:
  - technology readiness levels, timeliness/ geographical suitability and footprint/CO_2 emission reduction potential;
  - cost evaluation (USD/t CO_2 saved), return on investment evaluation (annual emission reduction/required investment); several tools from the IAEA and OECD/NEA, as well as from participant organizations if available as open source tools, could be applied;
  - boundary conditions that are necessary to make such systems viable (e.g. cost of competing energy sources such as natural gas, levels of CO_2 tax, interest rates, discount rates).

With these results, GIF could deliver advice to policymakers, industry, licensing authorities and investors in terms of which nuclear applications are likely to be best suited (i.e. effective, timely) for meeting specific policy goals, how much these options may cost and how much economic benefit they could potentially offer. Participation in the GIF NEaNH TF has been confirmed by the following member countries: Australia, Canada, China, Euratom, France, Japan, Korea, Russia, the United Kingdom and the United States GIF plans regarding the NEaNH TF were shared at the IAEA Technical Meeting on the Role of at a Nuclear Cogeneration Applications Towards Climate Change Mitigation, held on 11-13 October 2021.[1]

Shannon Bragg-Sitton
Chair of the NEaNH TF,
with contributions from NEaNH TF members

1. See GIF presentation by G. RODRIGUEZ entitled "Non-Electric Applications of Nuclear-Heat: A new Generation IV International Forum initiative" IAEA Technical Meeting on the Role of Nuclear Cogeneration Towards Climate Change Mitigation, Virtual Meeting, Oct. 11-13, 2021 (Event code: EBT2003995).

6 Market and industry perspectives and the GIF Senior Industry Advisory Panel (SIAP) report

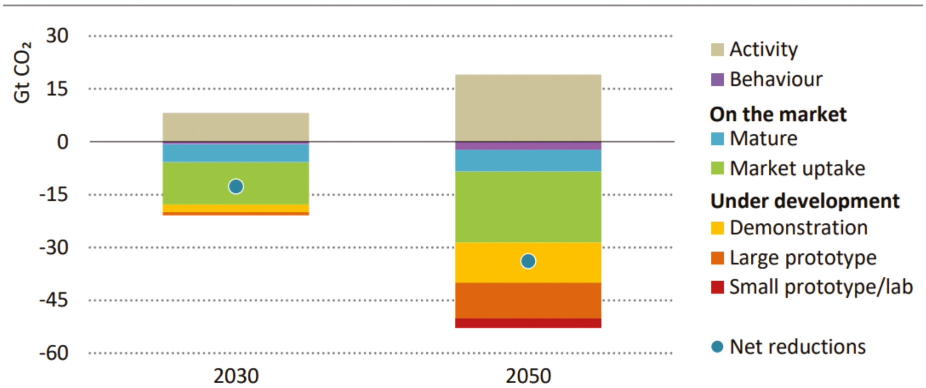
Nuclear energy is expected to play an important and growing role in a worldwide net-zero CO_2 emissions scenario and in achieving the decarbonization objectives set by the Paris Agreement. Continued operation of the existing fleet of nuclear reactors, as well as the new build of large-scale and small modular reactors (SMRs) could avoid 87 gigatonnes of cumulative emissions between 2020 and 2050 (NEA, 2021). By 2050, nuclear energy could displace 5 gigatonnes of emissions per year, which is more than what the entire US economy emits annually today. Pathways to net zero will further benefit from the development and deployment of near-term innovative nuclear technologies.

In line with the increasing number of country pledges to achieve carbon neutrality by 2050, the International Energy Agency (IEA) published its *Net Zero by 2050: A Roadmap for the Global Energy Sector* (IEA, 2021). More than 400 milestones across sectors and technologies have been identified to guide the global economy in its journey towards carbon neutrality by 2050. The "net-zero roadmap" also sees nuclear power playing an increasing role. Its global output should double by 2050, although its overall share in the global electricity generation falls below the current 10% with variable renewables covering most of the electricity needs.

While the emissions reductions in 2030 rely primarily on technologies currently on the market, those under development today account for almost half of the emissions reductions in 2050 (see Figure 6.1). The report calls for important decisions to be made concerning nuclear power in different areas, including the extent of government support for advanced modular reactors. Technologies still under development hold the potential to expand markets for nuclear power beyond electricity and to foster decarbonization in hard-to-abate sectors. The current and next decade will be critical to demonstrate the commercial viability of Generation-IV (Gen-IV) systems in order to achieve sufficient market uptake to remain relevant and aligned with net-zero targets.

Since its inception, the Generation IV International Forum (GIF) has been at the forefront of international cooperation in the research and development of Gen IV systems. After more than 15 years of research, GIF has accumulated a vast amount of know-how and research data.

Figure 6.1: Global CO_2 emissions changes by technology maturity in the IEA "net-zero roadmap"



Source: IEA (2021)

In parallel, private sector interest in advanced nuclear technology has been increasing over the last years, with a multiplicity of designs being proposed. Some of these advanced nuclear technologies aim to build first demonstration units before 2030.[1] While GIF activities concentrate on technical viability, private vendors are driven by commercial deployment. Therefore, there is a role for GIF to play in advancing Gen-IV concepts towards the construction of demonstration and/or first-of-a-kind (FOAK) units through well-designed collaboration with private actors. Collaboration can be mutually beneficial and can take place under the existing GIF governing framework, while respecting member countries' interests and intellectual property rights. Potential areas of collaboration, as well as specific actions and pilot projects, were already identified in the 2019 and 2020 SIAP "charges"[2] and recommendations. In 2020 in particular, it was underlined that "*the SIAP recommendations explore ways to create a perennial collaborative framework with industry with specific initiatives while respecting existing governing frameworks, member countries' interests and intellectual property rights. Complementarities and synergies with ongoing activities should be explored", with recommendation #9 more specifically highlighting that GIF should: "Continue to interact periodically with the private sector with dedicated meeting/workshops in order to identify critical research areas and priorities and match emerging experimental and qualification needs with existing or future R&D infrastructure capabilities.*"

Activities to accelerate deployment of Gen-IV systems and enhance collaboration with the private sector

The GIF Experts Group and Policy Group have well recognized the critical importance of building a smart connection with the private sector and increasing involvement of industrial partners in GIF activities. In February 2020, a GIF workshop was organized, with the participation of the private sector and involving two GIF task forces – the Advanced Manufacturing and Materials Engineering Task Force (AMME-TF) and the Research and Development Infrastructures Task Force (RDTF) – in order to assess the maturity and readiness of the existing research and manufacturing infrastructure to support the deployment of first demonstration and/or FOAK units. Participants in the workshop acknowledged the success of this workshop and the necessity to leverage on these preliminary interactions to build a perennial collaborative framework with the private sector. New activities recently engaged within GIF to achieve this goal include:

- GIF Forum with Industry 2022: organization of a new event, approved by the Policy Group in May 2021, to replace the traditional GIF Symposium (held every three years between 2009 and 2018), and merging with the AMME/RDTF Workshop, will adopt a more industry-oriented approach in order to foster collaboration on Gen-IV systems between the private and public sector.
- GIF paper on "Considerations Towards Establishing a Framework for Collaboration with Industry": drafting of a GIF report to explore options towards establishing a framework for collaboration between GIF members and private vendors. The objective of this framework would be to further the basic understanding of, and find solutions to, the common challenges associated with advancing Gen-IV and small modular technologies towards demonstration and deployment.
- Non-Electric Applications of Nuclear Heat (NEaNH) Task Force: creation of a new GIF task force, after three brainstorming sessions and official approval of the Policy Group in May 2021. Forum members agreed to create a cross-cutting task force dedicated to the non-electric applications of nuclear heat, including district and industrial heat applications, desalination and large-scale hydrogen production. The task force is already operating as an interim task force. Ideally, this task force will have a strong connection with industry and the potential end users of nuclear heat.

Senior Industry Advisory Panel report

Since the creation of the Generation IV International Forum in 2000, market conditions have continued to evolve, and they continue to be a common concern among users and developers of Gen-IV concepts. The role of the GIF Senior Industry Advisory Panel (SIAP) is to understand core drivers, opportunities and constraints related to the market environment with the objective of identifying the most appropriate advice in terms of GIF activities, in collaboration with the System Steering Committee (SSC) chairs, task forces and working groups, and with the guidance of the members of GIF Policy Group.

The Policy Group supported the work initiated in 2019 to build a smarter connection with the private sector and recommended the pursuit of these activities. SIAP continues to provide practical advice on strategic issues and specific activities already taking place within GIF to enhance collaboration with the private sector. SIAP recommendations suggest pathways to create a perennial collaborative framework with industry. Complementarities and synergies with ongoing GIF activities should be explored.

SIAP guidance and feedback is vital to support GIF initiatives aimed at enhancing the collaboration with the private sector. In terms of the specific requests for the three new activities noted above, this guidance and feedback is outlined below.

1. Some of these reactors have already made considerable progress, for example the HTR-PM helium-cooled gas reactor, the BREST-OD 300 LFR and the thorium molten salt reactor (TMSR-LF1).

2. A "charge" addressed to SIAP is a technical question or a problem emanating from the Policy Group and the Experts Group with the objective of obtaining SIAP feedback, and its vision and recommendations from an industrial point of view. Generally, a maximum of one charge per year is addressed to SIAP.

- For the GIF Forum with Industry 2022: SIAP is providing feedback and advice on the Forum's program, ideas for workshops (e.g. sessions involving the private sector) and relevant contacts from the industry, as well as for the potential end users of nuclear services (electricity/hydrogen/heat). The participation of interested SIAP members in the Organizational Committee of the Forum could be justified.
- For the GIF paper on "Considerations towards Establishing a Framework for Collaboration with Industry": SIAP is providing feedback on the first draft of the paper and advice on the associated action plan.
- For the Non-Electric Applications of Nuclear Heat (NEaNH) Task Force: SIAP is advising on the potential research program of the task force (e.g. integration of end-user requirements) and providing relevant industry contacts, in particular the end users of nuclear heat. It is putting a special focus on hydrogen production as one of the potential research topics for this cross-cutting task force. If possible, it will serve as a relay between GIF experts and end users, to better understand their principal features.

SIAP intentions for 2022

The Senior Industry Advisory Panel will continue to advise and support GIF in building better connections with the private sector and increasing involvement of industrial partners. It will stand ready to offer advice on how to interact with the private sector and implement recommendations in relation to previous SIAP charges. SIAP will also continue to provide industrial insight for GIF activities, create opportunities for the industry to bring together different viewpoints and to share experience in terms of the development and application of new technologies for Gen-IV concepts.

References

NEA (2021), *Mobilising climate and Development Finance to Meet the Paris Agreement: Perspectives on Nuclear Energy*, OECD Publishing, Paris, available at: https://oecd-events.org/cop26.

IEA (2021), *Net Zero by 2050: A Roadmap for the Global Energy Sector*, IEA, Paris, available at: www.iea.org/reports/net-zero-by-2050.

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Appendix 1. Country reports 2021

Australia

Australia remains a committed member of the Generation IV International Forum. We welcome the continuing expansion in the very-high-temperature reactor (VHTR) System Arrangement, and its Project Arrangements with the recent addition of the United Kingdom and the rejoining of Canada to the VHTR Materials Project Management Board (MAT PMB), as well as the further interest expressed by Russia.

We also support the development of stronger cooperation and collaboration within the molten salt reactor (MSR) system, initially through more coordinated programs of work under the existing memorandum of understanding (MoU) and then by the signing of an MSR System Arrangement and associated Project Arrangements.

Turning to national matters, Australia is investing more than half a billion dollars into building strategic international partnerships to make low-emissions technologies cheaper than high emitting alternatives. Australia has signed a letter of intent with the United Kingdom to establish a partnership on low-emissions solutions, which includes advanced small modular reactors (SMRs) that can use VHTR and MSR technologies.

Regarding the development of a national repository, on 11 August, the Minister for Resources and Water, announced his intention to make a declaration under the National Radioactive Waste Act that would confirm Napandee as the site for the National Radioactive Waste Management Facility. We are currently engaging in a public consultation process following that expression of intent.

The Australian government has no plans to lift the moratorium on nuclear energy in Australia. It recognizes that nuclear energy is a mature technology used to deliver reliable electricity in many countries, with virtually no greenhouse gas emissions.

Any decision to remove the current prohibition on nuclear power generation would require bipartisan agreement and wide-spread community support.

Australia continues to look at all emerging low-emissions technologies as they evolve over time, including nuclear energy, and to examine whether they are right for Australia

Australia acknowledges once again the GIF Secretariat and the GIF leadership team for their hard work and perseverance in developing and implementing virtual Experts Group and Policy Group meetings.

Canada

SMR activities and developments

Update on government policy and actions

Since the launch of Canada's SMR Action Plan, the government of Canada has continued to work with interested parties to advance this important initiative. The action plan is a pan-Canadian initiative with chapters submitted by more than 100 partner organizations from across the country. As of 31 December 2021, it includes 513 concrete actions which partners are taking to advance the development, demonstration and deployment of SMR technologies in Canada. Under the action plan, the government of Canada will establish an annual SMR Leadership Table, convening the Canadian nuclear family, from mining to production, as well as Indigenous Peoples.

The government of Canada has continued its engagement with Indigenous Peoples across Canada on SMRs, acknowledging that genuine, meaningful partnerships with Indigenous Peoples are critical components for Canada to capture the SMR opportunity. An Indigenous Advisory Council was created to enable a coordinated, Indigenous leadership voice from across Canada on the development of SMRs.

Under the *Hydrogen Strategy* for Canada, work is being conducted to explore the benefits and requirements of producing hydrogen with nuclear energy. Under the strategy, a working group was established in August 2021 dedicated to nuclear hydrogen, with the intention to advance the policy and technical work required to take actions on the strategy's recommendations.

In February 2021, The Premier of the Province of New Brunswick announced that the province is investing CAD 20 million (Canadian dollars) in ARC Clean Energy to support its SMR technology. These investments demonstrate strong support for New Brunswick's Advanced SMR Nuclear Energy Research Cluster, and in general for SMR R&D in Atlantic Canada.

In March 2021, Canada's Minister of Innovation, Science and Economic Development Canada, along with the Minister responsible for the Atlantic Canada Opportunities Agency, announced that the government of Canada is investing CAD 50.5 million in Moltex Energy Canada to help it develop a stable salt reactor – wasteburner reactor. This technology uses a waste to stable salts process that would use spent CANDU fuel. The federal government will also invest CAD 5 million in New Brunswick Power to help it prepare its Point Lepreau generating station site for the proposed installation of SMRs designed by Moltex and ARC Clean Energy. In addition, an investment of CAD 560 000 is being made for a research center at the University of New Brunswick to support SMR development and deployment in the province.

In April 2021, the province of Alberta formally became a signatory to the 2019 MoU between the provinces of Ontario, Saskatchewan and New Brunswick, which established a framework for the deployment of nuclear energy by way of SMRs in each jurisdiction. These provinces have agreed to collaborate on the advancement of SMRs as a clean energy option to address climate change and regional energy demands, while simultaneously supporting economic growth and innovation. For Alberta, this poses an opportunity to reduce the province's greenhouse gas emissions, power its remote communities, and provide a form of economic diversification.

The government of Canada is committed to continuous improvement with respect to ensuring that safe solutions are in place for managing radioactive waste, including waste that may result from new and innovative nuclear technologies. For this reason, Canada launched an engagement process to modernize Canada's radioactive waste policy. The Department of Natural Resources (NRCan) undertook an extensive engagement process to hear the views and perspectives of interested Canadians on how they would like to see the radioactive waste policy modernized. NRCan, with the support of other federal departments and agencies having responsibility for radioactive waste, engaged with non-governmental organizations, Indigenous Peoples, industry, other levels of government and other interested Canadians. In parallel to the radioactive waste review, NRCan is also undertaking a review of nuclear liability limits.

Utilities, vendors and regulatory SMR updates

On 2 December 2021, Ontario Power Generation (OPG) announced that it will work with GE Hitachi Nuclear Energy (GEH) to deploy GEH's BWRX-300 boiling water SMR at the Darlington nuclear site. OPG and GEH will collaborate on SMR engineering, design and planning, preparing the licensing and permitting materials, and performing site preparation activities, with the goal of constructing Canada's first commercial, grid-scale SMR, expected to be completed as early as 2028.

Global First Power Ltd (GFP)'s application to the Canadian Nuclear Safety Commission for a license to prepare a site for a Micro Modular Reactor (MMR) completed its preliminary evaluation step and has moved on to formal license review. Considerations of siting alternatives has led to the identification of a preferred location at the Canadian Nuclear Laboratories (CNL) Chalk River site, owned by Atomic Energy of Canada Ltd. The proposed project includes an MMR high-temperature gas-cooled reactor (HTGR) to provide process heat to an adjacent plant via molten salt.

The Canadian Nuclear Safety Commission (CNSC) continues work to ensure its readiness for SMRs in Canada. The CNSC undertakes an optional preliminary step before the licensing process, which is called a vendor design review (VDR), to assess vendors' understanding of Canada's regulatory requirements and the acceptability of a proposed design. There are currently twelve SMR proposals in the VDR process - ten VDR service agreements in force between vendors and the CNSC, and two more under development. These vendors are ARC Nuclear, GE Hitachi Nuclear Energy, LeadCold Nuclear, Moltex Energy, NuScale, SMR (Holtec), StarCore Nuclear, Terrestrial Energy, U-Battery Canada, Ultra Safe, Westinghouse and X Energy.

Industry action on climate change and SMRs

In June 2021, Canadian Natural Resources, Cenovus Energy, Imperial, MEG Energy and Suncor Energy formally announced the Oil Sands Pathways to Net Zero Initiative. These companies operate approximately 90% of Canada's oil sands production. Working collectively with federal and provincial governments, the initiative's goal is to achieve net-zero greenhouse gas emissions from oil sands operations by 2050 to help Canada meet its climate goals. A key pathway to net zero, identified in this initiative, will be evaluating, piloting and accelerating the application of potential emerging emissions-reducing technologies, including SMRs.

Other activities and developments

Refurbishment of OPG CANDU reactors at Darlington is progressing well. unit 2 has been successfully refurbished and restarted. In June 2021, the Bellows severing of unit 3 was completed, which marked an important milestone in the unit 3 refurbishment project.

Bruce Power's unit 6 major component replacement (MCR) project continues to proceed on track. The MCR project focuses on the replacement of key reactor components in units 3-8, including steam generators, pressure tubes, calandria tubes and feeder tubes.

International update

Canada's Nuclear Cooperation Agreement (NCA) with ITER was ratified in May 2021. The NCA opens the door to Canada's participation in the ITER project, enabling Canadian entities to provide goods and services to ITER in line with Canadian nuclear non-proliferation policy. This follows from the previous MoU between Canada and ITER that aimed to identify the precise domains in which Canadian suppliers could export expertise and technologies on a commercial basis to the ITER project. Under Canada's non-proliferation policy, Canadian-supplied nuclear material, equipment and technology may only be transferred where Canada has concluded a bilateral NCA. Under the terms of the NCA, a coordination committee was established between Canada and ITER as the main mechanism to notify both parties about possibilities of cooperative activities. The inaugural meeting of this coordination committee took place in November 2021. A working-level taskforce will explore specific areas of common interest, and discuss potential commercial and collaborative opportunities.

In June 2021, General Fusion announced that it intends to build and operate a fusion demonstration plant (FDP) at the UK Atomic Energy Authority's Culham Campus. General Fusion is the largest Canadian fusion energy company and among the leading private ventures pursuing a compact and economical fusion reactor. Construction of the FDP is anticipated to begin in 2022, with operations beginning approximately three years later. In July, 2021, General Fusion and CNL announced a partnership to develop tritium extraction techniques for use in commercial fusion power plants. Leveraging CNL's Canadian Nuclear Research Initiative, the organizations will identify the most promising approaches for managing tritium in fusion energy systems. In October 2021, General Fusion announced that it will open new headquarters, anchoring the company's operations in Canada.

In August 2021 the government of Canada signed an MoU with Romania on Civil Nuclear Cooperation, with the aim of employing the full breadth of Canada's nuclear supply chain to support Romania's refurbishment and new build civil nuclear projects. The MoU is a bilateral commitment that further strengthens government-to-government, industry-to-industry, lab-to-lab and regulator-to-regulator cooperation in nuclear energy development. Key areas of collaboration include CANDU refurbishments and new build, specifically related to Cernavoda units 3 and 4, SMRs, medical isotopes, supply chain integration, research and development, and best practices in waste management and decommissioning.

Canada's participation in GIF

Canada continues to participate in the supercritical supercritical-water-cooled reactor (SCWR) system with a current focus on materials and thermal hydraulics research to support development of the pressure-tube type small modular SCWR concept. In 2021, Canada re-joined the VHTR system, signing the GIF VHTR System Arrangement, and will be participating in the VHTR Materials Project. This is in addition to its current participation in the Hydrogen Production Project. Canadian entities continue their participation in the MSR Provisional System Steering Committee, and will work with other members on the future direction of MSR system collaboration.

China

Nuclear energy policy

On 3 September 2021, the operating license of the Qinshan nuclear power plant unit 1 was approved by the Ministry of Ecology and Environment to be extended to 30 July 2041. This decision will have a far-reaching impact on establishing a comprehensive nuclear operating license-extension system in China.

Nuclear energy development

By the end of 2021, there were 53 nuclear power units in operation in the Chinese Mainland, with an installed capacity of 54 646 gigawatts (GW), 20 nuclear power units under construction with an installed capacity of 21 356 GW. The total installed capacity is up to about 76 GW, ranking second in the world.

On 30 January 2021, unit 5 of Fuqing nuclear power plant, the world's first Hualong One reactor, entered commercial operation after 68 months of construction that began on 7 May 2015. It is the first Generation III nuclear power reactor completed as planned in the world. In addition, on 18 March 2021, unit 2 of the Karachi nuclear power plant in Pakistan, the first overseas Hualong One reactor, was successfully connected to the grid.

On 19 May 2021, President Xi Jinping and President Vladimir Putin witnessed the launching ceremony for units 7 and 8 of the Tianwan nuclear power plant and units 3 and 4 of the Xudapu nuclear power plant where four VVER-1200 reactors will be constructed.

On 17 June 2021, the first underground research laboratory for high-level radioactive waste began construction in Gansu Province, and will allow scientists to fully characterize the site's geology and determine its suitability for a high-level radioactive waste repository.

On 13 July 2021, the China National Nuclear Cooperation (CNNC) multi-purpose SMR demonstration ACP100 project, with a power generating capacity of 125 MW, began construction in Hainan Province. It will be the world's first commercial onshore SMR to start construction.

On 11 September 2021, China's first radioactive waste glassification facility was officially put into operation. It will convert hundreds of cubic meters of high-level radioactive liquid waste into a solid glass form, suitable for long-term storage and disposal.

On 9 November 2021, the nuclear heating commercial demonstration project, covering 4.5 million square meters and benefiting 200 000 residents, was put into operation in Haiyang city, Shandong province. It becomes the first city with zero-carbon heating in China, which fully demonstrates the innovative achievements and new prospects of the comprehensive utilization of nuclear energy.

On 20 December 2021, the high-temperature gas-cooled reactor – pebble-bed module (HTR-PM) demonstration project (unit 1) was connected to the grid successfully. It marks a substantial leap from laboratory to engineering application. At present, various tests are being carried out in an orderly manner to ensure unit 2's connection to the grid. Units 1 and 2 are expected to be in full commercial operation in mid-2022.

GIF activities in China

Sodum-cooled fast reactor (SFR): The China experimental fast reactor (CEFR) is preparing for material irradiation tests. CDFR's installation of the reactor vessel and its internals is undergoing and expected to be finished by the end of this year. It is planned to start commissioning in the first half of 2022.

VHTR: The HTR-PM loaded its fuel on 21 August 2021 and reached first criticality on 12 September 2021. It was connected to the grid in 20 December 2021 and reached full power operation in 2022. The R&D in fuel and fuel cycle, as well as MAT PMB, is going as planned, and the computational methods validation and benchmarks Project Arrangement was signed, with preparations to join the Hydrogen Production Project Management Board still ongoing.

SCWR: The CSR 1000 fuel design has been improved with better hydraulic performance. A small SCWR CSR-150 reactivity control method has also been proposed. The first China technical meeting about two international, thermal-hydraulic benchmark exercises was held in May. A round-robin test has been organized focusing on obtaining reliable corrosion rate data and reaching agreement on a corrosion mechanism and materials testing standards.

Lead-cooled fast reactor (LFR): The China General Nuclear Power Group has launched and R&D Alliance on the lead-bismuth-eutectic (LBE) cooled fast reactor, and established a series of experimental facilities to develop key technologies. The budget for the China Initiative Accelerator Driven System (CiADS) project was approved by government, and its kick-off meeting was held in July 2021.

Euratom

Nuclear energy policy

On 15 September, the President of the European Commission, Ursula von der Leyen, delivered her second State of the Union address in the European Parliament. Ms von der Leyen focused on Europe's recovery from the coronavirus crisis with the main drivers focusing on health preparedness, the Next Generation European Union (EU) financial program, and making Europe (and the world) greener, more digital and more socially just. On climate regulation and finance, the European Commission has introduced the "fit for 55" package. This is a set of proposals to make the EU climate, energy, transport and taxation policies fit for reducing net greenhouse gas emissions by at least 55% by 2030 (compared to 1990 levels), setting in stone the goals of making the EU the world's first climate-neutral continent by 2050.[1]

The inclusion of nuclear in the EU Taxonomy for sustainable activities regulation has been the subject of intense debate. Although nuclear energy is consistently acknowledged as a low-carbon energy source, opinions differ notably on economic aspects and on the potential impact on other environmental objectives, such as the long-term impact of nuclear waste. The European Commission requested that the Joint Research Centre (JRC) draft a technical report on the "do no significant harm" aspects of nuclear energy, which was delivered in March 2021. The Commission also mandated a review of this report by the experts of Article 31 of the Euratom Treaty (radiation protection and waste management), as well as by the SCHEER Scientific Committee on Health, Environmental and Emerging Risks. The Commission performed the analysis of the results of this consultation in accordance with the Communication on a Strategy for Financing the Transition to a Sustainable Economy published in July 2021. A complementary Climate Taxonomy Delegated Act draft covering nuclear and gas (not covered in the first EU Taxonomy Delegated Act) was prepared and will be finalized in early 2022. The document emphasizes adopting state-of-the-art and advanced technologies.[2]

Budget

The regulation establishing the Research and Training Programme of the European Atomic Energy Community for the period 1 January 2021 to 31 December 2025 was adopted on 10 May 2021.[3] The program defines a single set of objectives for indirect actions (co-funded research and innovation projects) and direct actions (the research program implemented by JRC). The aim of the regulation is to pursue nuclear research and training activities with an emphasis on the continuous improvement of nuclear safety, security and radiation protection, as well as to complement the achievement of Horizon Europe's objectives.[4] These objectives include that of the JRC, as the formally recognized Euratom implementing agent for the Generation IV International Forum (GIF), continuing to facilitate and coordinate the contribution and participation of the Euratom community in GIF's research and training activities. The contribution to GIF activities under the scope of the Euratom program is focused on safety, radiation protection, safeguards and non-proliferation research and training activities specific to Gen-IV systems.

EU member states agreed on an overall budget of EUR 1 382 million, of which EUR 583 273 million will be dedicated to indirect action for nuclear fusion. The multi-annual financial framework dedicated to nuclear fission research includes EUR 26 639 million for indirect actions and EUR 532 328 million for direct actions. The budget for 2021-2025 represents a decrease compared to the previous exercise. Appropriate measures will be introduced to cope with the reduced resource availability.

1. See: https://ec.europa.eu/info/strategy/priorities-2019-2024/european-green-deal/delivering-european-green-deal_en.

2. See: https://ec.europa.eu/info/business-economy-euro/banking-and-finance/sustainable-finance/eu-taxonomy-sustainable-activities_en#nuclear.

3. See: http://data.europa.eu/eli/reg/2021/765/oj.

4. See: https://ec.europa.eu/info/research-and-innovation/funding/funding-opportunities/funding-programmes-and-open-calls/horizon-europe_en.

Council Regulation (Euratom) 2021/100 of 25 January 2021, establishing a dedicated financial program for the decommissioning of nuclear facilities and the management of radioactive waste, and repealing Regulation (Euratom) No. 1368/2013,[5] was adopted. The scope of this new Instrument encompasses the Nuclear Decommissioning Assistance Programme for the dismissed nuclear power plants in Bulgaria and Slovakia, managed by the Directorate-General for Energy (DG ENER), and the operational Decommissioning and Waste Management Programme (D&WMP) of the JRC's obsolete facilities, managed by the JRC. The JRC is also tasked with fulfilling a knowledge management and dissemination role based on its own and collated experiences.

Research activities

Under the new Euratom Research and Training Programme, the Research & Innovation Work Programme 2021-22 was published on 1 July. The 49 eligible proposals received will be evaluated, and 30 projects will be launched by June 2022 for a total budget of around EUR 100 million of the total Euratom contribution to co-funded projects (indirect actions). This total includes an amount dedicated to actions addressing Generation IV advanced systems, with five proposals received.

The active involvement of JRC and Euratom representatives in GIF Steering Committees, working groups and task forces continues. There are significant contributions from JRC and EU member states to all of the six systems. In particular, the JRC contributes to the Risk and Safety and Proliferation Resistance and Physical Protection Working Groups ((RSWG and PRPPWG); and to the Advanced Manufacturing and Materials Engineering and Non-Electric Applications of Nuclear Heat Task Forces (AMME-TF and NEaNH TF). Moreover, the JRC is involved in 11 of the 13 indirect actions on Generation IV stemming from the previous Euratom Research and Training Programme. Support to the development of advanced nuclear systems remains a strategic component of JRC nuclear research in spite of the incompatibly with decreasing resources.

The Strategic Research and Innovation Agenda (SRIA) of the Sustainable Nuclear Energy Technology Platform (SNETP) was adopted in July 2021. SNETP is the largest nuclear stakeholder association in Europe, counting more than 100 major players. The SRIA emphasizes the important role of nuclear technologies and applications to achieve the decarbonization goals of the EU, deployed along the three pillars of SNETP: 1) the Nuclear Generation II and III Alliance (NUGENIA), with a focus on light water reactors; 2) the European Sustainable Nuclear Industrial Initiative (ESNII), with a focus on fast neutron reactors; and 3) the Nuclear Cogeneration Industrial Initiative (NC2I), focusing on high-temperature reactors and their applications. The document focuses on relevant challenges and the R&D needed to tackle them in the context of overall nuclear fission technologies and the specific aspects of each SNETP pillar; cross-cutting challenges with common R&D orientations are also highlighted.[6]

France

Nuclear energy policy

The French President Emmanuel Macron in October 2021 unveiled a five-year investment plan entitled "France 2030". The plan aims at fostering innovation. Decarbonizing French industry is one of the main targets. The plan includes a focus on SMRs, with a consortium led by Electricité de France (EDF) developing a 340 MWe SMR called NUWARD™. Its conceptual design development phase is ongoing until 2022. The consortium initiated a roadmap of technical discussions with the regulator in order to prepare the review of the Safety Options' Report in 2023.

Energy mix scenarios

French grid operator, Réseau de Transport d'Électricité (RTE, or the "Electricity Transmission Network"), released a report In October 2021 presenting six scenarios for the French energy mix, allowing France to achieve the aim of carbon neutrality by 2050. The scenarios consider different shares of nuclear and renewables. The report presents the main conclusions regarding consumption, energy mix evolution, economy, technologies, the environment and general considerations.

Status of the French fast reactor program

Regarding the SFR, digital simulation of the behavior of SFR subassemblies under irradiation is progressing, with the development of a methodology that couples thermal-hydraulics and thermal-mechanics computation. The comparison with the Japanese coupled calculation on the Joyo subassembly shows good agreement in computed deformation and temperature distribution.

Regarding the MSR, a one-week bootcamp has been held, gathering 70 people from the French Alternative Energies and Atomic Energy Commission (CEA), EDF, Orano, Framatome and the Institut de Radioprotection et de Sûreté Nucléaire (IRSN), as well as French start-ups. Strategy, scientific and organizational aspects were addressed.

Regarding facilities, the decision has been made to invest in experimental loops devoted to severe accident experiments (on both Gen-II, III and Gen-IV technologies). Commissioning a facility is scheduled in 2026. The facility will be broadly open for international cooperation.

5. See: https://eur-lex.europa.eu/eli/reg/2021/100/oj.

6. See: https://snetp.eu/wp-content/uploads/2021/09/SRIA-SNETP-1.pdf.

Japan

The government of Japan has outlined a “Carbon Neutral Policy” with the objective of decreasing carbon emissions by 46% in 2030, as compared with 2013, and reaching carbon neutrality by 2050.

According to a Cabinet Decision on the Sixth Strategic Energy Plan: the Advisory Committee for Natural Resources and Energy began its deliberations in October 2020, and a draft was presented on 21 July 2021. After amendments based on public and other comments, the Sixth Strategic Energy Plan was approved by the Cabinet on 22 October 2021.

For the stable use of nuclear power, the following points have been addressed:

- Restart of operation with safety as the top priority: launch of the restart acceleration task force, bringing human resources and knowledge together, and maintaining and improving technological capabilities;
- Measures for spent nuclear fuel: promotion of construction/utilization of interim storage facilities and dry storage facilities, etc. to increase storage capacity; and technology development for reducing the volume and harmfulness of radioactive waste;
- Nuclear fuel cycle: efforts towards the completion and operation of the Rokkasho Reprocessing Plant through a public and private partnership, obtaining the understanding of the relevant municipalities involved and international society, and through further promotion of plutonium-thermal (mixed oxide [MOX] fueled) power generation;
- Final disposal: steady implementation of literature surveys in two municipalities of Hokkaido, and commencement of surveys in as many areas as possible across Japan;
- Efforts in relation to various challenges in proceeding with long-term operation while ensuring secured safety: Fulfilling conservation activities and considering various issues depending on the individual roles of the public and private sectors;
- Public understanding: interactive dialogue, in regions where electricity is consumed, and easy-to-understand, polite public relations/public hearings.

For the promotion of R&D, making the most of the private sector’s ideas and wisdom, by 2030, the development of a fast reactor will be steadily promoted by utilizing international cooperation. The SMR technology will be demonstrated through international cooperation; and component technologies related to hydrogen production at a high-temperature gas-cooled reactor (HTGR) will be established. R&D on nuclear fusion will also be promoted through international collaboration, such as through the ITER Project.

The Nuclear Energy X Innovation Promotion (NEXIP) program is ongoing.

Several types of reactors (LWR, SMR, SFR, HTGR, MSR) are being proposed from the private sector. The government financially supports this program, and the Japan Atomic energy Agency (JAEA) offers technical support.

In terms of LWRs, 9 units have restarted, 7 units certified and 11 units are under examination under a new nuclear regulation.

The High Temperature Engineering Test Reactor (HTTR), the JAEA’s 30 MW experimental HTGR, restarted operation on 30 July 2021. While the HTTR was not severely damaged after the great earthquake of 2011 in Japan, regulatory requirements were enhanced in view of the lessons learned from the accident at the Tokyo Electric Power Company (TEPCO) Fukushima Daiichi Nuclear Power Station. After a five-year safety review by the Nuclear Regulation Authority, in conformity with new regulatory requirements, the JAEA obtained permission to restart the HTTR without significant reinforcement. The JAEA will carry out the safety demonstration tests by using the HTTR under the framework of an OECD/NEA project. Also, the JAEA has plans to conduct various tests to confirm safety, core physics and the thermal-fluid characteristics, as well as fuel performance. Furthermore, a demonstration plan for hydrogen production by the HTTR is under discussion.

The experimental reactor, Joyo, has been approved for the development of a large production of medical radioisotopes, which is seen as essential to Japan’s growth strategy from the perspective of economic security, and several billion yen is expected to be allocated in the budget for the fiscal year 2021. Additionally, plans are underway to build a new research reactor in Tsuruga.

Korea

Nuclear power in Korea

A total of 24 nuclear power reactors (21 pressurized water reactors [PWRs) and 3 CANDUs) were in operation as of October 2021, providing 160 184 GWh of electricity that corresponds to 29% of the total electricity production in Korea. The installed nuclear capacity of these 24 reactors accounts for 17.7% (23 250 MWe) of the total capacity. Two PWRs, Shin-Kori units 5 and 6, are under construction. Shin-Hanul unit 1 was recently permitted to operate, but Shin-Hanul unit 2 is still waiting for permission to operate.

Nuclear energy policy and R&D in Korea

The goal of Korea’s national energy policy has changed, with the portion of power generation from renewable energy sources increasing to 20% by 2030, and the share of nuclear and coal-based electricity production gradually reducing. During 2021, the Korean government established the 2050 Carbon Neutrality Committee under the direct control of the President of Korea. The committee presented the nationally determined contribution (NDC) by 2030 as a 40% reduction compared to 2018 emissions.

The government still supports both the export promotion of nuclear power plants, R&D activities relevant to enhancing nuclear safety and nuclear dismantling and disposal technology. Remarkably, in April 2021, the national assembly, the government, industry, academia and research institutes have gathered together to prepare a plan for the development of an innovative SMR. It was an official, new movement to develop an innovative SMR in Korea. The Korea Hydro & Nuclear Power Co. and the Korea Atomic Energy Research Institute announced the cooperative development of an LWR-based SMR by 2028.

Sodium-cooled fast reactor

As for SFR developments, current SFR R&D activities focus mainly on obtaining the technical feasibility of SFR technologies, coupled with pyro-processing to support back-end fuel cycle activities as an alternative option for domestic spent nuclear fuel management. In September 2021, a commission was established to examine the adequacy of pyro-processing and the SFR technology. A further R&D program to develop pyro-processing and SFR technologies will be planned just after the commission's decision by the end of this year.

SFR development is also focusing on the new SMR market in near future. The dedicated SFR project granted by the Korean government started in early 2021, and is focused mainly on the development of an SFR-based SMR that facilitates global export towards a market expected in the middle of 2030s. The ongoing SFR project consists of three groups of common and necessary technologies, which are dealing with a set of sodium experimental capabilities, advanced modeling and simulation, and innovations in metal fuel technologies. The specific R&D areas include multi-physics analyses methods coupling fluidic characteristics with thermal-structural behaviors, the uncertainty quantification methodology in the field of reactor core thermal hydraulics, high-fidelity advanced instrumentation technologies for practical applications and sodium-free annular metal fuel technologies.

Very-high-temperature gas-cooled reactor

In early 2019, the Korean government announced its national plan for the hydrogen economy, which centers on two axes of hydrogen-powered vehicles and the hydrogen fuel cell. Hydrogen demand in the 2040s is expected to reach 5.26 million tons per year. Nuclear hydrogen production using VHTRs was evaluated as one of the possible clean hydrogen production technologies.

A project, called "Very High Temperature System (VHTS) Key Technology Development", was launched in April 2020 to develop the performance evaluation technologies of design and analysis codes, and the performance verification technologies of high-temperature materials. In addition, a sub-project of VHTS key technology development, called "Development of essential technologies for hydrogen production coupled with VHTS", was launched in April 2021. This sub-project focuses on the development of coupled technologies between the very high-temperature system and high-temperature steam electrolysis (HTSE) hydrogen production. Hydrogen production tests will be completed with a 6 kWe HTSE module using high-temperature steam from the very high-temperature helium loop.

Russia

The generation of electricity by nuclear power plants in Russia in 2020 reached 215.74 billion kilowatt-hours. This is an absolute record in the country's nuclear power industry since Soviet Union times. According to Concern Rosenergoatom forecasts, in 2021, it is expected that Russian nuclear power plants will generate electricity at the level of 218 billion kWh.

In early June 2021, the first unit of the Belarusian nuclear power plant was put into commercial operation, and the construction of the second unit of this plant will be completed by the end of the year. In May 2021, China launched the construction of the units 7 and 8 of the Tianwan nuclear power plant, as well as units 3 and 4 at the Xiudapu nuclear power plant. In June 2021, the construction of the fifth unit of the Kudankulam nuclear power plant in India also began. In March, work started on the third unit of the Akkuyu nuclear power plant in the Republic of Türkiye.

Manufacturing of equipment for China's CFR-600 fast neutron reactor is underway and progressing well.

Rosatom has signed an agreement with the Russian Republic of Yakutia for the construction of a low-power, ground-based nuclear power plant, based on the SMR RITM-200, in Ust-Yansky Ulus. Four units are to be constructed, and a license for the construction has been obtained.

Rosatom's first wind power plant (WPP) – Adygeyskaya – with a capacity of 600 MW, has been commissioned. Another wind farm – Kochubeevskaya in Stavropol – is under construction.

The optimization of competitiveness for the BN-1200M project has been completed and a recommendation has been received from Rosatom's Scientific and Technical Councils on its construction as the 5th unit of the Beloyarsk nuclear power plant.

A full-scale industrial production of mixed uranium-plutonium oxide fuel for the BN-800 fast neutron reactor has been put in place at FSUE GHK. In 2020, the first reloading of the BN-800 reactor core with MOX fuel (one third of the core) was successfully completed, confirming the possibility to provide the full fuel supply for the BN-800 reactor. In the same year, the first batch of BN-800 fuel assemblies was manufactured with MOX fuel from highly-radioactive plutonium dioxide extracted from the VVER reactor's spent nuclear fuel (SNF).

In 2022, it is planned to load the next batch of BN-800 MOX fuel assemblies, which will completely transfer the core of the BN-800 reactor to this type of nuclear fuel.

Construction work at the site of the multi-purpose fast neutron research reactor MBIR is continuing at a good pace. In 2021, construction and installation works are 10-15% ahead of schedule.

The reactor vessel was manufactured at the Atommash factory. An intermediate slab was installed at the base of the reactor shaft, on which the installation of related equipment has begun.

In July 2021, the international consortium agreement was signed that legally formalized the relations of the first key participants of the consortium, and fixed the rights and obligations of participants for the use of the MBIR reactor resource after reactor commissioning.

Rosatom is open to international partners' accession to the International Research Center Consortium.

Rosatom has begun to shape the appearance of a new technological platform in Russia for nuclear generation in the horizon after 2030 – two-component nuclear energy with a closed nuclear fuel cycle. It will multiply the available fuel supply and solve issues related to the management of SNF.

In June of this year, the construction of a nuclear power unit with a fast neutron reactor, BREST-300, was launched in Seversk, in the Tomsk region. For the first time in the world, a nuclear power plant with a fast neutron reactor and facilities for a closed nuclear fuel cycle will be constructed at the same site. This is, without exaggeration, a milestone for the global nuclear power industry.

Mr Mariano Grossi, Director General of the International Atomic Energy Agency (IAEA), and other partners took part in the ceremony in virtual mode. The construction of the fuel fabrication and refabrication module has been completed, and the stage of installation of technological equipment has begun. For nitride fuel, the characteristics of fuel readiness for a burnout level of 6% have been fully confirmed. This research continues and results have been obtained for the burnout of 9%, and assemblies are being prepared to achieve a burnout of 12%, which is the most economically justified level for the BREST-300 reactor. Research for the BREST-300 SNF reprocessing cycle has been completed. It is has been proven that C-14 is exuded at the level of 99.9%.

A side event was held at the IAEA General Conference to demonstrate the success of this breakthrough project and to start organizing international partnerships based on the new technological platform mentioned above.

Rosatom has joined work recognizing nuclear energy as a "green" source of power and promoted nuclear energy as such at the 2021 United Nations Climate Change Conference in Glasgow.

New "developments for the future" in the nuclear industry are being carried out within the framework of the Russian state approved integrated program entitled "Development of equipment, technologies and scientific research in the field of nuclear energy use in Russia for the period up to 2024". The program includes research on two-component nuclear power, the development of an experimental infrastructure, thermonuclear and plasma technologies, new materials and technologies for advanced energy systems, and reference (baseline) nuclear power plant power units, including low-power nuclear power plants and VVERs with spectral regulation.

As part of GIF work, Rosatom is considering the possibility of signing the System Agreement on molten salt reactors, as well as additions to Project Agreements on "Improved Fuel and Safety and Operation of fast sodium neutron reactors".

A project agreement has been signed on the equipment projects and the SFR energy conversion unit, and a project agreement on thermal hydraulics and safety is being prepared as part of the work on supercritical reactors. In 2021, Mr A.V. Moiseev, a representative from Russia, was elected Chair of the interim System Steering Committee for the LFR.

In September 2021, Kuzina Yu.A., from the Institute of Physics and Power Engineering in Russia, held a webinar on "Experimental R&D in Russia in support of fast neutron reactor projects".

South Africa

Nuclear new build program

Following the issuance of a non-binding request for information (RFI) in June 2020 for 2 500 MW of nuclear energy, the Department of Mineral Resources and Energy received twenty-five responses globally from nuclear technology vendor companies. The assessment of received responses, which culminated in a consolidated report, was completed in March 2021 and it showed that there is a strong interest from nuclear technology vendor companies for delivery of the South African nuclear new build program, which is envisaged to comprise conventional nuclear power plants and SMRs.

In August 2020, the Department of Mineral Resources and Energy submitted a Determination under Section 34 of the Electricity Regulation Act of 2008 for 2 500 MW of nuclear energy to the National Electricity Regulator of South Africa (NERSA) for concurrence. The public consultation process was thereto undertaken in February 2021. In August 2021, NERSA issued a concurrence under Section 34 of the Electricity Regulation Act of 2008 for 2 500 MW of nuclear energy to be procured. The NERSA concurrence came with a number of suspensive conditions, which the department is currently engaged in a process to address.

Subject to Cabinet approval, discussions are ongoing to revive research and development of the Pebble Bed Modular Reactor Programme in line with the government policy position as stated in the Nuclear Energy Policy of 2008. R&D work is expected to consolidate all previous efforts, including work progress by Eskom for advanced high-temperature reactor technology development.

Koeberg long-term operation

Eskom's implementation of the Koeberg Long-Term Operation Programme is continuing as planned and is subject to the National Nuclear Regulator (NNR) Act, 1999 requirements. The Koeberg nuclear power plant reaches its 40-year end-of-design life in 2024, and in January 2021, Eskom submitted a formal application to the NNR to extend the operating license of the Koeberg nuclear power plant for another 20 years until 2044. It is anticipated that Eskom will submit a safety case for long-term operation to the NNR by July 2022.

Strengthening the regulatory framework for long-term operation of nuclear installations in South Africa, the Minister of Mineral Resources and Energy gazetted regulations in March 2021 in terms of Section 36 of the NNR Act, 1999, on long-term operations of nuclear installations.

Multi-purpose reactor project

In September 2021, the Cabinet favorably considered the project initiation report and approved the setting up of the multi-purpose reactor project to replace the SAFARI-1 research reactor by 2030. The prefeasibility report has been completed by the South African Nuclear Energy Corporation and has undergone an independent gateway review, following which the project has entered the feasibility stage and is planned to be completed by mid-2023. This will include an RFI, site licensing, environmental impact assessment and concept design, as well as communication and stakeholder engagement activities to be initiated in 2022.

Centralized interim storage facility

The Republic of South Africa, through the National Radioactive Waste Disposal Institute, and with oversight by the Ministerial Steering Committee, is implementing a centralized interim storage facility project for off-site storage of SNF. The pre-feasibility report for the centralized interim storage facility project has been submitted to Cabinet and noted. The project is currently in the feasibility phase and is planned for commissioning in 2030.

Radioactive Waste Management Fund Bill

The Radioactive Waste Management Fund Bill, aimed at collecting funds through the polluter pays principle for management of SNF, is at an advanced stage of drafting, with the bill currently being consulted via different Cabinet clusters with the aim of obtaining Cabinet approval to publish the bill for public comment soon.

National Nuclear Regulator Act Amendment Bill

On 12 May 2021, Cabinet approved the publication of the National Nuclear Regulator Act Amendment Bill for public comment, which was done in June 2021. The act has been amended to, among others, strengthen issues of nuclear security and give effect to enforcement powers for nuclear inspectors. The department aims to complete all of the necessary work to approach Cabinet so as to obtain approval for the bill to be tabled in Parliament soon.

Switzerland

GIF activities

The operation of the Paul Scherrer Institute (PSI), where most of GIF research activities are taking place, were maintained at all times in spite of the COVID-19 pandemic.

On the MSR research side, the major focus is on safety, without a particular preference for a specific MSR concept. However, the fast reactor option is being considered as a reference system and chloride salt as the potentially most promising salt type for fuel cycle simplicity. During 2021, studies related to the breed-and-burn fuel cycle in the molten chloride fast reactor were ongoing at PSI.

The SFR research studies are also focusing on reactor safety in order to define design improvements. The Horizon-2020 ESFR-SMART project, coordinated by the PSI and devoted to safety of one of the Gen-IV systems was prolonged by one year to address delays in experimental works resulting from COVID-19. Two project proposals related to SFR safety assessment, with significant participation on the part of the PSI, have been submitted to Euratom.

On the materials research side, a focus has been set on the development of new analytical tools for high-temperature materials. It includes the preparation of a laser and infrared-based equipment to measure the thermal conductivity of tubular silicon carbide (SiC) composite materials. This also represents an extension of a former approach using a unit based on thermocouples and a central heating wire. It is part of a running PhD thesis on the micro/macro-structure analysis of the SiC composite, based on X-Ray tomography and the connection to the measured conductivity values trough finite element method (FEM) models. In addition, a new synchrotron-based tomography analysis method, delivering a significantly increased resolution and contrast that not only shows the pores but also the fibers in the composite, is being developed. This will enable an extension of the thermal conductivity model by the interfaces between the matrix and the fibers.

A milestone has also been reached with the integration of the PSI creep data in the Materials Handbook.

Switzerland (Mr Manuel Pouchon) co-chaired the biannual VHTR Materials PMB virtual meetings in September 2021, and delivered an update on the Swiss situation and Gen-IV related materials research.

Switzerland, represented by Mr Pouchon, regularly participates in GIF AMME-TF meetings, which will potentially have some implications for the materials/components of Gen-IV reactors.

Finally, a new PhD thesis focusing on the basic mechanisms of irradiation induced creep has started at PSI. This study is based on the measurement of diffusion coefficients in unstrained, strained and strained plus irradiated samples. This thesis will help to develop the corresponding creep model, which is not well understood today. Knowledge of the mechanism will lead to a better understanding of the materials behavior under intense radiation and high temperatures. These conditions are typical for Gen-IV systems.

Politics and regulation

The Swiss government has confirmed its strategy to reach net-zero CO_2 emissions in 2050.

The new "Energy Research Masterplan of the Federal Government", for the period 2021-2024, was released in November 2020. It clearly states the need to maintain expertise in the nuclear energy field in Switzerland and maintains nuclear energy research as an important component, also explicitly underlining research on Gen-IV reactors. The support to nuclear energy research is also integrated into the research strategy of the ETH Board, the strategic management body of federal polytechnic schools and research institutes.

The Swiss Federal Nuclear Safety Inspectorate (ENSI) has released new guidelines, in particular concerning the requirement for a geological waste repository. ENSI continues to systematically update and adapt these guidelines to the IAEA requirements.

A two-week IAEA Integrated Regulatory Review Service (IRRS) mission took place in Switzerland from 18-29 October 2021.

Operation of Swiss nuclear power plants and waste management

The dismantling of the Mühleberg/boiling water reactor (BWR) is advancing quickly and according to plan.

The Leibstadt BWR reactor has undergone an extensive revision, including the replacement of the main reactor circulation pumps, in spite of the COVID-19 crisis. The plant was restarted at the beginning of December 2021, with about a one-month delay in scheduling.

Other reactors are in operation and running at nominal power.

Nagra, the company in charge of realizing the final repository for nuclear waste in Switzerland, has almost completed all deep drillings (one is still ongoing) to acquire detailed information on the geology of the three possible locations for a geological waste repository. The results of these studies are meant to determine the detailed geological differences between the possible sites and back up the final choice for the location of the repository, which should be announced at the end of 2022. Nagra published a detailed report on the state of knowledge and research at the end of 2021 as requested by the Swiss regulation. The safety authorities (i.e. ENSI) will evaluate the report during 2022.

Nuclear-power-related research in Switzerland

The focus of the Nuclear Energy and Safety Division (NES) at PSI is to deliver a strong contribution to the education of the next generation of nuclear experts, scientific support for the safe operation of LWRs, delivery of the scientific basis for the assessment of deep geological repositories' safety and the technology monitoring, including research work on Gen-IV concepts.

Three new professors with leading functions in NES have been appointed and will contribute to know-how conservation in nuclear engineering and associated fields (radiochemistry, high performance computing, modeling and data science, energy system analysis) in Switzerland. They are also actively participating in the Swiss Master Course in Nuclear Engineering at the two polytechnic schools (i.e. ETHZ and EPFL).

In spite of the non-association of Switzerland to the HORIZON-Europe research initiative, PSI researchers are nevertheless actively participating at many consortia. Their participation will be financed directly by Switzerland's State Secretariat for Education, Research and Innovation.

United Kingdom

Nuclear energy update

The United Kingdom was the lead on the *Pathways to net zero using nuclear innovation* publication for the Nuclear Innovation: Clean Energy (NICE) Future Initiative on the occasion of the 2021 Clean Energy Ministerial.

The UK government has also confirmed its decision to adopt the UK Climate Change Committee's Sixth Carbon Budget in full, thereby setting a new target to cut UK carbon emissions by 78% by 2035.

In the second quarter of 2021, low-carbon sources generated 53.1% of the United Kingdom's total electricity generation. Renewables accounted for 37.3% and nuclear 15.8%. These figures were lower than last year, driven by lower wind speeds, a decrease in solar and hydro generation due to weather conditions and outages at all but one of the UK's nuclear power plants.

Plans continued to establish the Regulated Asset Base (RAB) model to fund new nuclear projects at a low cost of capital, saving consumers money.[7]

The UK government aims to bring at least one large-scale nuclear project to the point of a final investment decision by the end of this Parliament, subject to clear value for money and all relevant approvals.

7. According to the UK government: "The Nuclear Energy (Financing) Bill will introduce a Regulated Asset Base (RAB) model as an option to fund future nuclear projects. A RAB model is a tried and tested method, typically used in the UK, to finance large scale infrastructure assets such as water, gas and electricity networks. Under this model a company receives a license from an economic regulator to charge a regulated price to consumers in exchange for providing the infrastructure in question." See: www.gov.uk/government/news/future-funding-for-nuclear-plants.

In December 2020, the UK government announced the start of formal negotiations on Sizewell C, and those negotiations are ongoing.

A new GBP 120 million Future Nuclear Enabling Fund is to provide targeted support in relation to barriers to entry for advanced nuclear technologies. Further details of how this fund will operate will be published in 2022, alongside details of a roadmap for deployment that takes into account value for money.

This new fund is also supported by the existing Advanced Nuclear Fund of up to GBP 385 million to invest in the next generation of nuclear energy. Including up to GBP 215 million for SMRs and up to GBP 170 million for an R&D program to deliver an advanced modular reactor by the early 2030s.

The Generic Design Assessment (GDA) was opened in May 2021 to advanced nuclear reactors.[8] Rolls-Royce (as part of the UKSMR consortium) have stated their intention to make an application to enter the GDA.

COP26 update

The UK COP26 Presidency convened the Climate and Development Ministerial on 31 March 2021. The COP26 President emphasized the need for partners to come together around practical solutions. He reiterated that steps that would be taken today were in support of the COP26 Presidency's stated goals of cutting emissions to keep 1.5 degrees in reach, facilitating greater action on adaptation, mobilizing finance for climate action and coming together to make the COP26 negotiations a success.

The UK Presidency continued to work with countries, institutions, civil society and others on issues and actions in the run up to COP26.

UK GIF membership update

The UK is continuing to present project proposals for engagement with the GIF SFR and VHTR Project Arrangements and will be seeking formal agreement from the other partners to join these arrangements as soon as possible. A big thank you to all those coordinating this approval process and the hard work supporting the United Kingdom's wider engagement with GIF's exciting agenda.

The GIF Economic Modelling Working Group (EMWG) report, covering financing of nuclear energy, was released in August. We would like to thank Fiona Reilly, Co-Chair of the EMWG, for her work leading this white paper, which identified barriers to private sector investment and provided recommendations on the changes required to remove these barriers. This report will help realize the potential of advanced nuclear systems to meet the wider world energy demands and the decarbonization agenda in a flexible manner, and alongside other low-carbon energy production.

The United Kingdom has joined the interim Non-Electric Applications of Nuclear Heat Task Force (NEaNH) and looks forward to working with other task force members on taking this work forward across all six GIF technologies.

Public perception

"Sciencewise" is a program set up to assist UK government policymakers in conducting public dialogue to inform their decision making on science and technology issues. The UK government is working with Sciencewise to deliver a number of virtual 'public dialogue' events in order to explore views around modular nuclear technologies (SMRs and AMRs). The outputs from this project were published in August 2021.

International collaboration

The United Kingdom was the lead on the *Pathways to net zero using nuclear innovation* publication for the NICE Future initiative on the occasion of the Clean Energy Ministerial 2021.

This publication brought together policy perspectives on nuclear innovation's role in helping countries achieve their decarbonization goals.

Regulation and Generic Design Assessment

Generic Design Assessment (GDA) is the process carried out by the Office for Nuclear Regulation (ONR) and the Environment Agency to assess the safety, security and environmental protection aspects of a nuclear power plant design.

The GDA provides confidence that the proposed design is capable of being constructed, operated and decommissioned in accordance with the standards of safety, security and environmental protection.

UK Government funding since 2017 has enabled the ONR to shape multilateral and bilateral cooperation towards practical deliverables, common regulatory positions and harmonization on key technical regulatory expectations that are aligned with the UK regulatory framework.

In May 2021, the United Kingdom opened the GDA to advanced nuclear reactors.[9]

Research and development

In July 2021, the UK government launched a "call for evidence", inviting views on the UK government's preference to explore the potential of HTGRs to enable an AMR demonstrator by the early 2030s and support net zero by 2050.[10]

The Call for Evidence closed on 9 September 2021 and found no significant, additional evidence to materially change the outcome of the government's underpinning analysis.

8. See section entitled "Regulation and Generic Design Assessment" on page 91, and www.onr.org.uk/new-reactors/.

9. See: www.gov.uk/government/publications/entry-to-the-generic-design-assessment-for-advanced-nuclear-reactors.

10. See: www.gov.uk/government/publications/advanced-modular-reactors-amrs-technical-assessment.

As a result, the AMR RD&D Program will focus on HTGRs with the ambition for this to lead to a HTGR demonstration by the early 2030s, at the latest. The scope of the program is being developed.

United States

Nuclear energy continues to be a vital part of energy development strategy to put the United States on a path to net-zero emissions by 2050. The US Department of Energy (DOE) is aggressively working to revive, revitalize and expand nuclear energy capacity as we appreciate a historic period of unified congressional support for nuclear energy. The US Office of Nuclear Energy (NE) is leading the effort to move new and innovative advanced reactors, SMRs and microreactors from the conceptual and development stages into the commercial energy sector. The NE executes its mission through investments in research and development efforts with national laboratories, US universities and industry technical organizations, as well as through partnerships with the US industry and commercial stakeholders to develop and demonstrate advanced reactor technologies and designs.

The following summary briefly highlights some of the activities supported by the DOE-NE in 2021 to accelerate the development and deployment of advanced reactor technologies.

Advanced Reactor Demonstration Program

Recognizing the importance of advanced reactors meeting the Nation's energy security and climate change goals, the Advanced Reactor Demonstration Program (ARDP) was initiated in fiscal year 2020 to develop federal and US nuclear industry partnerships in the construction and demonstration of domestic advanced nuclear reactor designs that are safe and affordable to build and operate. The ARDP directly addresses advanced reactor development, demonstration and deployment through two focused pathways: advanced reactor demonstration and risk reduction for future demonstration.

Advanced reactor demonstration pathway

The goal of the advanced reactor demonstration pathway is to realize the construction and operation in the US of two innovative reactor designs within a seven-year time period, following the initiation of the awards. The NE selected two projects to be supported through the demonstration pathway, and cost-shared cooperative agreements for the two reactor projects were completed in early 2021. The two designs that are to be built and operated by calendar year 2028 under this pathway are:

Natrium by TerraPower, LLC: Natrium is a 345 MWe SFR that leverages decades of development and design undertaken by TerraPower and its partner, General Electric-Hitachi. The high operating temperature of the Natrium reactor, coupled with a molten salt thermal energy storage system, will allow the plant to provide flexible electricity output that could complement variable renewable generation technologies such as wind and solar. In November 2021, Team Natrium announced its intent to site the demonstration reactor at the Naughton coal plant site in Kemmerer, Wyoming. By demonstrating the Natrium reactor at a retired coal plant, TerraPower will not only take advantage of the existing energy infrastructure that is in place, but also the workforce. This project will leverage innovations from the concentrated solar power, tunneling and combined cycle gas turbine industries, and intends to utilize the latest advanced construction and manufacturing methods to bring thousands of jobs back to the area. Also, given the potential synergies between Natrium and the DOE Versatile Test Reactor (VTR) project, stakeholders from both projects have developed an MoU to facilitate collaborations between the two projects, with the intent of optimizing technology development cost and schedule for both efforts.

Xe-100 by X-energy, LLC: The Xe-100 is a 100 MWe pebble-bed high-temperature gas-cooled reactor that is ideally suited to provide flexible electricity output, as well as process heat for a wide range of high-temperature industrial heat applications, such as desalination and hydrogen production. The reactor will be fueled by TRISO pebbles, providing a robust safety profile. The demonstration is currently planned to be sited at the Hanford, Washington reservation on land leased to the US utility, Energy Northwest. X-energy is working with an affiliated local utility, Grant County Public Utility District, to become the owner/operator of the plant so as to provide electric power to their service district.

Risk reduction for future demonstration pathway

The goal of the risk reduction for future demonstration pathway is to support potential future demonstration of additional advanced reactor technologies through cost-shared, competitively awarded projects that are designed to maximize the utility of the results across the nuclear energy industry. The DOE NE selected five projects to aid advanced reactor developers in resolving technical, operational and regulatory challenges to enable future demonstration of a diverse set of advanced reactor designs. The risk reduction projects support the development of safe and affordable advanced reactor technologies that can be licensed and deployed over the next 10 to 14 years. The five projects selected for award are:

- Hermes Reduced-Scale Test Reactor by Kairos Power, LLC: Kairos Power will design, construct and operate its Hermes reduced-scale test reactor, which is intended to lead to the development of their commercial-scale fluoride salt-cooled high temperature reactor concept. The NE is working to complete this cooperative agreement in the near term.
- eVinci™ Microreactor by Westinghouse Electric Company (WEC), LLC: WEC will advance the design of a heat pipe-cooled microreactor which uses TRISO fuel for operation. In December 2021,

the cooperative agreement for the WEC award was finalized, and project activities were initiated.

- BWXT Advanced Nuclear Reactor (BANR) by BWXT Advanced Technologies, LLC: BWXT Advanced Technologies will develop a commercially viable transportable microreactor with the design focused on using TRISO fuel particles to achieve higher uranium loading and an improved core design. The NE is working to complete this cooperative agreement in the near term.
- Holtec SMR-160 Reactor by Holtec Government Services, LLC: The Holtec-led project will support early-stage design, engineering and licensing activities to accelerate the development of Holtec's light water-cooled SMR design, and is the only water-cooled design supported through the ARDP. In November 2021, the cooperative agreement for the Holtec award was finalized, and project activities were initiated.
- Molten Chloride Reactor Experiment by Southern Company Services Inc.: The Southern Company Services-led project will support the design, build and operation of the Molten Chloride Reactor Experiment (MCRE) at Idaho National Laboratory (INL). In the first year of the project, the team will focus on completion of the MCRE conceptual design to support development of safety and environmental compliance documents. In September 2021, the cooperative agreement for the Southern Company Services award was finalized, and project activities were initiated.

National Reactor Innovation Center

The ARDP also provides support for the National Reactor Innovation Center (NRIC). The NRIC was authorized by the Nuclear Energy Innovation Capabilities Act (NEICA) of 2017 and was formally established in 2019. The mission of NRIC is to enable and accelerate the development and demonstration of advanced reactors by harnessing the unique capabilities of US national laboratories. In July 2021, the NRIC announced a cost-shared partnership with a team led by General Electric Hitachi to develop three advanced construction technologies that together can reduce the cost of new nuclear build by more than 10%. The project team will leverage promising technologies from other industries that have not been tested within the context of nuclear energy. These include:

- vertical shaft construction, a best practice from the tunneling industry that could reduce construction schedules by more than a year;
- Steel Bricks™, modular steel-concrete composite structures, much like high-tech LEGO® pieces, which could significantly reduce the labor required on site;
- advanced monitoring, coupled with digital twin technology, which can create a 3-D replica of the nuclear power plant structure.

These technologies can be applied to a variety of advanced reactor designs to significantly improve the economics of bringing advanced reactors to market.

Advanced reactor regulatory development

The Advanced Reactor Regulatory Development program, funded through the ARDP, coordinates with the US Nuclear Regulatory Commission (NRC) and industry to address and resolve key regulatory framework issues that directly impact the critical path to advanced reactor demonstration and deployment. Since 2019, the DOE-NE has supported the development of an industry-driven proposal to the NRC for risk-informed and "right-sized" license application content for advanced reactors to reduce regulatory uncertainty and support near-term demonstrations and deployments. In August 2021, this license application content guide was submitted to the NRC for review and potential endorsement. A decision by the NRC on endorsement of the content guide is expected in 2022.

Advanced reactor concepts-20

The DOE-NE selected three awards to support the development of designs that could have a significant impact on the energy market in the mid-2030s or later. In 2021, the cooperative agreements for all of the three advanced reactor concepts-20 (ARC-20) awards were finalized, and project activities were initiated. The three projects selected are:

- Inherently Safe Advanced SMR for American Nuclear Leadership by Advanced Reactor Concepts, LLC: The Advanced Reactor Concepts award will support the development of a 100 MWe pool type SFR design.
- Fast Modular Reactor Conceptual Design – General Atomics: The General Atomics award will support development of a 50 Mwe gas-cooled fast modular reactor.
- Horizontal Compact High Temperature Gas Reactor by the Massachusetts Institute of Technology (MIT): The MIT award will support the development of a modular integrated gas-cooled high temperature reactor.

Versatile test reactor

The versatile test reactor (VTR) project, formally launched in February 2019, is a critical part of the US effort to modernize its nuclear R&D user facility infrastructure. The VTR will provide a leading-edge capability for accelerated testing and qualification of advanced fuels and materials in support of the next generation of nuclear reactors.

Following issuance of the Draft VTR Environmental Impact Statement (EIS) for public review and comment on 31 December 2020, the DOE made revisions and created a comment response document. The subsequent Final VTR EIS has been prepared and will be issued in early calendar year 2022. The Final VTR EIS retains the INL as the VTR's preferred location and examines all environmental factors to support the DOE's final decision. The final decision, known as the Record of Decision, can be issued 30 days after the Final VTR EIS is published. The DOE is working with Congress to ensure that the VTR is appropriately funded in fiscal year 2022 and in a manner that reflects the administration's priority for the advanced reactor demonstration projects and the need to sustain VTR progress on critical systems in order to deliver the critical capabilities that the VTR will provide to support long-term nuclear energy innovation.

Clean hydrogen generation

The NE, in collaboration with other DOE offices, has been working to further the integration of hydrogen production processes with nuclear power plants, including high-temperature steam electrolysis (HTSE). In the past year, the DOE launched the Energy Earthshots Initiative to accelerate breakthroughs of more abundant, affordable, and reliable clean energy solutions within the decade. The first Energy Earthshot – Hydrogen Shot – focuses on reducing the cost of clean hydrogen. In support of the Hydrogen Earthshot, the DOE issued an RFI on viable hydrogen demonstrations, including specific locations, which can help lower the cost of hydrogen, reduce carbon emissions and local air pollution, create good-paying jobs and provide benefits to disadvantaged communities. The DOE also recently announced USD 20 million in funding to demonstrate technology that will produce clean hydrogen energy from nuclear power.

Fiscal year 2022 congressional budget request

In May 2021, the DOE released its FY22 Congressional Budget request, which seeks a record USD 1.8 billion for the NE. This is a historic level of commitment by the Biden-Harris Administration, and it clearly recognizes the role new reactors could play in helping to combat climate change, as well as the need for investments to deploy new technologies to market. The FY22 request includes nearly USD 700 million to help drive innovative US advanced reactor technologies to market within the decade. It includes USD 245 million to support the demonstration of two US reactors[11] in the near future and USD 305 million to support the maturation of additional reactor designs.[12] The request also supports further development of microreactor and SMR technologies.

In the summer of 2021, the DOE-NE received FY22 Congressional Marks from the House and Senate. The Congressional marks also emphasized the importance of advanced reactors in meeting our climate change goals, and provided strong support for advanced reactor development and demonstration programs. Both marks provided USD 245 million for the two ARDP demonstration projects, USD 55 million for the NRIC, USD 15 million for advanced reactor regulatory development, and they included funding for the Advanced Reactor Technologies program, advanced SMR RD&D, TRISO fuel and graphite qualification and the ARDP Risk Reduction awards. Both marks also provided support for a new program, which would ensure the availability of high-assay low-enriched uranium to support advanced reactor demonstration needs. The next step in the budget process requires the House and Senate to resolve any discrepancies via conference prior to finalizing the FY22 appropriation.

Infrastructure Investments and Jobs Act

The Infrastructure Investments and Jobs Act was signed into law on 15 November 2021. The act includes more than USD 62 billion[13] for the DOE to help transition the United States to a clean energy economy, and that includes leveraging the nation's largest single source of clean power—nuclear energy.[14] The act includes USD 6 billion to start a Civil Nuclear Credit program, where owners or operators of commercial US reactors can apply for certification, and competitively bid on credits to help support their continued operations and avoid premature retirements due to financial hardship. The DOE plans to seek public feedback to help set up the program and could start awarding its first credits to US nuclear power plants as early as this fall. The new legislation also allocates USD 8 billion to demonstrate regional clean hydrogen hubs, including at least one hub dedicated to the production of hydrogen with nuclear energy. Initial selections could be awarded before the end of the year. The DOE currently supports four clean hydrogen demonstration projects at commercial nuclear power plants across the country which are also part of the Department's Hydrogen Shot goal[15] to reduce the cost of hydrogen to USD 1 per 1 kilogram in one decade. Finally, the act provided USD 2.477 billion to support the two ARDP demonstration projects.

11. See: www.energy.gov/ne/articles/us-department-energy-announces-160-million-first-awards-under-advanced-reactor.
12. See: www.energy.gov/ne/articles/infographic-advanced-reactor-development.
13. See: www.energy.gov/articles/doe-fact-sheet-bipartisan-infrastructure-deal-will-deliver-american-workers-families-and-0.
14. See: www.energy.gov/ne/articles/5-fast-facts-about-nuclear-energy.
15. See: www.energy.gov/eere/fuelcells/hydrogen-shot.

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Appendix 2. List of abbreviations and acronyms

Generation-IV specific acronyms

AMME-TF	Advanced Manufacturing Materials Engineering Task Force
EG	Experts Group
EMWG	Economic Modelling Working Group
ETWG	Education and Training Working Group
GIF	Generation IV International Forum
ISAM	Integrated safety assessment methodology (GIF RSWG)
MWG	Methodology working group
NEaNH TF	Non-Electric Applications of Nuclear Heat Task Force
PA	Project Arrangement
PD	Policy Director
PG	Policy Group
PMB	Project Management Board
PRPPWG	Proliferation Resistance and Physical Protection Working Group
pSSC	provisional System Steering Committee
RDTF	R&D Infrastructure Task Force
RSWG	Risk and Safety Working Group
SA	System Arrangement
SCWR	Supercritical-water-cooled reactor
SDC	Safety design criteria
SFR	Sodium-cooled fast reactor
SIA	System integration and assessment
SIAP	Senior Industry Advisory Panel
SSC	System Steering Committee
TS	Technical Secretariat

Technical terms, projects and facilities

AF	Advanced fuel
AFA	Alumina forming austenitic
AFR	Advanced fast reactor
AGR	Advanced gas-cooled reactor (US)
ALFRED	Advanced Lead Fast Reactor European Demonstrator
ALLEGRO	The European Gas Fast Reactor Demonstrator Project
AMME	Advanced manufacturing and materials engineering
AMR	Advanced modular reactor
ANSTER	Advanced nuclear technology cost reduction strategies and systematic economic review
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
BWR	Boiling water reactor
CD&BOP	Component design and balance of plant
CEFR	China experimental fast reactor
CFD	Computational fluid dynamics
CFR	Chinese sodium fast reactor
CHP	Combined heat and power
CMVB	Computational methods, validation and benchmarking
COVID-19	Coronavirus-19
DHR	Decay heat removal
dpa	displacements per atom
ECC-SMART	Joint European Canadian Chinese – Small Modular Reactor Technology project
ECS	Energy conversion system
ELFR	European lead fast reactor
ESFR	European sodium fast reactor
ESFR-SMART	European sodium fast reactor – Safety Measeure Assessment and Research Tools
FFDL	Failed fuel detection and location
FLIBE	mixture of lithium and beryllium fluoride (BeF_2)
FLiNaK	salt mixture of lithium fluoride, sodium fluoride and potassium fluoride
FOAK	First-of-a-kind
GFR	Gas-cooled fast reactor
GW	Gigawatt
GWD/MTHM	Gigawatt days per metric ton of heavy metal
HEEP	Hydrogen Economic Evaluation Programme (IAEA)
HINEG	High intensity D-T fusion neutron Generator
HTGR	High-temperature gas-cooled reactor
HTR	High-temperature gas-cooled reactor
HTR-PM	High-temperature gas-cooled reactor – pebble-bed module
HTSE	High-temperature steam electrolysis
HTTR	High-temperature test reactor

IRRS	Integrated Regulatory Review Service (IAEA)
ISI&R	In-service inspection and repair
JRC	Joint Research Centre (EU)
JSFR	Japanese sodium-cooled fast reactor
KALIMER	Korea advanced liquid metal reactor
LBE	Lead-bismuth eutectic
LES	Large eddy simulation (model)
LFR	Lead-cooled fast reactor
LOF	Loss of flow
LWR	Light water reactor
MA	Minor actinides
MAT	Materials
MBIR	Russian multipurpose fast neutron research reactor
MCRE	Molten chloride reactor experiment
MEACTOS	Mitigating environmentally-assisted cracking through optimisation of surface condition
MMR	Micro modular reactor
MOSART	Molten salt actinide recycler and transmuter
MoU	Memorandum of understanding
MOX	Mixed oxide (fuel)
Mpa	megapascal
MSFR	Molten salt fast reactor
MSR	Molten salt reactor
MW	Megawatt
MYRRHA	Multi-purpose hybrid research reactor for high-tech applications
NEST	Nuclear Economics Support Tool (IAEA)
NUWARD	French PWR SMR Project
ODS	Oxide dispersion-strengthened
PGSFR	Prototype Generation IV sodium-cooled fast reactor
PIE	Post-irradiation examination
PWR	Pressurized water reactor
RANS	Reynolds-averaged Navier-Stokes (model)
R&D	Research and development
RD&D	Research development & demonstration
SAMOSAFER	Safety assessment for fluid-fuel energy reactors
SEM	Scanning electron microscopy
SiCf/SiC	Silicon carbide fiber reinforced silicon carbide
S-I	Sulfur-iodine (process)
SMR	Small modular reactor
SNETP	Sustainable Nuclear Energy Technology Platform
SNF	Spent nuclear fuel
SRP	System research plan
SSTAR	Small, secure, transportable, autonomous reactor
STELLA-2	Large-scale Sodium Integral Effect Test Facility
TH-U	Thorium-uranium
TMSR	Thorium molten salt reactor (China)
TRISO	Tri-structural isotropic (nuclear fuel)
UCO	Uranium oxycarbide
UOX	Uranium oxide
VHTS	Very high-temperature system
VTR	Versatile test reactor (US)
VVER	Russian light water power pressurized reactor model
WGSAR	Working Group on the Safety of Advanced Reactors (NEA)

Organizations, companies and agencies

ANL	Argonne National Laboratory (US)
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire (France)
CAS	Chinese Academy of Science
CEA	Alternative Energies and Atomic Energy Commission (France)
CIAE	China Institute of Atomic Energy
CNL	Canadian Nuclear Laboratories
CNNC	China National Nuclear Corporation
CNSC	Canadian Nuclear Safety Commission
DOE	Department of Energy (US)
EDF	Electricité de France
EU	European Union
IAEA	International Atomic Energy Agency
INET	Institute of Nuclear and New Energy Technology (China)
INL	Idaho National Laboratory (US)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre (Euratom)
KAERI	Korea Atomic Energy Research Institute
KIT	Karlsruhe Institute of Technology (Germany)
NCBJ	National Centre for Nuclear Research (Poland)
NE	Office of Nuclear Energy (US)
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission (US)
OECD	Organisation for Economic Co-operation and Development
OPG	Ontario Power Generation (Canada)
ORNL	Oak Ridge National Laboratory (US)
PSI	Paul Scherrer Institute (Switzerland)
WEC	Westinghouse Electric Company

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Appendix 3. Selection of GIF publications (2021)

General papers

GIF Newsletters nos. 1 to 6, and 1 Special Summer Edition, available at: www.gen-4.org/gif/jcms/c_122378/newsletters-archive.

Lead-cooled fast reactor

Adamov Ye.O., A.V. Kaplienko, V.V. Orlov, V.S. Smirnov, A.V. Lopatkin, V.V. Lemekhov, A.V. Moiseev (2020), "BREST Lead-Cooled Fast Reactor: from the Concept to the Technology Implementation", *Atomic Energy*, Vol. 129, No. 4, pp. 185-194.

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THE GENERATION IV INTERNATIONAL FORUM

Established in 2001, the Generation IV International Forum (GIF) was created as a co-operative international endeavour seeking to develop the research necessary to test the feasibility and performance of fourth generation nuclear systems, and to make them available for industrial deployment by 2030. The GIF brings together 13 countries (Argentina, Australia, Brazil, Canada, China, France, Japan, Korea, Russia, South Africa, Switzerland, the United Kingdom and the United States), as well as Euratom – representing the 28 European Union members − to co-ordinate research and development on these systems. The GIF has selected six reactor technologies for further research and development: the gas-cooled fast reactor (GFR), the lead-cooled fast reactor (LFR), the molten salt reactor (MSR), the sodium-cooled fast reactor (SFR), the supercritical-water-cooled reactor (SCWR) and the very-high-temperature reactor (VHTR).

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