GEN IV International Forum

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ANNUAL REPORT 2022

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Foreword from the Chair

The Generation IV International Forum (GIF) 2022 Annual Report highlights the achievements and progress in the Generation IV (Gen IV) reactor systems research and development.

This year I was grateful to preside over the first in-person Policy Group meeting for GIF members since the COVID-19 pandemic began and welcomed representatives from ten GIF member countries in Sydney, Australia. While there was a virtual component of this meeting, the importance of meeting in person was significant, providing the time needed for discussions to move GIF initiatives forward and the opportunity for side conversations. The GIF has had a productive year, with many accomplishments in the form of publications, webinars and outreach, in addition to the technical progress made on the six GIF systems. Under the leadership of the GIF Policy Director, Fiona Rayment, the GIF refreshed its Policy Statement on Governance and created a new Policy Statement on Industry Engagement.

A significant achievement in 2022 was hosting the first GIF Industry Forum that was co-located with the Generation IV and Small Reactors fourth conference (G4SR-4) in Toronto, Canada. The GIF Industry Forum was co-organized with the Canadian Nuclear Society with the objective for GIF members to engage with industry and vendors and the small modular reactor (SMR) community. Combining the two events successfully provided opportunities for the private and public sectors to discuss and collaborate to accelerate demonstration of Gen IV systems. The technical sessions on advanced nuclear energy systems and workshops on cross-cutting applications were well attended by nuclear developers and industry sectors seeking to benefit from low-carbon electricity and heat to meet global carbon reduction goals. My appreciation goes to Bob Hill, Sylvestre Pivet, Philippe Guiberteau, Gina Abdelsalam and John Kelly for coordinating this first Industry Forum.

Many thanks are extended to Hideki Kamide for developing and executing one of the forum highlights – Innovation Night. This was a special social event that showcased early career researchers and their experience with Gen IV/SMR systems. Winners shared their career paths, discussed the challenges they faced in moving a concept forward and highlighted the rewards of working in the advanced nuclear field. Feedback received from industry has been very positive, with requests to continue the GIF Industry Forum every two years.

Looking forward, the GIF will continue to provide global leadership and support through the GIF Framework to ensure that safe, secure and sustainable nuclear energy is available for the future. Over the next decade, the GIF will continue to strengthen Gen IV systems to support areas such as non-electric applications, climate change, technical readiness, regulatory harmonization and economic improvement in support of commercial deployment of advanced systems. In addition, the GIF plans to enhance industry engagement through the GIF Senior Industry Advisory Panel to focus on the transition of research and development to demonstration and commercial deployment.

Alice Caponiti
GIF Chair

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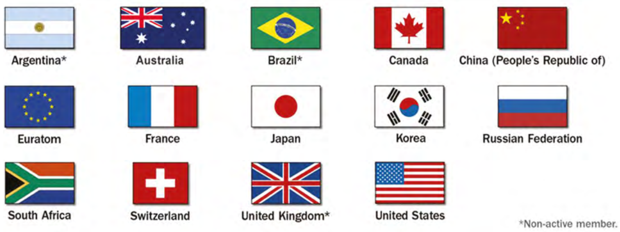
1 GIF membership and organization highlights

In 2022, the structure and organization of the Generation IV International Forum (GIF) remained the same. The GIF is composed of 14 member countries, all of which are signatories of its founding documents.

Figure 1.1. GIF membership



This chapter focuses only on major changes that occurred in 2022. More detailed information on the GIF membership and organization are available at: www.gen-4.org/gif/jcms/c_59452/governance-stucture.

GIF organization

Figure 1.2 summarizes the GIF's global governance structure and main bodies.

Figure 1.2. GIF governance

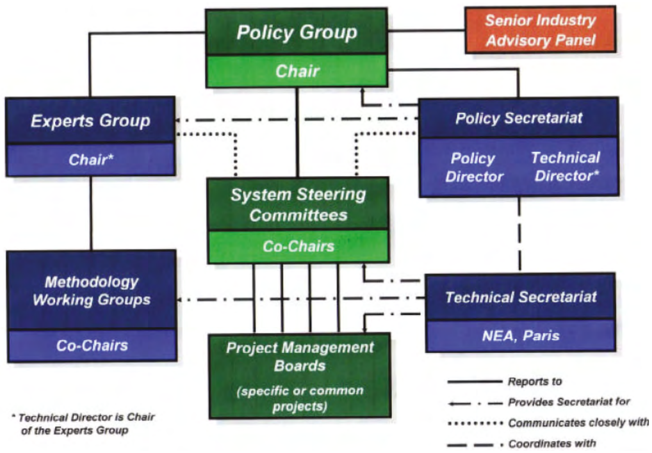
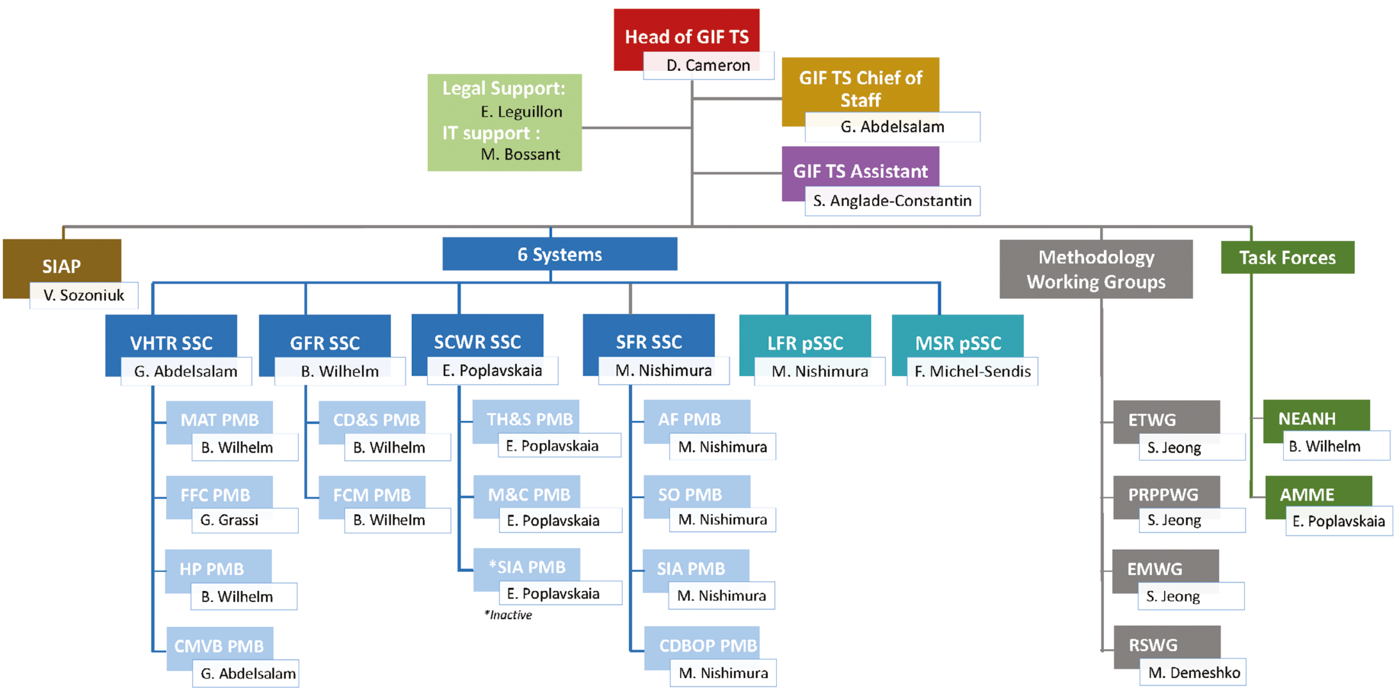


Figure 1.3. Structure of the GIF Technical Secretariat (located at the NEA) as of October 2022



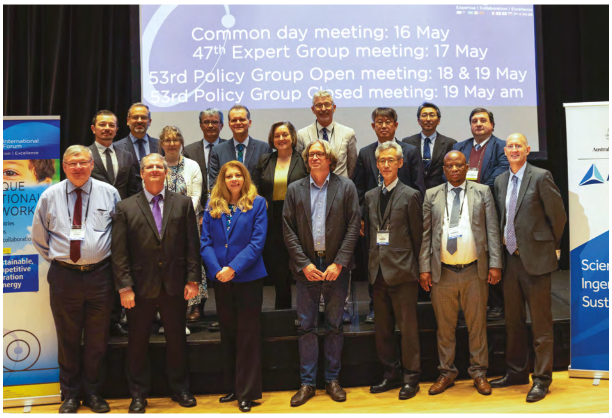


Figure 1.4. GIF Policy Group photo in Sydney (May 2022) and Toronto (October 2022)

In January 2022, the GIF Policy Group thanked the outgoing Chair Hideki Kamide of the Japan Atomic Energy Agency (JAEA) and acknowledged the notable accomplishments of the previous three years under his leadership. Alice Caponiti, Deputy Assistant Secretary for Reactor Fleet and Advanced Reactor Deployment in the United States Department of Energy's Office of Nuclear Energy, was elected GIF Chair for the period 2022-24.

With the aim of ensuring the continuity of operational requirements within the GIF Technical Secretariat, several staff changes occurred at the OECD Nuclear Energy Agency (NEA) in 2022. Figure 1.3 shows the resulting structure as of October 2022.

Throughout 2022 and in light of the recent improvements with regards to the COVID-19 pandemic, GIF members resumed in-person meetings and managed to make substantive progress in their work. The NEA continued to provide support to the GIF technical bodies in charge of developing the different systems, to the methodology working groups, task forces and Senior Industry Advisory Panel (SIAP). The NEA also maintained the GIF website and organized several GIF meetings, including the 48th Experts Group and the 54th Policy Group meetings in Toronto, Canada, in October 2022.

Strategic priorities, perspectives and objectives

Recognizing the evolving research, development and demonstration (RD&D) needs and collaboration with new partners, the GIF has reviewed and revised its Policy Statement on Governance. The updated policy statement highlights the important role of industry in enabling mission-driven R&D and recognizes the emerging role of the SIAP to advise the Policy Group on strategic and longer term challenges. In addition, the statement now provides greater transparency on roles and responsibilities across the GIF and the terms of reference have been revised for all GIF leadership positions.

In 2022, the GIF Technical Secretariat pursued its efforts to improve GIF communications and outreach to industry, researchers, policy makers, government stakeholders and the general public to reinforce the GIF position as a leading collaborative organization at the international level, with technical expertise focused entirely on fourth generation nuclear energy systems. The GIF's priorities in the near-term focus on accelerating the readiness of fourth generation nuclear systems to meet net zero targets.

In this context and with the support of the NEA, a GIF forum with industry partners was organized in October 2022 in Toronto, Canada, co-located with the International Conference on Generation IV and Small Reactors (G4SR-4). The forum focused on supporting the transition from R&D to demonstration and deployment of Gen IV reactor systems, as well as on supporting initiatives to develop a talent pipeline to run those systems. The GIF Technical Secretariat led the Organizing Committee and executed all meeting logistics for the debut GIF Industry Forum.

Ongoing engagement between the GIF and the International Atomic Energy Agency (IAEA) continues to play an important role in advancing Gen IV technologies. The GIF and the IAEA first established an annual interface meeting in 2003 with the objective of exchanging information, progress, status and future plans in twelve areas of shared interest related to RD&D and technology innovations. The GIF and the IAEA are currently implementing new approaches to further enhance collaboration between the two organizations. This includes providing more regular updates (biannual) to the joint collaboration matrix document as well as focusing annual interface meetings to "deep dive" discussions on one to three topics.

Finally, the GIF Policy Group has begun a dialogue on extending the GIF Framework Agreement, which is due to conclude in early 2025. The Policy Group is working closely with the GIF Technical Secretariat to produce an updated draft Framework Agreement for consideration by GIF members.

Perspectives and objectives for 2023 are to continue to assume and improve the GIF Technical Secretariat's services as follows:

- Continuing to develop GIF communications (technical, external, and internal) to increase the GIF's engagement with industry while enhancing support to GIF members and bodies.
- Providing assistance in reviewing or renewing GIF procedures, contracts and agreements, including the GIF Framework Agreement, system and project arrangements, and memoranda of understanding.
- Maintaining necessary coordination and, as appropriate, close cooperation with other international organizations active in the field of Gen IV nuclear reactors (e.g. the NEA, the IAEA and the World Nuclear Association).
- Developing cross-cutting projects within the GIF and with the NEA.
- Remodelling the GIF website to enhance its performance and usability, improve collaboration between members, and enhance the GIF's brand awareness and website ranking.
- Developing a special project to create a web-based searchable inventory of the GIF's information assets (e.g. videos, databases, expert contacts and intellectual property) that would be useful to industry and the broader private sector; the website would also provide information on how to access the information assets. This inventory should increase utility and impact with industry and support efforts to accelerate demonstration and deployment of Gen IV reactor concepts.

Diane Cameron
Head of the GIF Technical Secretariat

Fiona Rayment
GIF Policy Director

2 GIF outlook and current initiatives

Today, several of the six Generation IV (Gen IV) systems are entering the demonstration phase, which presents challenges and opportunities for the Generation IV International Forum (GIF) community. In general, the research and development (R&D) infrastructure (both expertise and facilities) used to develop advanced reactor technology remains useful for Gen IV demonstration and deployment. However, key R&D topics need to be informed and prioritized by implementation and operational experience. Therefore, an intimate working relationship with the advanced reactor industry working to license, construct and operate advanced reactors is vital to ensure the continued relevance of Gen IV collaborations.

Furthermore, demonstration systems often rely on low-risk (high technical maturity) design choices, even if new technology options offer significant performance improvements. Thus, a robust R&D infrastructure, where promising design features can be quickly matured, needs to be sustained to enable future innovations. This provision for future improvements is critical to enable widespread deployment (after the demonstration phase) of robust, high-performance Gen IV advanced reactors.

In 2022, **GIF activities focused on improved engagement with the private sector**. Specific collaboration opportunities have been identified and initiated. In addition, opportunities have been pursued to showcase the GIF's collaborative research achievements and their impact on advanced reactor sustainable development goals.

Policy Statement for Industry Engagement

The GIF was launched in 2001 as a multilateral collaboration focused on R&D of the next generation of advanced reactor technologies. It seeks to identify, assess and, ultimately, contribute to the development of new nuclear energy systems.

From its inception, the GIF has always had the ability to engage with the nuclear industry through its existing arrangements and processes. Although the GIF already engages with the nuclear industry, and is a regular host of knowledge-sharing meetings on Gen IV technologies, there is recognition that further action in a more enhanced and structured manner is required to drive the implementation of advanced nuclear technologies that are ready to deploy within the challenging time frame of the early 2030s. Recognizing that the processes for this engagement are not always clear, the GIF has agreed on a 2022 Policy Statement on Engagement with the Nuclear Industry, reemphasizing the ability of the nuclear industry to engage with and benefit from the work carried out by the GIF.

As an organization created through an international treaty, it is important to have rules of engagement so that interactions with industry abide by the treaty. The policy statement provides these, bringing together in one place the different procedures and processes by which industry can engage with, and benefit from, the GIF's work. The nuclear industry will ultimately be the conduit through which Gen IV reactors will become a reality, and it is through the nuclear industry that the GIF can contribute to the demonstration and deployment of new nuclear energy systems.

GIF Industry Forum

To promote engagement between industry and GIF experts, it was decided to replace the triennial GIF Symposium with a targeted Industry Forum. The meeting was co-located with the 4th International Conference on Generation IV and Small Reactors,[1] G4SR-4, to attract a broad audience of advanced reactor private industry. The Industry Forum's technical programme was synchronized to allow participation in both meetings. The GIF leadership provided a plenary session on 4 October. The historical framework, collaboration agreements and goals were identified for GIF collaborations. The envisioned role

1 G4SR-4 Conference organized by the Canadian Nuclear Society. See: www.g4sr.org.

of Gen IV technologies to enable decarbonization was presented. Recent achievements of GIF collaboration projects were highlighted and the contributions and benefits to industry identified. The plenary session concluded with a vision for GIF collaboration with the nuclear industry, explaining the key principles and instruments for engagement.

The GIF Industry Forum was attended by 231 registered participants, including 67 from the private sector. GIF experts also interacted with the over 500 participants in the G4SR-4 Conference. The technical programme for the Industry Forum consisted of ten sessions:[2]

- non-electric applications of nuclear – full-day workshop;
- technical session on economic challenges and opportunities of Gen IV reactors;
- technical session on artificial intelligence for nuclear;
- technical session on very high-temperature reactors bridging the gap to commercial demonstration;
- technical session on sodium-cooled fast reactor demonstrations;
- technical session on molten salt reactor demonstrations;
- technical session on lead-cooled fast reactor demonstrations;
- two sessions on Gen IV reactor licensing: regulatory readiness plans and licensing experiences;
- Gen IV innovators panel: how to mature new concepts and technology;
- international knowledge management and preservation panel.

There was widespread interest at the workshop on 3 October in non-electric applications; over 150 private and public experts participated in the session organized by the GIF Non-Electric Applications of Nuclear Heat Task Force. GIF experts also participated in G4SR-4 Workshops on Advanced Materials and Integrated Safety Assessment Methodology. GIF technical sessions were attended by 25-100 participants, including active engagement by industry experts; private sector participants comprised roughly half of the panel/speaker roster and audience. Each session included two primary themes: 1) highlights of recent achievements from the GIF R&D collaborations; and 2) opportunities for collaboration and synergy with industry Gen IV demonstration efforts.

Industry engagement initiatives

Feedback from the GIF Industry Forum indicated limited awareness in the private sector of GIF collaborations and products. A wide range of collaboration topics were identified, ranging from technology R&D to design and methodology sharing. Industry experts indicated collaboration products need to be provided within a short time frame. Sustained R&D efforts generated some interest, but the current situation is demonstration projects with tight deadlines.

Results and feedback from the GIF Industry Forum were discussed at the October Experts Group and Policy Group meetings. The Senior Industry Advisory Panel was asked to solicit specific feedback on collaboration mechanisms and communication of R&D accomplishments. Current practices for industry nomination to the GIF Working Group and task forces were reviewed; the system and project arrangements would impose membership requirements for direct inclusion. It was reiterated that timing collaborations to provide quick results is paramount for results to be useful for industry. To explore opportunities to streamline GIF mechanisms and accelerate GIF collaboration with industry, it was agreed to pursue focused efforts on four "case studies" that range from information sharing to collaboration on joint R&D tasks:

- Industry Involvement in GIF task forces. Pursue nomination of industry experts as members and continue to actively engage the private sector through workshops and other mechanisms.
- Gen IV demo examples for the Safety, Security and Safeguards (3S) Project being pursued jointly by the Proliferation Resistance and Physical Protection Working Group and the Risk and Safety Working Group. Solicit industry teams to provide design and implementation examples to test the 3S methodology.
- Sharing of *Materials Handbook Data* from the very high-temperature reactor (VHTR) Materials Project. Several private sector companies have expressed interest in the high temperature materials testing data compiled for the GIF VHTR Materials Project. Guidelines need to be clarified and streamlined to promote sharing of important testing data with emerging industry in member countries.
- New sodium-cooled fast reactor (SFR) technical project on sodium thermal fluid dynamic validation. Several private sector companies have expressed interest to collaborate with GIF members. The goal is to accelerate the GIF project arrangement process by focusing on a small membership project with a short time frame to produce demonstration-driven products.

In summary, enhancing GIF engagement with the advanced reactor industry was a major theme in 2022. A policy statement was issued to emphasize the use of various GIF interfaces. The GIF Industry Forum provided unique opportunities for GIF experts to engage with private sector experts and discuss opportunities for collaboration. The forum feedback resulted in the identification of several specific avenues that will be pursued in 2023 to promote and accelerate GIF-industry engagement.

Robert Hill
GIF Technical Director

2 Summaries of each technical session are available at: www.gen-4.org/gif/jcms/c_206200/industry-forum-2022.

3 System summaries

Gas-cooled fast reactor

Signatories of the System Arrangement for collaboration on gas-cooled fast reactor (GFR) research and development (R&D) are the following 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 Kyoto University as members.

The R&D collaboration activities pursued in the two GFR technical projects focus on the ALLEGRO gas-cooled fast reactor demonstration concept. The GIF projects have scope for conceptual design, safety analysis, testing of start-up fuel and core materials, and fuel performance modelling.

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 proposed as a longer term alternative to liquid metal-cooled fast reactors. The main advantages of GFRs besides allowing adoption of the closed fuel cycle are:

- high operating temperature, allowing increased thermal efficiency and high temperature heat for industrial applications similar to the very high temperature reactor;
- a chemically inert and a non-corrosive coolant (helium);
- a single phase (no boiling) coolant (helium);
- relatively small (albeit positive) helium coolant void reactivity coefficient;
- helium doesn't dissociate or activate;
- helium is transparent, facilitating in-service inspection and repair, as well as fuel handling.

High-outlet temperature (850°C) for high thermal efficiency and hydrogen production, and a direct cycle for compactness, are key reference objectives. Unit power will be considered in the range of 200 megawatt electrical (MWe) modular size, up to larger 1 500 MWe.

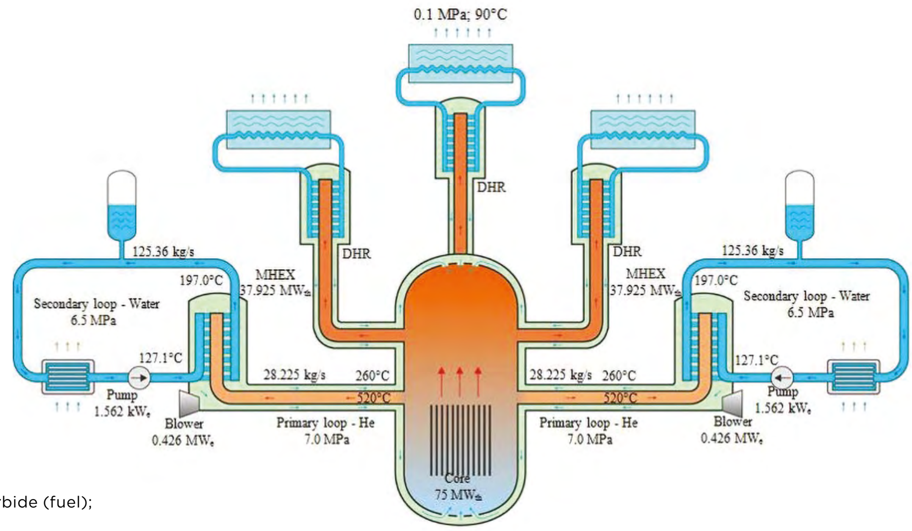
The objective of high fuel burn-up, together with actinide recycling, results in spent-fuel characteristics (isotopic composition) that are unattractive for handling. Consensus has been reached in the project to minimize feedstock usage with a self-sustaining cycle, which requires only depleted or reprocessed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

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

ALLEGRO Demonstration Project Overview

The objectives of ALLEGRO are to demonstrate the viability 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. The demonstration of the GFR technology assumes that the basic features of the GFR commercial reactor can be tested in the 75 MWth ALLEGRO reactor.

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 3.1. ALLEGRO design overview

The original design of ALLEGRO consists of two helium primary circuits, three decay heat removal (DHR) loops integrated into a pressurized cylindrical guard vessel (Figure 3.1). The two secondary gas circuits are connected to gas-air heat exchangers. The ALLEGRO reactor would serve not only as a demonstration reactor, hosting GFR technological experiments, but also as a test pad to:

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

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

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

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

R&D objectives and technology innovations

All members of the GFR System Arrangement (Euratom, France and Japan) are active contributors to the SafeG project. SafeG has received funding from the Euratom Horizon 2020 Nuclear Fission and Radiation Protection Research programme (NFRP-2019-2020) under grant Agreement No. 945041; the project is still in its first half and most of the expected outcomes are still under development. 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 electricity and the industrial process heat generation technology through the following main objectives:

1. Strengthen safety of the GFR demonstrator ALLEGRO through the use of innovative technologies, materials and systems. The most important areas of ALLEGRO safety improvements addressed by the SafeG project are:

a. to solve remaining open questions in residual heat removal in accident conditions, leading to practical elimination of severe accidents through innovative design of the reactor core, diversified ways of passive reactor shutdown, passive decay heat removal systems and instrumentation;

b. to strengthen the inherent safety of the key reactor components through a review of obsolete material and technology reference options, selection of innovative options, and designs based on these innovative options.

2. Review the GFR reference options in materials and technologies.
3. Adapt GFR safety to changing needs in electricity production worldwide – with an increased and decentralized portion of nuclear electricity.
4. Bring in students and young professionals, boosting interest in GFR research.

The main ambition of the SafeG project is to bring cutting-edge technological innovations to the concept of GFR and implement them in its demonstrator ALLEGRO. The SafeG project has significant potential to come up with solutions that will go far beyond the state-of-the-art of the GFR technology. In some special cases, specifically the high-temperature material testing and fuel qualification options, the outcomes of the project will represent a breakthrough in the level of knowledge in these areas in a pan-European context.

The SafeG project aims to bring the design and safety of this reactor a considerable step further, mainly in the following areas:

- Core safety – significant progress beyond the state of the art of GFR core safety has already been made (start up core optimization was completed). Further work will include optimization of reactivity feedback coefficients and irradiation capabilities of the ALLEGRO core designs.
- Automatic shutdown system – the current design will be updated, using state-of-the-art knowledge possessed by the consortium members who will work on this task.
- Instrumentation – instrumentation of GFRs will be solved with unprecedented depth of the involved analyses, assessment of the possible use of advanced measuring technologies and techniques, and is expected to bring beyond-state-of-the-art ideas.
- Decay heat removal system – so far, decay heat removal for GFRs has been solved in a very similar way for all the reference concepts. Within SafeG, effort will be put into developing an innovative decay heat removal solution based on cutting edge technology – supercritical CO_2 cycles.

Experiments that will be carried out in the framework of analysing compatibility of materials will bring first-of-a-kind results of structural materials behaviour in $He\text{-}N_2$ mixtures at very high temperatures. Application of these results far exceeds applications in GFR and nuclear in general.

Fuel qualification assessment of innovative advanced fuel for GFRs is almost completed. It will provide unique insight into this important issue that can serve as a basis for a general methodology of advanced fuels qualification for advanced nuclear reactors in Europe in the future.

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

Lead-cooled fast reactor

The following GIF members are participating in the GIF memorandum of understanding for collaboration on lead-cooled fast reactor (LFR) R&D: 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, this section summarizes the highlights for the development of LFRs in GIF member countries and entities, as shared within the GIF collaboration.

Main characteristics of the system

The GIF has identified the LFR as a technology with great potential to meet the needs of both remote sites and central power stations, fulfilling the four main goals of the GIF. In the technology evaluations of the *Generation IV Technology Roadmap* (2002), and its update in 2014, the LFR system was ranked at the top in terms of sustainability (i.e. a closed fuel cycle can be easily achieved), and in proliferation resistance and physical protection. It was also assessed as good in relation to safety and economics. Safety was considered to be enhanced by the choice of a relatively inert coolant.

Gen IV LFR concepts include three reference systems: 1) a large system rated at 600 MWe – the European lead fast reactor, 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 size – the US small secure transportable autonomous reactor, which features a very long core life (see Figure 3.2 and Table 3.1). 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 coordination of efforts carried out among participating countries.

R&D objectives

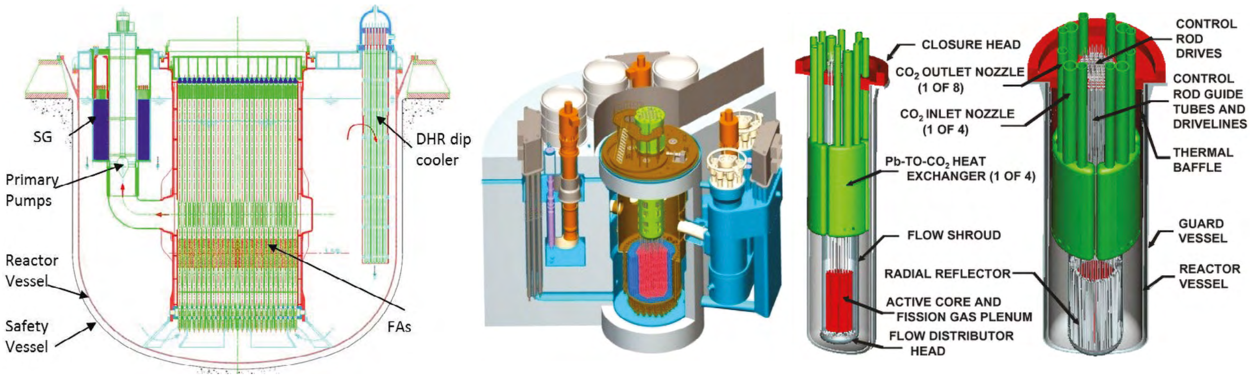
The LFR System Research Plan developed within the GIF is based on the use of molten lead as the reference coolant and lead-bismuth eutectic (LBE) as the back-up option. Given the R&D needs for fuel, materials and corrosion-erosion control, the LFR system is expected to require a two-step industrial deployment. In the first step, reactors operating at relatively modest primary coolant temperatures and power densities would be deployed by 2030 and higher performance ones by 2040. Following the reformulation of the GIF LFR pSSC in 2012, the System Research Plan 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 defining project arrangements once the decision of a transition from the present memorandum of understanding status to a system arrangement is engaged.

Table 3.1. Key design parameters of the GIF lead-cooled fast reactor reference systems

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	535	567
Secondary cycle	Superheated steam	Water-steam	Supercritical CO_2
Net efficiency (%)	42	42	44
Turbine inlet pressure (bar)	180	170	200
Feed temperature (°C)	335	340	402
Turbine inlet temperature (°C)	450	505	553

Notes: MWt: megawatt thermal; Mwe: megawatt electrical; T: temperature.

Figure 3.2. GIF lead-cooled fast reactor reference systems: The European lead fast reactor, the BREST and the small secure transportable autonomous reactor



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

Technical highlights – provisional System Steering Committee activities

Two meetings of the GIF LFR pSSC took place in 2022. The first was held as a videoconference on 5-6 April. The second was held at the Italian National Agency for New Technologies (ENEA) Brasimone Research Centre (Camugnano, Italy) on 30 August-1 September. These meetings were characterized by the presentations of the status of activities in memorandum of understanding signatories with descriptions of their respective national LFR programmes related to IAEA activities (e.g. LFR benchmarks and code validation), and joint work with the NEA Working Group on the Safety of Advanced Reactors on benchmark specifications for the validation of LFR codes.

As part of the Gen IV Industry Forum, a special LFR Demo Session was held, with presentations from GIF LFR pSSC, ENEA, the Belgian Nuclear Research Centre (SCK-CEN), Ansaldo Nucleare, Westinghouse, LeadCold and Newcleo. The objective of this session was to provide an overview of the main LFR concepts under development by national laboratories and the private sector, highlighting the recent progress being made towards commercialization.

The pSSC has prepared the following reports over the past few years: *Lead-cooled Fast Reactor (LFR) System Safety Assessment*, issued in June 2020; *GIF Lead-cooled Fast Reactor: Proliferation Resistance and Physical Protection White Paper*, issued in October 2021; and *Safety Design Criteria for Generation IV Lead-cooled Fast Reactor System*, issued in March 2021. In 2022, the IAEA and the NEA Working Group on the Safety of Advanced Reactors provided comments on the safety design criteria report; these revisions are being analyzed.

National LFR demonstration and development highlights

In Russia, the BREST-OD-300 innovative LFR is developed as a pilot demonstration prototype for base-type commercial reactor facilities of the future nuclear power industry with the closed nuclear fuel cycle. Experimental justification of components, elements and equipment of the reactor was carried out using small- and medium-scale mock-ups and pilot models. Verified and certified software was used for computational design justification. The developed design documentation makes it possible to proceed to the construction of the BREST-OD-300 power unit with concurrent completion of activities aimed at bringing safety justification documentation into conformity with current common and developed specific standards for lead-cooled reactor facilities. The construction license from Rostekhnadzor was issued in February 2021 and the construction of the nuclear power plant with the BREST-OD-300 LFR began on 8 June 2021. In 2022, the construction continued in accordance with the adopted road map (see Figure 3.3).

The BREST-OD-300 reactor is being created as one of the most important components of the Pilot Demonstration Energy Complex operating in a closed nuclear fuel cycle, together with collocated modules for fuel fabrication, refabrication and reprocessing of spent fuel. In addition to operation (power generation), the most important task is the implementation of the R&D programme at the reactor. Various studies and life tests of components and equipment with irradiation experiments in both a lead coolant and a fast neutron spectrum will be carried out.

Source: Rosatom.

Figure 3.3. Construction of the BREST-OD-300 power unit as part of the Pilot Demonstration Energy Complex

As part of the concept of small modular reactors, a project for the SVBR-100 reactor plant is being developed in Russia. The project foresees the construction of a pilot power unit with such a reactor at the site near Dimitrovgrad city. The choice of power level (100 MW) is due to transportability by rail, which ensures a fully factory manufactured reactor of high quality. Mastered parameters of the coolant ensure the use of traditional stainless steels (18% Cr, 9% Ni) for the reactor vessel. Serial 4-6 units (or more) nuclear power plants with modular reactors are focused on a wide range of tasks, from regional energy and heat supply to desalination of sea water or hydrogen production. The R&D performed is mainly aimed at issues related to the justification of the design of the core elements, as well as justification of the service properties of the materials used, including a study of the effects of irradiation. A significant amount of work has been done related to the substantiation of the design of the fuel rods, refuelling equipment and primary pumps. Necessary documentation has been developed for testing models of the steam generator and drive mechanisms of control rods. In preparation for obtaining a license for the construction, proposals have been developed for the regulatory definition of the maximum design limit for fuel rods.

In Japan, reactor design studies have been performed for several lead and lead-bismuth eutectic-cooled fast reactors. The study is performed at the Tokyo Institute of Technology on the feasibility of breed-and-burn fast reactors with lead or lead-bismuth eutectic coolant. The breed-and-burn fast reactors need natural uranium or depleted uranium only as the fuel. The fertile is converted into fissile material in the core and consumed by fissions. It can achieve high burnup without reprocessing. It was shown possible to design a rotational fuel-shuffling breed-and-burn fast reactor (RFBB) using nitride fuel and lead coolant. In the concept, the reactor core design and the fuel assemblies design are essentially the same as conventional LFRs. It is possible to realize the equilibrium condition of breed-and-burn by the application of the rotational fuel shuffling. to the achieved burnup of 202 MWd/kg-HM with natural uranium loading in a small 750 MWt RFBB, supports both effective utilization of natural resources and spent fuel minimization without reprocessing.

In Europe, activities related to LFR mainly focus on five projects: 1) the development of the multi-purpose hybrid research reactor for high-tech applications research infrastructure, which is being carried out by SCK CEN in Mol (Belgium) and aims to demonstrate an accelerator-driven system and support the development of fast neutron spectrum Gen IV systems; 2) R&D activities for the construction of an LFR demonstrator in Romania, i.e. the Advanced Lead Fast Reactor European Demonstrator project; 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; 4) R&D activities for the construction of an LFR demonstrator in Sweden (Oskarshamn), i.e. SEALER (Swedish Advanced Lead Reactor), which is designed for commercial power production in a highly compact format; 5) R&D activities carried out in France, Italy and the United Kingdom in collaboration with several EU organizations for the development of the newcleo LFR concept. newcleo is presently investing up to EUR 50 million to build-up a large-scale research infrastructure in collaboration with ENEA by the Brasimone Research Centre (Italy). Finally, newcleo is presently working to set-up a MOX fuel factory in Europe to complement its commercial LFR programme; the programme is fully funded by private investments.

In parallel, several ongoing European collaborative projects are ongoing (EURATOM co-funded initiatives, i.e. ANSELMUS, INNUMAT, PIACE, PASCAL, PATRICIA, PUMMA, TANDEM and HARMONSE) 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 the United Kingdom, the AMR Program is almost completed. Funded by the UK Department of Business, Energy and Industrial Strategy, it supports LFR development with GBP 10 million. The programme, led by Westinghouse, is supported by Ansaldo Nucleare and the ENEA in Italy; and Jacobs, Ansaldo Nuclear, the University of Manchester, the University of Bangor, the Nuclear National Laboratory, the Nuclear Advanced Manufacturing Research Centre and others in the United Kingdom. The programme is devoted to material and coolant chemistry, technology and fuel development, code validation, innovative manufacturing, and supply chain implementation. A number of new facilities have been constructed to support the programme (Figure 3.4).

In Korea, the conceptual design of MicroURANUS has been completed by a university-industry consortium with Ulsan National Institute of Science and Technology leadership with the financial support of the Ministry of Science, information and communications technology (ICT) and Future Planning, and the Ulsan metropolitan city government. Two sets of conceptual designs have been developed, both with 20 MWe power rating and lead-bismuth eutectic coolant. In the first conceptual design for near-term application for both electricity and hydrogen production in a near-shore submerged fixed-platform, the core consists of cylindrical fuel rods in hexagonal lattice containing UO_2 pellets with about 13% enrichment on average and 15-15Ti cladding with a corrosion-resistant overlay of alumina forming austenitic stainless steel. The second conceptual design for long-term application for ship propulsion has an inverted core design with identical composition and neutronic characteristics of core. Designs of reactor vessel, containment vessel and the balance of plant based on superheated steam turbine cycle are also identical for both sets of conceptual designs, aiming at maritime applications.

The deterministic safety analysis for both conceptual designs showed that it is feasible to practically eliminate severe accidents. Corrosion-resistant alumina-forming austenitic alloy has been developed at Ulsan National Institute of Science and Technology and tested in laboratories at elevated temperature using proton ion and heavy-ion accelerators to confirm adequate performance for the entire 40-year life without significant oxidation and swelling. The multi-unit electromagnetic pumps located outside its containment vessel efficiently drive up to 50% of full flow rate while natural circulation covers the balance. Agile load follow operation is shown to be feasible with the benefit of the electromagnetic pumps. In 2023, Korea National Research Foundation will assess the results of the conceptual design study to examine the viability of the standard design and pre-license engagement in the next phase.

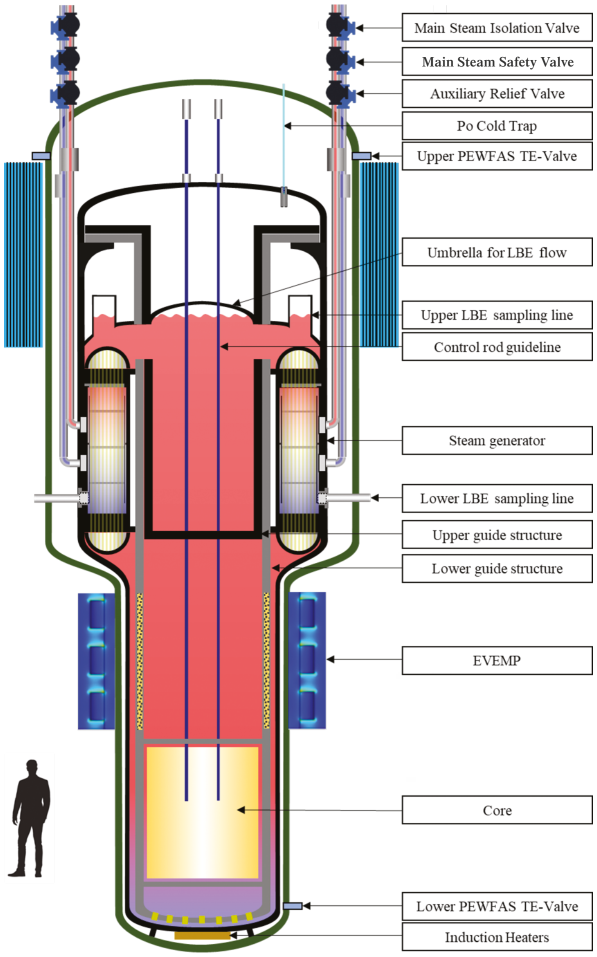
Figure 3.4. LFR facilities at the Westinghouse-Springfields site, United Kingdom

LEFREEZ (LEad FREEZing facility)

Source: Westinghouse (WEC).

LEWIN (LEad-to-Water INteraction facility)

Source: Westinghouse (WEC).



Source: Ulsan National Institute of Science and Technology.

Figure 3.5. Outline of MicroURANUS conceptual design

In the United States, work on LFR concepts and technology has been carried out since 1997. In addition to reactor design efforts, these activities have included work on lead corrosion/material compatibility and thermal-hydraulic testing at a number of organizations and laboratories, as well as the development and testing of advanced materials suitable for use in lead or LBE environments. University-related activities include several projects sponsored by the US Department of Energy (DOE) under Nuclear Energy University Project funding. Recent projects directly related to LFR technology include: an effort led by the Massachusetts Institute of Technology in the area of corrosion/irradiation testing in lead and lead-bismuth eutectic; a project at the University of Pittsburgh to develop a versatile liquid lead testing facility and test material corrosion behaviour and ultrasound imaging technology in liquid lead; and 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 and sublayers.

In the industrial sector, Westinghouse is currently completing conceptual design and plant layout activities. It is also engaged with several universities and national laboratories to pursue technology developments related to LFR, including the above-mentioned Nuclear Energy University Project and the Department of Energy's Technology Commercialization Fund, which is supporting the adaptation of the fast reactor analysis software developed at national laboratories for industry needs.

Columbia Basin Consulting Group, under a DOE-funded project, has completed a Design Report for a new conceptual design intended to take maximum advantage of "off-the-shelf" modularization in the balance of plant systems and an innovative reactor building construction approach. The Design Report reflects significant technical, affordability and cost of power to the grid improvements.

In China, the Chinese government has provided continuous national support to develop lead-based reactor technology since 1986, by the Ministry of Science and Technology, the National Science Foundation, China's 13^{th} and 14^{th} Five-Year Plan, etc. After more than 30 years of research on lead-based reactors, the China LEAd-based Reactor (CLEAR), proposed by the Institute of Nuclear Energy Safety Technology, was selected as the reference reactor for the ADS project, as well as for the technology development of the Gen IV LFR. A 10 MW-grade CLEAR-M10 project aiming at the construction of a small modular energy supply system has been launched. In August 2021, an electric heated pool type LBE-cooled integration facility CLEAR-M0 with more than 5 MWt power completed construction and started commissioning operation at the International Academy of Neutron Science.

For the ADS system, several concepts and related technologies are under assessment. For example, the detailed conceptual design of CLEAR-I for MA transmutation with dual operation modes and CLEAR-A travelling-wave reactor driven by the advanced external neutron source for energy production, led by the Institute of Nuclear Energy Safety Technology with the support of local government, has been carried out. The China initiative Accelerator Driven System project, led by the Chinese Academy of Science, in collaboration with the China Institute of Atomic Energy and other industrial organizations, aims to build a 10 MWth sub-critical experimental LBE-cooled reactor coupled with an accelerator. The Ministry of Ecology and Environment approved the environmental impact assessment report of the first phase of the project in 2022 and the civil construction of the first phase began on 1 October.

In recent years, other organizations have started paying more attention to LFR development. The State Power Investment Corporation focuses on BLESS-D (Breeding Lead-Based Economical Safe System) reactor design. China General Nuclear Power Group proposed a modular size China LFR concept which aims to demonstrate the technical feasibility of LFR. In addition, the China National Nuclear Corporation and several universities such as Xi'an Jiaotong University and the University of Sciences and Technology of China have been carrying out LFR-related research.

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

Molten salt reactor

The following GIF members participate in the memorandum of understanding for collaboration on molten salt reactor (MSR) R&D: Australia, Canada, Euratom, France, Russia, Switzerland and the United States. In addition, there are three observers: China, Japan and Korea. The mission of the MSR pSSC is to support international collaboration on the development of future 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.

In 2022, the pSSC meeting was held twice in person (the 32nd and 33rd meetings). Based on a decision from the 52nd Policy Group meeting, the GIF MSR group will continue as a pSSC and will evaluate its technical situation in two years. In 2022, members developed a plan to re-examine the GIF Proliferation Resistance and Physical Protection white paper on MSRs and to focus co-operation within the new system research plan for the MSR pSSC around three main axes: 1) salt behavior; 2) materials properties; and 3) system integration.

Main characteristics of the system

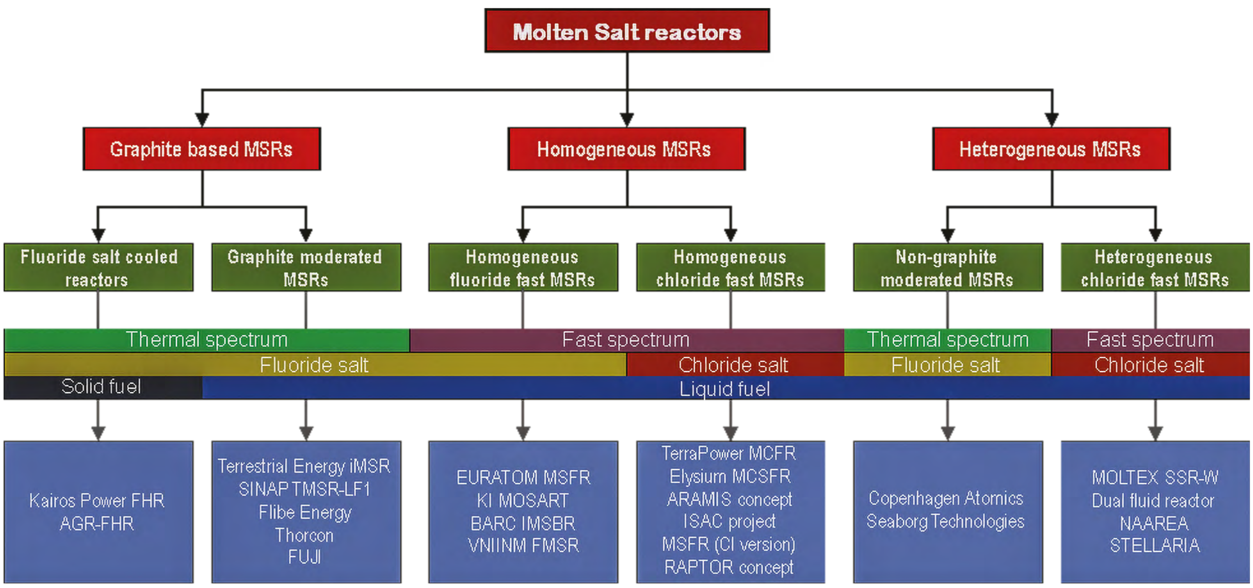
An MSR is any reactor where a molten salt has a prominent role in the reactor core (i.e. fuel, coolant and/or moderator). Liquid-fuel MSRs are a type of nuclear fission reactor in which a halide salt serves as the nuclear fuel and may also serve as the coolant. In solid-fuel MSRs, the halide salt serves as the coolant for solid phase nuclear fuel. MSRs were originally conceived in the 1940s.

Both solid- and liquid-fuelled MSRs have seen a resurgence in interest over the past two decades. Proposed designs, with molten salt fluorides and chlorides 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 in an MSR fuel cycle. MSRs can be grouped into three classes and six families according to their technical characteristics (IAEA, 2021), as shown in Figure 3.6, along with prominent MSR developers.

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 by liquid fuels can necessitate fully remote maintenance. Molten salt can also become highly corrosive if exposed to oxidative impurities. Overall, MSRs have substantial technology differences from both existing light water reactors (LWRs) and other advanced reactor concepts, necessitating different approaches to safety assessment, safeguards and operations.

Figure 3.6. The most studied molten salt reactor concepts, with some key developers



Note: MSR: molten salt reactor.

R&D objectives

The GIF's objective is to support collaboration on technology, data and analysis methods. While MSRs may be demonstrated in the near term (next few years), their performance could be improved through the development of improved technologies and techniques as well as increasing the amount of validated fuel salt thermo-physical and thermochemical data. Potentially useful collaborative projects include:

- salt fabrication and measurement of thermochemical and thermo-physical properties;
- performance of integral and separate effects tests to validate safety performance;
- development of improved neutronic and thermal -hydraulic models and tools;
- study of materials issues of 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;
- development of a safety and licensing approach dedicated to liquid-fuelled reactors.

National molten salt reactor demonstration and development highlights

The Australian Nuclear Science and Technology Organisation (ANSTO) is conducting research to evaluate the impact of thermal aging, high-temperature creep, radiation damage, and molten salt corrosion on the performance of candidate alloys for MSR systems. In partnership with the Idaho National Laboratory, ANSTO investigates the mechanical and fracture properties of nuclear graphite through split-disk testing and advanced numerical modelling. Moreover, ANSTO is investigating the potential application of its SYNROC waste encapsulation technology for the safe disposal of molten salt nuclear waste.

In Euratom, the project SAMOSAFER (severe accident modelling and safety assessment for fluid-fuel energy reactors) aims to develop simulation models and tools validated with experiments complemented with the design and demonstration of new safety barriers. The ultimate objective is to ensure that the MSR can comply with all expected nuclear regulations in 30 years' time.

Simulation tools that couple chemistry with multi-physics to simulate the behaviour of the fuel salt are being developed for system design and safety analysis. The chemical state of the fuel salt has been predicted using the thermodynamic Molten Salt Database. The modelling of the solid fission products was conducted to extend the tools to simulate the transport of noble metal particles. A reactivity insertion transient has been modelled with individual zones using a compressible fuel salt model integrated into a full model with incompressible salt. Validation needs are being defined and prioritized. Melting and solidification models are also being improved with testing under forced conditions in the ESPRESSO facility.

Models are being developed and validated for tracing the source term and its chemical form and mobility during nominal and accidental conditions. The source term distribution accounts for the neutron irradiation, radioactive decay, removal constants and residence times. Other models are being developed and validated for decay heat removal from the fuel circuit. Effects of the turbulence flow were studied, as was helium bubbling. The main purpose is to investigate phenomena like radiation heat transfer, salt flow with a free surface and salt solidification/melting. Two experiments have been developed: 1) forced convection inside a closed channel; and 2) flow along an open channel. In total, 13 experiments have been performed.

To improve safety margins and define monitoring needs, the effect of turbulence on the reactivity and the power has been evaluated. Normal plant operational states and a list of the main plant parameters have been identified. A methodology was developed to follow the salt composition in the core and define the best conditions. Experiments were done after dissolution of the pure fluoride salts in the molten salt FLiNaK at 600°C.

At JRC, R&D efforts in 2022 included work to find the optimal method to synthesize highly pure $PuCl_3$. The method is still being optimized, but it already provides a good source of initial material for studies on thermophysical properties of chloride-based molten salt reactor fuel. Using the thus synthesized $PuCl_3$, key fuel systems have been re-evaluated by calorimetric devices, looking into phase equilibria and/or enthalpy of fusion data.

A design capsule for corrosion studies was manufactured in 2022, respecting internationally respected standards for conducting such experiments. The method will first be tested on inactive FLiNaK salt with nickel ingot immerged into the molten salt and will later be extended to experiments using selected fuel compositions, including fluoride and chloride families.

With the increasing demand for reliable measurements of density for MSR fuels, significant effort has been expended to design and test the novel densitometer at JRC Karlsruhe. The selected method of measurement is based on the Archimedean buoyancy effect, and the entire set up was designed such that it fits the current glove boxes. In 2022, the spherical bob that is immersed into the molten salt during measurement was redesigned into a semi cylindrical shape. The bob is made of nickel to avoid corrosions at high temperatures, but the selection of the material can be adopted to the type of salt measured. After a successful test on FLiNaK, NaCl-KCl, LiCl-KCl eutectic salts, the work has been extended by a series of measurements on the NaCl-$CeCl_3$ samples which are direct surrogates to NaCl-$PuCl_3$. The results confirm the high reliability of the method. In the second half of 2022, the set up was installed in the hot glove box for actinide-containing sample measurements and successfully tested to high temperature. Moreover, the method was downscaled so that in the future there will not be a need to use major amounts of actinide materials. First measurements on plutonium-containing chloride and fluoride salts are planned for 2023.

Development of the JRC thermodynamic database on the key fuel and coolant MSR systems (known as JRCMSD) is one of the key ongoing activities. The database has been improved by novel experimental data of some of the PuF_{3}-, ThF_{4}- and $PuCl_{3}$- containing binary and ternary systems and extended by a series of systems containing corrosion and fission products.

In the newly launched European project MIMOSA, in which the use of fast spectrum chloride MSR is proposed for the incineration of actinides from MOX-type spent fuel, the Czech Research Centre Řež has started work on the preparation of a carrier melt of NaCl - $MgCl_2$ composition for the measurement of its neutron characteristics in the experimental LR-0 reactor.

In France, the R&D programmes around MSR initiated a few years ago by the French National Centre for Scientific Research (CNRS) and more recently by the CEA was continued and developed. Since 2020, the CEA has developed an MSR programme, supported by ORANO, around the ARAMIS concept (Advanced Reactor for Actinides Management In Salt). The goal of the programme was to study the opportunity of fast chloride MSRs with regards to plutonium management (Pascal et al., forthcoming). The CEA focused its programme on reactor design, neutronics studies, salt deletion evaluation, and first experiments on active salt synthesis and material corrosion studies induced by inactive chloride salt. During this period, the CNRS, also supported by ORANO, has also contributed to neutronic studies around actinide management solutions based on fast chloride MSR in the frame of Laura Mesthiviers' PhD thesis for burner MSRs (Mesthiviers et al., 2022; Tillard et al., 2022; Mesthiviers, 2022) and Hugo Pitois' PhD thesis funded by the SAMOSAFER project on a breeder chloride MSR (Pitois et al., 2022). Since 2022, the CEA and the CNRS, along with industrial partners (ORANO, FRAMATOME, EDF) have been contributing to the French common project ISAC (Innovative System for Actinide Conversion), supported by the "France 2030" investment plan. The main objective of this project is to assess the feasibility of a fast chloride MSR for americium and eventually curium transmutation.

Three options are being considered, all in a fast spectrum and in molten chloride: 1) iso-generator; 2) Pu burner; and 3) minor-actinide transmutor. These programmes are multidisciplinary and cover:

- the reactor system (i.e. neutronics, salt depletion, materials, components);
- the reactor operation and safety (i.e. normal transients, accidental transients, start-up, draining);
- the associated fuel cycle (i.e. salt behaviour, corrosion, fission product management, salt polishing, salt synthesis [$PuCl_3$], scenario studies) and the refuelling and polishing strategy;
- multi-physics and chemistry modelling and simulation (neutronic/thermal-hydraulic simulations, coupling between salt depletion and thermochemical calculations, etc.);
- code development for MSR multiphysics modelling: salt depletion code (REM/MOSARELA), neutronic/thermahydraulic code (TFM/OpenFoam, TrustNK, etc.).

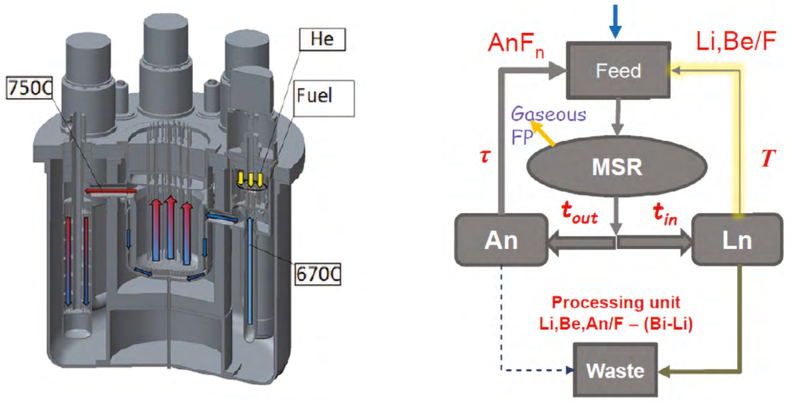
In Russia, in 2022, Rosatom continued to provide support to preliminary MSR design development for:

- small test reactor (10 MWt) and large power units (2 to 4 GWt) with a homogeneous core;
- fuel salt clean-up unit at the site of the Mining and Chemical Combine Zheleznogorsk to demonstrate the control of the reactor and fuel salt management with different long-lived actinide loadings, drain-out, shut down, etc.

The design is based on Li,Be,An/F MSR plant applied for the transmutation of long-lived actinides from used LWR fuel. For the removal rates, it is foreseen that $T >> \tau$, where T represents soluble fission product removal time of 300 days and τ actinides recycling time of 30 days. The MSR design uses high Ni alloy as the containment vessel and other metallic parts of the system. Fuel salt temperatures at the core outlet can be up to 750°C and energy will be transferred to an intermediate coolant salt which in turn is used to produce supercritical steam. The effort related to MSR R&D in 2022 is published in the following references (Rudenko et al., 2022; Redkin et al., 2022; Karfidov et al., 2022; Orlov et al., 2022; Izhutov et al., 2022; Kotov et al. 2022; Zakirov and Ignatiev, 2022; Kupriyanov et al., 2022; Shutova et al., 2022; Gatsa et al., forthcoming; Abalin et al., 2022; Abalin, 2023).

In the United States, a number of both salt-cooled and salt-fuelled MSR supportive activities were performed in 2022. The DOE's Office of Nuclear Energy-supported MSR demonstration projects continue to make progress. Southern Company Services and TerraPower have commissioned their chloride salt integrated effects test facility (Office of Nuclear Energy, 2022). Development of their molten chloride reactor experiment is also proceeding. The Office of Nuclear Energy is sponsoring fuel salt synthesis for the Molten Chloride Reactor Experiment using plates from the Zero Power Physics Reactor at Idaho National Laboratory. Kairos Power's construction permit application for its Hermes low-power demonstration reactor is proceeding through the Nuclear Regulatory Commission (NRC) review process. Both the safety evaluation and environmental review processes are scheduled to be completed in 2023 (US NRC, 2022a). Abilene Christian University submitted a construction permit application to the NRC for a molten salt research reactor (US NRC, 2022b) with operation intended for 2026.

Terrestrial Energy USA also continues to file multiple documents in pursuit of standard design approval for its IMSR400 (US NRC, 2019). Multiple private companies are proposing to create molten salt fuel from used water reactor fuel; for example, Exodys Energy proposes the UP-CYCLE for used light water reactor fuel to fast-spectrum chloride salt reactor fuel. Moltex continues to increase its interest in US waste conversion based on its WATSS (waste to stable salt) conversion technology. Alpha Tech Research focuses on the continuous electrochemical extraction process to convert used nuclear fuel into multiple useful products. TerraPower is investigating chloride volatility to separate materials from used fuel under APRA-E funding.



Source: Zakirov and Ignatiev (2022).

Note: LWR: light water reactor; MSR: molten salt reactor, LWR; An: actinides; Ln: lathanides.

Figure 3.7. Li,Be,An/F molten salt reactor plant (left) and the respective flowsheet (right) for the transmutation of long-lived actinides from used LWR fuel

The US NRC continues its activities to develop technology-inclusive, performance-based, risk-informed licensing practices for advanced reactors. The NRC is revising its radioactive waste, physical security and emergency preparedness rules. The commission has instructed NRC staff to develop radioactive waste disposal rules based on hazards as opposed to prescriptive quantities for both greater than Class C and transuranic wastes (Nuclear Newswire, 2022). The NRC has also recently approved a fuel salt qualification methodology (US NRC, 2022c).

Fuel salt is the defining element of liquid-fuelled MSRs, serving as both the nuclear fuel and coolant. Fuel salt properties derive from its composition and state (primarily temperature). Mapping fuel salt composition to thermophysical and thermochemical properties is consequently a major area of emphasis for the DOE Office of Nuclear Energy's RD&D. The *Molten Salt Thermal Properties Database – Thermochemical* (MSTDB-TC) and *Molten Salt Thermal Properties Database – Thermophysical* (MSTDB-TP) are now available for public use. The MSTDB-TC contains Gibbs energy models and values for molten salt components and related systems of interest with respect to MSR technology. The MSTDB-TP consists of tabulated thermophysical properties and relations for computing properties as a function of temperature or composition (Oak Ridge National Laboratory, 2022a). The Office of Nuclear Energy's activities also emphasize the quality assurance necessary to use the data for MSR safety evaluation (Rose and Creasman, 2022; Rose, 2022). It also continues to improve the tools used to evaluate salt (Karfidov et al., 2022) and cover gas properties (Gallagher et al., 2022; Felmy et al., 2021; Medina et al., 2022; Andrews, McFarlane and Myhre, 2022).

The DOE also continues to support multiple projects focused on developing the understanding of MSR accident progression sequences such as performing salt spill testing (Thomas and Jackson, 2022). The Oak Ridge National Laboratory's fluoride and chloride (Oak Ridge National Laboratory, 2022b) salt-pumped test loops also both operated during 2022.

The Office of Nuclear Energy also selected multiple, additional MSR supportive university projects, including "Bridging the Gap between Experiments and Modelling to Improve the Design of MSRs" at the University of California at Berkeley and "Integrated Effects of Irradiation and Flibe Salt on Fuel Pebble and Structural Graphite Materials for MSRs" at MIT. The DOE's Office of Science also initiated a project at Los Alamos National Laboratory to improve the fundamental understanding of the interaction of molten salts with reactor materials (Los Alamos National Laboratory, 2022).

In Canada, Canadian Nuclear Laboratories (CNL) continued to develop expertise and capabilities in support of molten salt small modular reactor (SMR) concepts. Funded through the CNL's Canadian Nuclear Research Initiative programme, agreements with three MSR developers include work on electro-chemical separation methods, tritium management, reactor physics, thermal-hydraulics and safeguards studies to advance MSR technology.

Under Atomic Energy of Canada Limited's Federal Nuclear Science & Technology Work Plan, the CNL continued to develop molten salt capabilities across a wide range of areas, including molten salt fuel behaviour in accident conditions, salt chemistry and thermodynamic properties, multi-physics core behaviour, thermal-hydraulics modelling and system behaviour, and decay heat removal. In 2022, the work focused on:

- molten salt thermo-physical property measurement and atomistic simulations to predict fuel salt transport and thermodynamic properties;
- design of fuel salt irradiation rig for low power irradiation in a research reactor;
- commissioning of a natural circulation loop for carrying out molten salt corrosion tests;
- commissioning of a molten salt natural circulation heat transfer loop to support fluid flow and heat transfer studies of chloride and fluoride salt mixtures.

In Switzerland, MSR research continued in 2022 at Paul Scherrer Institute (PSI), with the major aim of technology monitoring, education of new experts, and development of knowledge and simulation capabilities in fuel cycle, system behaviour and thermodynamic areas of MSR research.

In the area of fuel cycle assessment, another stage of the studies related to the breed-and-burn fuel cycle in the molten chloride fast reactor (MCFR) was finalized in 2022. The proper fuel concentration control implemented the EQL0D procedure for the multi-fluid MCFR. The salt 40% UCl_3 60% NaCl was applied initially as a feed composition to the reactor. Later using the new control system, the molar share of actinides was preserved in the core. A sensitivity study was repeated for single-fluid MCFR of a 150 m^3 core size, where the dependency on off-gas system intensity and on salt replacing, or actually tapping, velocity was analyzed. As a second step, the impact of a second fluid in the blanket was analyzed for a core volume of 100 m^3 and a blanket volume ranging from 10 m^3 to 100 m^3 (Figure 3.8). A potential for core size reduction or fuel burnup increase was identified in a form of excess reactivity (Santora, 2022).

In China, the Shanghai Institute of Applied Physics, the Chinese Academy of Sciences is steadily promoting the related work of thorium MSR. The 2 MWth molten salt test reactor was granted a construction license. The installation of the main equipment of the experimental reactor and the individual commissioning of each system and the overall pre-nuclear commissioning have been completed. Research work based on the follow-up development of the MSR is also being carried out, notably in relation to:

- Continue to promote the R&D and optimization of corrosion- and irradiation-resistant high-temperature alloy of MSRs and complete the corrosion performance evaluation and tellurium embrittlement resistance evaluation of the alloy.
- The methodology for evaluating the irradiation life of nuclear graphite based on ion beam has been developed. The analysis methodology of reactor material mechanics, irradiation and other properties has been developed. Several material standards and specifications have been established.
- The thermodynamic and kinetic characteristics of the Eu(III)/Eu(II) couple in molten FLiBe were investigated by electrochemical methods. Metallic uranium and molybdenum can be obtained via elelctrolytic reduction of mixed uranium and molybdenum oxides in molten chlorides.

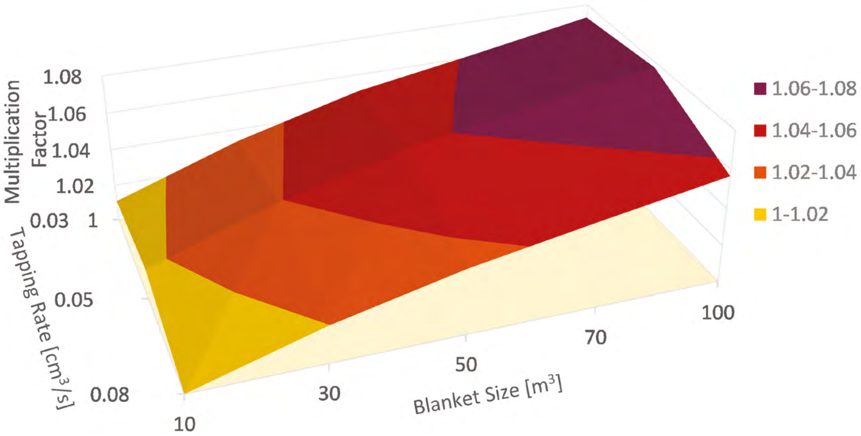
In Japan, the International Thorium Molten-Salt Forum (ITMSF) has been participating as an observer of the GIF-MSR-pSCC since 2005. The ITMSF proposed a revised white paper on *Proliferation Resistance and Physical Protection* at the 2022 GIF-MSR-pSSC meetings. The paper was a revised version of the one from 2020 discussed in previous meetings. The improvements are as follows:

- As for proliferation resistance, one important assumption is the “actor is the state”. So, it must be assumed that the IAEA safeguard is not working. Also, a proliferation resistance white paper should cover possible design options for MSRs not just one specific design. Eleven possible design variants are: 1) fuel type; 2) coolant type; 3) neutron spectrum; 4) fuel salt type; 5) fuel feeding after startup; 6) reprocessing type; 7) fissile and fertile material (combination of 4 different fissile materials and 3 different fertile materials); 8) blanket loop; 9) addition of minor actnides; 10) the purpose of MSRs; and 11) core structure.
- As for physical protection, a study must start from design basis threat and beyond-design basis threat should be studied. These are included in the revised 2022 version.

The Japanese government started to support the development of MSR technology in 2019. Two start-up companies (TTS and MOSTECH) are promoting an MSR with fluoride salt that is moderated by graphite, and one (BERD) is promoting a fast-spectrum MSR with chloride salt. Some of their studies were presented at conferences of the Atomic Energy Society in Japan in 2022.

Jiri Krepel
Chair of the MSR SSC,
with contributions from MSR members

Figure 3.8. Dependency of keff for multi-fluid molten chloride fast reactor with 100 m^3 core volume operated in breed-and-burn cycle mode and with blanket volume ranging from 10 m^3 to 100 m^3



Source: Santora (2022).

References

Abalin, A. (2023), "Container materials for molten salt reactors: Problems and research", *Physics of Atomic Nuclei*, Vol. 1, forthcoming.

Abalin, S. et al. (2022), "Facility for testing MSR container materials at the NRC 'Kurchatov Institute'", *Physics of Atomic Nuclei*, Vol. 5, pp. 12-22.

Andrews, H.B., J. McFarlane and K.G. Myhre (2022), "Monitoring noble gases (Xe and Kr) and aerosols (Cs and Rb) in a molten salt reactor surrogate off-gas stream using laser-induced breakdown spectroscopy (LIBS)", *Applied Spectroscopy*, Vol. 76/8, pp. 988-997, http://dx.doi.org/10.1177/00037028221088625.

Felmy, H.M. et al. (2021), "On-line monitoring of gas-phase molecular iodine using Raman and fluorescence spectroscopy paired with chemometric analysis", *Environmental Science & Technology*, Vol. 55/6, pp. 3 898-3 908, http://dx.doi.org/10.1021/acs.est.0c06137.

Gallagher, R.C., A. Birri, N. Russell and N.D.B. Ezell (2022), "Design and performance of a variable gap system for thermal conductivity measurements of high temperature, corrosive, and reactive fluids", *International Journal of Heat and Mass Transfer*, Vol. 192, pp. 122763, https://doi.org/10.1016/j.ijheatmasstransfer.2022.122763.

Gatsa, P. et al. (forthcoming), "Thermal-hydraulic analysis of the MSR fueled with transuranic elements", *Physics of Atomic Nuclei*, Vol. 3.

IAEA (2021), "Status of molten salt reactor technology", *Technical Report Series*, No. 489, International Atomic Energy Agency, Vienna.

Izhutov, A. et al. (2022), "Corrosion resistance of high nickel alloys under irradiation in the SM-3M test reactor", *Proceedings of the X Russian Conference "Radiochemistry 2022*, 26-30 September, Saint Petersburg.

Karfidov, E. et al. (2022) "Corrosion behavior of candidate functional materials for molten salts reactors in LiF-NaF-KF containing actinide fluoride imitators", *Materials*, Vol. 15/3, pp. 761, https://doi.org/10.3390/ma15030761.

Karfidov, E., Y.P. Zaikov, E.V. Nikitina, K.E. Seliverstov and A.V. Dub (2022), "High-temperature passivation of the surface of candidate materials for MSR by adding oxygen ions to FLiNaK salt", *Materials*, Vol. 15/15, p. 5 174, http://dx.doi.org/10.3390/ma15155174.

Kotov, Ya. et al. (2022), "Study on minor actinide incineration in molten salt reactors with uranium or plutonium loadings", *Problems of Nuclear Science and Engineering Series: Physics of Nuclear Reactors*, Vol. 2, pp. 30-37.

Kupriyanov, K. et al. (2022), "Effective fraction of delayed neutrons in molten salt reactor with circulating fuel", *Physics of Atomic Nuclei*, Vol. 2, pp. 97-106.

Los Alamos National Laboratory (2022), " Los Alamos National Laboratory to lead study of molten-salt nuclear reactor materials", https://discover.lanl.gov/news/0816-molten-salt.

Medina, A.S. et al. (2022), "Iodine and carbonate species monitoring in molten NaOH–KOH eutectic scrubber via dual-phase in situ Raman spectroscopy", *ACS Omega*, Vol. 7/44, pp. 40 456-40 465, http://dx.doi.org/10.1021/acsomega.2c05522.

Mesthiviers, L. (2022), "Capacité de conversion des transuraniens en réacteurs à sels fondus", PhD thesis, Grenoble Alpes University.

Mesthiviers, L., D. Heuer, E. Merle, G. Grassi, B. Morel and G. Senentz (2022), "Study on actinide conversion capabilities of molten salt reactors (MSR)", *Proceedings of the International Conference on Fast Reactors and Related Fuel Cycles FR22: Sustainable Clean Energy for the Future*, IAEA Headquarters, Vienna.

Nuclear NewsWire (2022), "NRC to consolidate LLW rulemaking", www.ans.org/news/article-3838/nrc-to-consolidate-llw-rulemaking.

Oak Ridge National Laboratory (2022a), Molten Salt Thermal Properties Databases, https://sc.edu/study/colleges_schools/engineering_and_computing/docs/research/mstdb_access.pdf.

Oak Ridge National Laboratory (2022b), "Facility to alleviate salt technology risks (FASTR): Design report", www.ornl.gov/publication/facility-alleviate-salt-technology-risks-fastr-design-report.

Office of Nuclear Energy (2022), "Southern Company Services and TerraPower build world's largest chloride salt system", www.energy.gov/ne/articles/southern-company-services-and-terrapower-build-worlds-largest-chloride-salt-system.

Orlov, A. et al. (2022), "Production of plutonium and neptunium fluorides – Components of the fuel composition for MSR", X Russian Conference "Radiochemistry-2022", 26-30 September, Saint Petersburg.

Pascal, V., P. Allegre, C. Chabert, P. Gauthe, E. Martin-Lopez, J. Martinet, L. Matteo, M. Mascaron, L. Tardieu and C. Venard (forthcoming), *Molten Salt Reactor Technology: Opportunities of Molten Salt Fuel for Actnides Management*, Technical Meeting on Back End of the Fuel Cycle Considerations for Small Modular Reactor, IAEA Headquarters, Vienna, 20-23 September 2022, forthcoming.

Pitois, H., D. Heuer, A. Laureau, E. Merle, M. Allibert and S. Delpech (2022), "A closed fuel cycle option using the MSFR concept with chloride salts and the U/Pu cycle", *Proceedings of the Global 2022 International Conference*, Reims, France.

Redkin, A., E. Il'ina, S. Pershina, P. Mushnikov, S.Stankus, A. Agazhanov, Y. Zaikov, A. Kholkina and A. Artamonov (2022), "Thermal properties of Li2BeF4 near melting point", *Thermo*, Vol. 2/3, pp. 107-115, https://doi.org/10.3390/thermo2030010.

Rose, M.A. (2022), "Approach to precision and bias analysis of interlaboratory salt study results", ANL/CFCT-22/25, http://dx.doi.org/10.2172/1884558.

Rose, M.A. and S. Creasman (2022), "Report on data quality required for modeling an MSR", ANL/CFCT-22/17, http://dx.doi.org/10.2172/1880780.

Rudenko, A., A. Redkin, E. Il'ina, S. Pershina, P. Mushnikov, Y. Zaikov, S. Kumkov, Y. Liu and W. Shi (2022), "Thermal conductivity of FLiNaK in a molten state", *Materials*, Vol. 15/16, p. 5603, https://doi.org/10.3390/ma15165603.

Santora, J. (2022), "Assessment of core minimization options for breed-and-burn molten chloride fast reactors", MSc thesis, Swiss nuclear master course, http://dx.doi.org/10.13140/RG.2.2.25119.84646.

Shutova, K. et al. (2022), "Modeling of tritium behavior in molten salt reactor", *Physics of Atomic Nuclei*, Vol. 2, pp. 118-130.

Thomas, S. and J. Jackson (2022), "MSR salt spill accident testing using eutectic NaCl-UCl3", ANL/CFCT-22/32, http://dx.doi.org/10.2172/1889274.

Tillard, L., L. Mesthiviers, E. Capelli, D. Heuer, R. Maréchaux, Y. Meridiano, E. Merle, B. Morel, I. Morlaes and G. Senentz (2022), "Estimation of the vitrified canister production for PWR fleets integrating MSRs with different fuel cycle strategies", *Proceedings of the Global 2022 International Conference*, Reims, France.

US NRC (2019), "Integral molten salt reactor (IMSR)", web page, www.nrc.gov/reactors/new-reactors/advanced/licensing-activities/pre-application-activities/imsr.html.

US NRC (2022a), "Hermes – Karios Project status dashboard", web page, www.nrc.gov/reactors/non-power/new-facility-licensing/hermes-kairos/dashboard.html.

US NRC (2022b), "MSRR – Abilene Christian University application", www.nrc.gov/reactors/non-power/new-facility-licensing/msrr-acu.html.

US NRC (2022c), "Fuel qualification for molten salt reactors (NUREG/CR-7299, ORNL/TM-2022/2754), www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7299/index.html.

Zakirov, R. and V. Ignatiev (2022), "Fuel cycle of the LiF-BeF2 molten salt actinide burning reactor", *Russian journal Physics of Nuclear Reactors*, Vol. 2, pp. 38-47.

Super-critical water reactor

The following GIF members are signatories of the System Arrangement for collaboration on super-critical-water-cooled reactor (SCWR) research and development: 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.

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 megapascals [MPa]) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: 1) pressure-vessel concepts first proposed by Japan and more recently by a Euratom partnership and China; and 2) 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 due to 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.

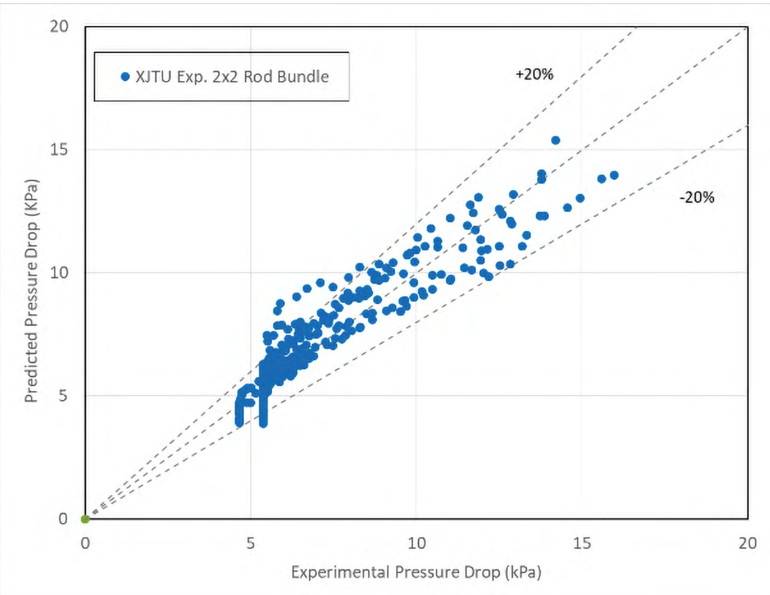
Technical highlights – Thermal Hydraulics and Safety Project

In Canada, the CNL conducted validation exercises to quantify the accuracy of the subchannel analysis code, ASSERT-PV 3.2-SC (Nava-Dominguez, Rao and Waddington, 2014) and system code CATHENA (Beuthe et al., 2020). For ASSERT-PV 3.2, the system code validation exercises were performed on pressure drop and heat transfer, both under supercritical conditions. The pressure drop validation exercise used the experimental data from three supercritical experiments:

- The National Technical University of Ukraine's (Razumovskiy et al., 2016) heat transfer and hydraulic resistance experiments with the tube, 1-rod channel and 3-rod channel test sections;
- the Xi'an Jiao-Tong University's (XJTU) (Wu, Wang and Bi, 2022) pressure drop experiments with a 4-rod channel test section;
- the Atomic Energy of Canada Limited's (Pioro, Duffey and Dumouchel, 2004) supercritical CO_2 pressure drop experiments with a tube.

A total of 777 supercritical pressure drop tests were simulated. The code accuracy of ASSERT-PV 3.2-SC in predicting total pressure drops over the various test sections was evaluated from the simulation results. The validation exercise also includes sensitivity studies conducted to investigate the effects of nodalization, boundary conditions and model options on the simulation results. Figure 3.9 depicts a comparison between the ASSERT-PV 3.2-SC predictions and the XJTU experimental data. In a

Figure 3.9. Comparison of ASSERT-PV predictions with measurements for XJTU experiments



Source: CNL.

similar manner, 70 supercritical heat transfer tests were simulated. The code accuracy of ASSERT-PV 3.2-SC in predicting maximum wall temperatures in various test sections was evaluated.

Likewise, a validation exercise was performed to quantify the accuracy of system thermal-hydraulics code, CATHENA in predictions of fuel channel pressure drops under supercritical conditions. The CATHENA models for various test sections were developed. A total of 647 supercritical pressure drop tests were simulated. CATHENA's accuracy in predicting total pressure drops over the various test sections was evaluated from the simulation results. The validation exercise also included sensitivity studies conducted to investigate the effects of nodalization, boundary conditions on the simulation results.

China organized an international benchmark on SCWR thermal-hydraulics characteristics (IBSCTH), which is part of an international collaboration (ECC-SMART) to establish the design requirements for a supercritical water-cooled SMR. The IBSCTH includes heat transfer simulation of supercritical water flow in 2 × 2 rod fuel assemblies and a flow instability simulation of two parallel channels operating at supercritical water conditions. Regarding the 2 × 2 rod bundles, there are three geometries to analyze, including bare rods, partially wire-wrapped rods and a full-length wire-wrapped rod. In 2022, two technical meetings were held to address comments and questions related to the blind test. Results were collected from participants and the simulation models and general results have been compared. Computational fluid dynamics (CFD), subchannel and system analysis codes were part of the tools used to carry out the benchmarking tasks. The experimental data have been open to participants, which would help to modify and improve simulation models established in the blind test stage.

The Nuclear Power Institute of China, alongside the Xi'an Jiaotong University, the China Institute of Atomic Energy and other Chinese institutes, are leading an activity to compile a comprehensive experimental database for supercritical water and surrogates such as supercritical CO_2 and freons.

In Europe, the Centre for Energy Research and the Budapest University of Technology and Economics (BME) continued experimental activities in collaboration with the Department of Energy Engineering, the Department of Chemical and Environmental Process Engineering, and the Institute of Nuclear Techniques of the Budapest University of Technology and Economics. The Budapest University of Technology and Economics' Department of Energy Engineering has been working on theoretical research of water chemistry and thermal-hydraulic issues of SCW. Its Institute of Nuclear Techniques has focused on the investigation of the European small modular SCWR.

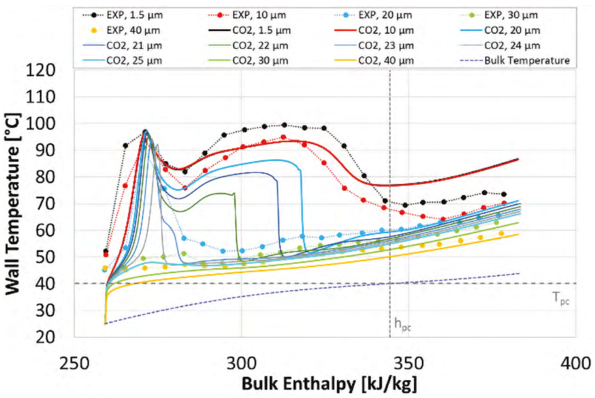
In the Czech Republic, the Super Critical Water Loop, which is hosted in the Centrum výzkumu Řež (CVR), remains the cardinal point on the material testing and safety assessment done in CVR. In a new IAEA coordinated reserach project launched in 2022, CVR SCW and the Czech National Radiation Protection Institute presented a proposal for a new benchmark for system codes, in which this complex facility will be simulated from the active channel up to the primary cooler in the secondary circuit. The benchmark aims to identify the capability of the system codes in order to simulate and follow complex transient scenarios from subcritical to supercritical regimes.

The KTH Royal Institute of Technology is working on the SWANS (Supercritical Water Applications in Nuclear Systems) project using the HWAT loop (High-pressure WAter Test loop). Detailed measurements are performed, including both the heat transfer at the wall and the internal flow structure, using a new experimental approach. The experiments aim to elucidate the mechanisms behind heat transfer deterioration and capture the influence of flow and heat flux conditions on the onset of deterioration.

Direct numerical simulation studies

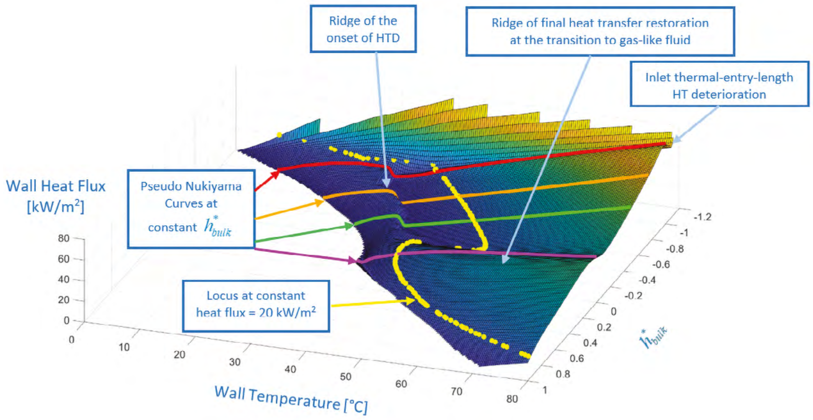
In Canada, the University of Carleton is completing a direct numerical simulation (DNS) analysis to study the effect of roughness in a heated channel flow. The test case is a heat channel flow of supercritical water at 24 MPa (for which the pseudocritical temperature of 654.38 K), with a heat flux value at the top and bottom walls of 25 kW/m^2. The present study uses a notably high spatial grid resolution along the lower wall, based on the dimensions of the surface roughness chosen for the lower wall of the channel. The simulation was run with this refined grid along the lower wall for both smooth- and rough-surface conditions. The smooth-surface results were also compared to prior results to confirm that the increased spatial resolution did not alter the turbulence. The validity of this roughness modelling approach was established through separate simulations where the momentum and thermal fields in close vicinity of the wall were compared for modelled surface roughness and three-dimensional represented surface roughness. This favourable comparison provides the foundation for future DNS studies of rough-surface effects on boundary-layer and channel flows based on modelled surface roughness, improving computational efficiency.

In Europe, the University of Nottingham is leading the analysis of the Reynolds-averaged Navier-Stokes (RANS) modelling of turbulent heat and mass transfer along corroded surfaces. This analysis aims to assess and further develop turbulent heat and mass transfer models for corroded surfaces. Different models investigate the influences of surface roughness induced by corrosion on heat transfer deterioration and pressure drop (Pioro, and Duffey, 2007). The eddy effect of corrosion roughness is assessed in the turbulence flow model. Two models were mainly used in 2022: the k-ε standard and k-ω SST models. The benchmark data provided by other partners are adopted in the CFD simulation programme. The temperature and pressure variations are similar to other research carried out with smooth pipe (Pioro, and Duffey, 2007; Kiss, Vágó and Aszódi, 2015; Brogna, 2017), which validates the developed simulation models.



Source: University of Pisa.

Figure 3.10. Wall temperature profiles with different roughness for CO_2 at 9 MPa, inlet temperature 25°C, heat flux of 50 kW/m² and mass fluxes of 350 kg/m²s



Source: University of Pisa.

Note: HT: heat transfer; HTD: heat transfer deterioration.

Figure 3.11. Representation of supercritical pressure heat transfer deterioration phenomena obtained processing results for Kline's experimental apparatus with CO_2 at 8.35 MPa for G = 300 kg/(m²s) and Tin = 0°C: continuous surface form

At the University of Sheffield, research on the SCWR primarily focuses on using DNS to produce detailed data and provide insights into the fundamental physics. An immersed boundary method is currently being implemented in a recently released CHAPSim 2.0. This method has now been tested for a simple flow over a sphere. Simulations of flows over pyramid rough surfaces are being developed and tested. The method will also be extended to the energy, thus allowing the consideration of the solid temperature distribution in complex geometries. On a separate but related research stream, the University of Sheffield has been working on extending the novel sub-channel CFD, a coarse mesh method that combines CFD and sub-channels approaches to SCWR applications. Sub-channel CFD and convectional CFD, both based on the CFD solver Code_Saturne, are used in the benchmarking exercise organized by the Nuclear Power Institute of China.

Research carried out at the Karlsruhe Institute of Technology in SCWR focuses on experimental heat transfer studies, system design and safety, and numerical simulations using large eddy simulations and DNS to produce reliable data to complement physical experiments and support the development of practical engineering models. At the KIT Model Fluid Facility, experimental investigation of the influence of corrosion on the heat transfer to supercritical fluid are under investigation. This task is performed in co-operation with the CVR in the Czech Republic.

At the University of Pisa, CFD RANS modelling employs an in-house methodology that takes into account the impact of wall roughness effects. In particular, the CFD results were compared with a recent publication reporting experimental results for CO_2 flowing upwards in circular pipes with different roughness heights. The CFD model was able to qualitatively reproduce the experimental phenomena, with heat transfer improving with the increase of the wall roughness. Taking into account both the experimental uncertainties and the sensitivity of the adopted turbulence models, the obtained results look promising and further applications are expected. Figure 3.10 presents simulation results against experimental data.

Safety analysis

In Europe, this work is led by the University of Pisa and consists of using both system thermal-hydraulics codes and CFD. This task aims to perform thermal-hydraulics analyses of normal and accidental scenarios for a supercritical water-cooled small modular reactor (SCW-SMR) proposed concept. Both normal operating conditions postulated accidents were considered in the analyses. Currently, a single transient investigating the behaviour of the plant in case of a reactor scram and its capabilities during a long-term station blackout has been considered. The addressed transient considers that, after the scram, the system pressure is gradually reduced. The calculation was successful; nevertheless, problems have been observed when the fluid works in the vicinity of the critical point, possibly due to numerical inconsistencies of the predicted fields owing to the sharp changes in fluid properties.

Another research path consisted in using the RANS model developed in recent years as a valuable tool to predict heat transfer capabilities in case of imposed temperature conditions. In this frame, the CFD code revealed that maps similar to the ones proposed by Nukiyama for the two-phase flow conditions may be obtained for supercritical fluids as well. Owing to the observed similarity, they were termed pseudo-Nukiyama curves. 3D heat flux maps (see Figure 3.11) were created as a function of the wall temperature and the dimensionless bulk enthalpy. The map clearly reports a depression of the local heat flux, which was connected to the occurrence of deteriorated heat transfer for cases addressing imposed heat flux conditions. As an additional result, by selecting a precise heat flux value on the map, it is possible to recognize the wall temperature trends as a function of the bulk enthalpy obtained for the corresponding imposed heat flux (yellow line) experimental case which inspired this study.

Thermal-hydraulics modelling for fuel clad corrosion

At the CNL, the proposed thermal-hydraulics model for fuel cladding focuses on the SCWR fuel cladding structural material candidates Alloy 800H and SS 310S. The proposed model is preliminary and requires verification. Depending on the exposure time, the model prediction is divided into two phases: the short-time exposure (initial phase) and the steady-state long-time exposure.

Short-time exposure (Initial phase)

This oxide development phase is expected to extend to hundreds of operation hours depending on water chemistry and fuel clad surface finish/condition. Based on observations from Guzonas et al. (2018), heat transfer enhancement is expected during the initial phase of oxide development. The fuel-clad surface roughness seems to increase, which will increase the flow turbulence and forced convective heat transfer to the coolant. The initial oxide layer is thin during this phase.

For uncoated fuel cladding, heat transfer to the coolant is expected to improve (see Chen et al., 2022). For Cr-coated fuel cladding, experiments on LWR fuel cladding showed a slight improvement in heat transfer coefficients (see Lee et al., 2021), and this is expected to be valid at SCWR conditions. Chen et al.'s (2022) heat transfer correlation for rough tubes can be used for the initial oxide development phase.

Steady-state long-time exposure

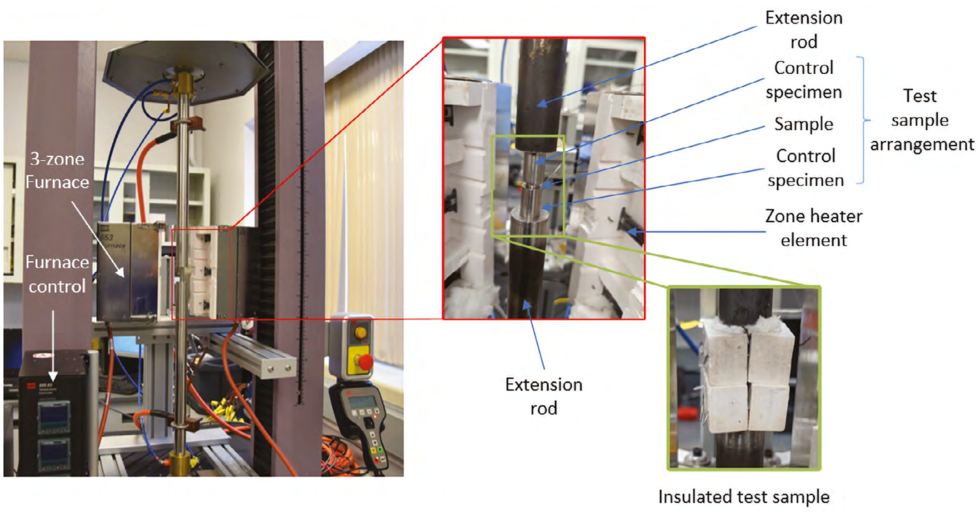
Oxide thickness is estimated using Guzonas et al.'s (2018) correlations for Alloy 800H and SS 310s.

Heat transfer will be calculated using the heat transfer model in the Idaho National Laboratory: the Cladding Corrosion Model "Bison". Bison's Theory Manual (Hales et al., 2014) presents a heat transfer model for zirconium cladding oxidation under normal LWR conditions and high temperature. The present study considers only the heat transfer part of the Bison model. The corrosion correlations do not apply to the SCWR. The Bison's fuel cladding corrosion model approach was adapted to the SCWR fuel-clad material candidates Alloy 800 and SS 310S to estimate the oxide-clad interface and waterside oxide temperatures.

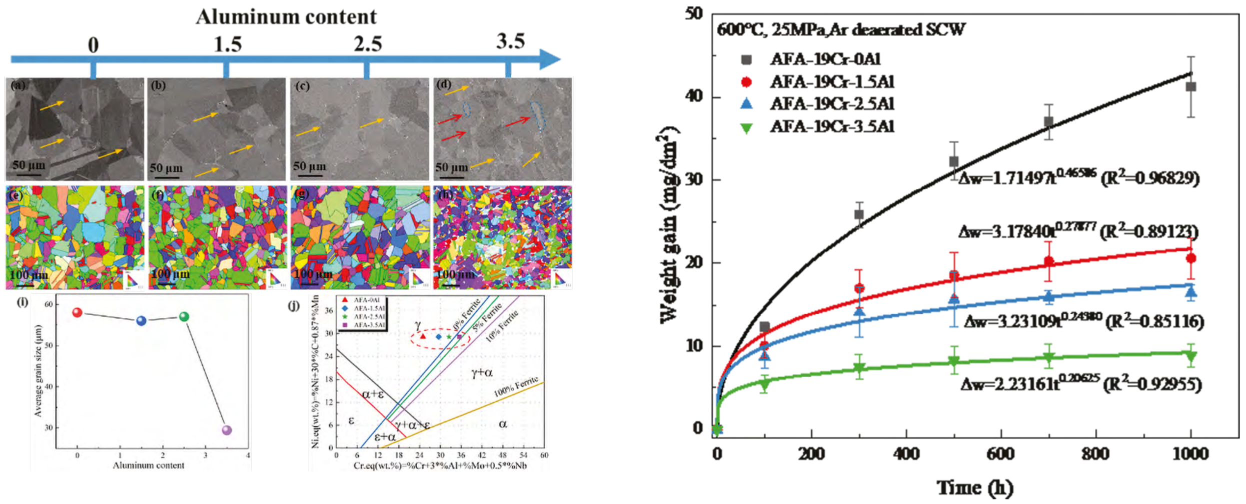
To complete this analysis, the thermal conductivity of the oxide layer needs to be measured experimentally. An apparatus was developed to measure the thermal conductivity of coated samples, where the surface coating may be intentionally deposited on the sample to achieve a desired outcome or may be due to uncontrolled processes such as oxidation. The thermal conductivity apparatus used for the present investigations adopts the Divided Bar methodology, with several modifications implemented to accommodate the nature of the specimen in the present investigation. The apparatus is shown in Figure 3.12. It consists of a 5 kN load bar connected to a furnace that has three independently heated zones capable of 800°C in each zone. Using the three-zone furnace, a temperature gradient is imposed on the test sample arrangement. Each control specimen is instrumented with four thermocouples in series to measure the heat flow and capture the rate of radial heat loss from the insulated test sample arrangement.

To quantify the thermal conductivity of a coated sample, a benchmark test is performed with a polished sample (benchmark sample) that is not coated. The test is then repeated with the coated specimen (test sample). The thermal conductivity is obtained from steady-state data using the Fourier

Figure 3.12. Test apparatus for thermal conductivity measurements of coated samples



Source: Canadian Nuclear Laboratories (CNL).



Source: University of Science and Technology Beijing (USTB).

Figure 3.13. Alumina forming austenitic model alloys with different concentrations of Al obtained by BSE and EBSD scanning microscopy Phase micrographs (left) and corrosion weight gain (right)

heat conduction equations across the test sample arrangement. A significant portion of measurement uncertainty is incurred when estimating the interface conductance between the control specimen and the sample based on the measurements.

Technical highlights – Materials and Chemistry Project

Novel alumina forming austenitic materials for fuel cladding

China performed studies on the short-listed SCWR candidate materials for fuel cladding, austenitic alloys 310S, 800H and other similar materials, showing that the major problem is still their corrosion resistance under normal operating and offset conditions. The largest material loss on the cladding tube must be controlled below 20 µm after about 5 years exposure to supercritical water at a minimum temperature of 600°C, which is predicted to be the hot spot temperature during normal operation conditions in a conceptual design with outlet temperature of 500°C. Alloys 310S and 800H fall in the margin in corrosion performances, and their creep and environmental cracking performances are also major concerns when considering the 0.5 mm thin cladding tube wall.

Based on previous work, alumina forming austenitic (AFA) material has been recognized as a potential candidate for the future applications of an SCWR. Alloying steel with aluminum (Al) enhances its corrosion resistance in high temperature environments with steam, and the higher concentration of Al quickly forms a protective alumina oxide film which reduces the oxygen in the environment to diffuse into the material matrix while resisting Fe atoms in the matrix to diffuse out to the surface, and hence significantly reduces the corrosion rate. However, increasing the Al concentration up to 3.5 wt% causes the materials to be very brittle and lowering its concentration below 2 wt% provides less corrosion resistance of the material. Finding an optimized Al concentration to balance the corrosion resistance and embrittlement were our focus in 2022.

The University of Science and Technology Beijing and Shanghai Jiao Tong University did parallel design optimization and preparation of novel AFA materials for SCWR fuel cladding. AFA steels with different Al concentrations up to 3.5 wt% were prepared based on the balanced Cr and Ni concentration of alloys 310S and 800H.

Figure 3.13 provides the phase micrographs of the alloys. The AFA alloy with an Al concentration less than 3.5 wt% shows a single austenitic phase at room temperature. Increasing Al to 3.5 wt% and above, the δ-ferrite appeared. The δ-ferrite is haphazardly distributed in the 3.5Al alloy matrix and can be easily distinguished from the face centered cubic phase. Generally, the σ-FeCr and δ-ferrite should be avoided when designing AFA alloys due to the harmful effect on high-temperature mechanical performances of brittle precipitates.

These AFA alloys were exposed in SCW at a temperature of 600°C and pressure of 25 MPa to evaluate their corrosion performances and determine the optimized Al concentration to meet the requirement of fuel cladding of an SCWR. Figure 3.13 also shows the corrosion weight gain of the coupons after exposure for 1 000 hours. The result is in very good accordance with the Al concentration, as the increasing Al content is beneficial in reducing the corrosion rate of the alloys. It has been suggested that a minimum of 2.4 wt.% Al is required to maintain the intact alumina scale to protect the matrix from being oxidized by high-temperature steam. The weight gain of specimens follows parabolic law and can be fitted with the empirical equation of $\Delta w=k\_p\ t\^n$, which indicates that the corrosion process of the alloy in supercritical water is a diffusion-controlled step, similar to other austenitic stainless steels such as 310S.

Based on tests done in the previous two years, silicon (Si) is effective in reducing the corrosion rate of AFA steels in SCW. For AFA alloys with a basic nominal chemical composition of 2.5Al-26Ni-19Cr, different concentrations of Si were evaluated in a similar manner. Exposure tests show that Si is more effective in reducing the corrosion rate of alloys in a deaerated SCW environment, showing that dissolved oxygen oxidizes Si more quickly.

The main activity in Europe in the field of SCWR R&D is the European-funded project ECC-SMART (Joint European Canadian Chinese development of small modular reactor technology). ECC-SMART Work Package No. 2, the "Materials Testing", focuses on the study of candidate cladding materials such as alloy 800H, stainless steel 310S and an AFA alloy based on the 310S. This material was manufactured by the University of Science and Technology of Beijing, which has extensive experience studying this type of material and is also involved in ECC-SMART. The test conditions chosen for this project are: 380°C and 25 MPa (pseudocritical point of water) and 500°C and 25 MPa, or average operating temperature of the reactor. Both are of great interest for the pre-design studies of the reactor.

In terms of the thermal efficiency of the reactor and thermohydraulics, it is planned to carry out studies of thermal conductivity of the oxides formed in the selected test conditions that will complete the tests carried out by Canada. Throughout this year, the European laboratories participating in ECC-SMART (VTT [Finland], RATEN [Romania], UCT and CVR [the Czech Republic], CIEMAT [Spain], STUBA [the Slovak Republic], and JRC [the Netherlands]) have focused their efforts on carrying out corrosion tests under the above defined test conditions, as well as under simulated accident conditions (1 200°C).

Regarding the crack initiation tests (SSRT), VTT and JRC have performed the first SSRT tests. Both laboratories modified their SSRT test rig or manufactured a new one to enable testing of half-tube and/or full-tube specimens of selected alloys. JRC performed its first test at 380°C in air, where a half-tube 310S specimen was loaded with a strain rate 1x10-6 s-1. This test was completed in December 2022. The first test at 380°C SCW for 310S half-tube specimen loaded with 1x10-7s-1 started at the end of 2022. Moreover, JRC designed and built a new SSRT test device based on pneumatic bellows. The device will provide additional capacity for SSRT testing within the ECC-SMART project and GIF activities.

One of the most important advances of the project in terms of reactor safety is the study of the corrosion behaviour of cladding materials pre-irradiated with neutrons. Samples of the three candidate alloys were irradiated in the Czech Republic (CVR) to study the effect of this variable on oxidation processes in supercritical water and on the mechanical properties of the materials. The irradiation process of the materials ended in June 2022 and they are now stored, waiting for the dose to drop to safe levels. It is planned to start the oxidation tests in supercritical water with irradiated material in 2023.

The activities are progressing well. For example, around 60% of the oxidation tests are finished and the first SSRT tests have begun. According to the preliminary results obtained from these oxidation tests, the 800H and 310S alloys (the tests with AFA will begin shortly) have shown weight gains in supercritical water with a higher corrosion resistance for A 800H. On the other hand, tests carried out under simulated accident conditions (1 200°C) have shown more stable oxide layers in samples of A 800H than in 310S steel. Furthermore, electrochemical studies have shown greater dissolution of materials around the pseudocritical point, decreasing as the system transitions to supercritical conditions.

In Canada, validation of a candidate fuel cladding material for a small modular SCWR requires data on the effects of irradiation on corrosion in a supercritical water environment. These conditions are, however, difficult to produce. Moreover, for preliminary experiments in validating candidate materials for SCW-SMR applications, neutron irradiations are not economical, with high associated irradiation and administration costs related to the shipping of radioactive materials. One of the CNL's selected candidate materials for fuel cladding applications is chromium-coated Zr-2.5%Nb pressure tube (PT) material, which was selected for its high neutron transparency compared to austenitic stainless steels and nickel-based alloys and better corrosion resistance compared to uncoated PT material. In this study, we investigated the irradiation effects on corrosion behaviour of the Cr-coated Zr-2.5%Nb PT material by performing dual-beam irradiations with protons and heavy ions.

The nominal Cr coating thickness applied on Zr-2.5%Nb ranged between 6 µm to 10 µm. The specimens in the present study were irradiated by heavy ions and protons to emulate neutron irradiation effects. The specimens were irradiated at 450°C with heavy ions and protons at Michigan Ion Beam Laboratory to achieve the desired damage depth and dose (dpa).

An oxidation experiment was conducted on the irradiated and Cr-coated specimens alongside as-received Zr-2.5Nb and Cr-coated (but unirradiated) specimens. Specimens were exposed to conditions relevant to the SCW-SMR at 500°C and 23.5 MPa with 630 µg/kg oxygen in the feed water in a refreshed autoclave system. After 500 hours, specimens were removed from the autoclave, then dried and reweighed. The weight gain results are shown in Figure 3.14. While the Cr-coated PT specimen shows a 50% decrease in weight gain over the as-received PT specimen, the Cr-coated and irradiated specimen shows a 30% increase over the Cr-coated unirradiated specimen.

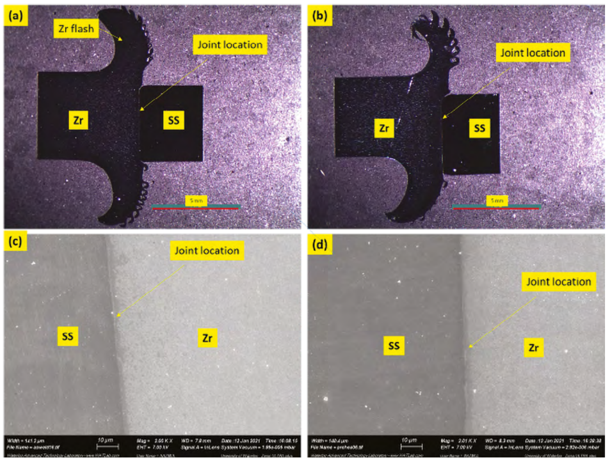
Radiolysis of supercritical water

The effects of radiation on corrosion extend beyond material damage and microstructural changes to environmental effects through the radiolytic splitting of water into highly oxidising species (e.g. OH, e-aq, H, H_2O_2). These species affect the redox environment in which materials corrode, changing the rates of detrimental reactions, including the reactive evaporation of chromia, an important protective oxide. Downstream of the SCWR core, the only stable products of radiolysis are H_2 and O_2, but this can still create a highly oxidising environment for the outlet plenum, main steam line and turbines in a direct-cycle SCWR. Conventional pressurized reactors add hydrogen to suppress "net radiolysis" (the net production of H_2 and O_2 by radiation) and reduce the corrosion potential.

The CNL has been working to bring an irradiated supercritical water rig online. The rig delivers supercritical water at pressures up to 27.5 MPa and temperatures up to 537°C to a target aligned with the CNL's Van de Graaff electron accelerator. In November 2022, modifications were completed and the system passed hydrostatic tests. The system is now capable of injecting up to 700 mL/kg of H_2. Scientists and technologists at the CNL's Van de Graaff accelerator facility began testing the system to gain experience. Preliminary tests were conducted with 400-460°C water at 26 MPa to see if net radiolysis could be observed and, by the addition of hydrogen, suppressed. Net radiolysis was observed across this temperature range, producing O_2 above 2.5 ppm (above the instrument's detection limit). Through the addition of ~25 mL/kg of H_2, however, the O_2 concentration rapidly declined and eventually reached a baseline concentration of 20 parts per billion. Better methods to sparse oxygen from the injection tanks need to be implemented, and work is still needed to improve quantification of many parameters, including dissolved gas concentrations, test section temperature and deposited dose.

Joining of dissimilar metals

A zirconium-stainless steel transition is important to the pressure tube design of the Canadian SCWR. Joining dissimilar materials is often more challenging than joining the same materials because of differences in mechanical and chemical properties. The overall objective of the current work is to fabricate joints by joining tubes of Zirconium (Zr) and 304 stainless steel by rotary friction welding and co-extrusion techniques.



Note: Both samples were produced at 1 200 rpm, using a burn off pressure of 59.5 MPa and a burn-off length of 1.0 mm, but sample 7 was preheated to 600°C for 30 minutes before welding.

Source: Canadian Nuclear Laboratories (CNL).

Figure 3.15. Optical images of the welded joints in a cross-section of (a) sample without SS preheat and (b) sample with SS preheat, as well as SEM micrographs of the joints

CNL's prior work showed results from joining of Zr to 304 SS rods by friction welding and co-extrusion. Preliminary results from tube joints made from linear friction welding were reported in 2021. In 2022, the work continued, with the goal to produce defect-free tube joints of Zr to 304 SS by rotary friction welding, improve the mechanical integrity of the joints through thermal annealing, and study the microstructure and chemical distribution to better understand how the elements diffuse during friction welding.

Figure 3.15 shows examples of two rotary friction-welded specimens: macro images of the joints produced on Sample 4 (without SS preheat) and Sample 7 (with SS preheat). The zirconium tube

Figure 3.14. Zr-2.5Nb pressure tube samples before and after exposure to 500°C and 25 MPa supercritical water for a total of 500 hours

	As-received	Coated	Coated and Irradiated
Weight gain	361.5 mg/dm^2	188.3 mg/dm^2	247.8 mg/dm^2
Before			
After			

Source: Canadian Nuclear Laboratories (CNL).

deformed significantly, while no observable deformation occurred on the SS side. The macro image of Sample 7 may indicate some misalignment, which may have originated from the welding machine fixture tolerances or a fit-up error. The SEM images indicate uniform reaction layers at the interfaces in both samples, with and without preheat. The apparent reaction layer thicknesses by optical analysis were 1.90 µm (Sample 4, without preheat) and 2.04 µm (Sample 7, with preheat).

The tensile tests' results were also completed for different welding conditions. The similar SS-SS joint exhibits a yield strength of 456 MPa, with an ultimate strength of 623 MPa, while the similar Zr-Zr joint exhibits a yield strength of 251 MPa with an ultimate strength of 452 MPa. The dissimilar metal-welded joint produced a maximum yield strength of 299 MPa with an ultimate tensile strength of 403 MPa, and a highest elongation of 3.2%. The ductility of the dissimilar metal joints is lower than the Zr-Zr weld joint. The stiffness of the elastic region of the dissimilar metal joint is closer to the similar SS-SS joint rather than the Zr-Zr joint.

Armando Nava-Dominguez
Chair of the SCWR SSC, with contributions from SCWR members

References

Beuthe, T., A. Vasić, H. Zahlan and A.N. Domínguez (2020), "Integration of modeling capabilities in CATHENA for supercritical water reactors", *Journal of Nuclear Engineering and Radiation Science*, Vol. 6/3, https://doi.org/10.1115/1.4047288.

Nava-Dominguez, A., Y.F. Rao and G.M. Waddington (2014), "Assessment of subchannel code ASSERT-PV for flow-distribution predictions", *Nuclear Engineering and Design*, Vol. 275, pp. 122 132, https://doi.org/10.1016/j.nucengdes.2014.05.001.

Brogna, C. (2017), CFD Modelling of Heated Rod Bundles with Water at Supercritical Pressure, Master's thesis, University of Pisa.

Chen, J., S.-G. Yang, R. Zhao and W.-L. Cheng (2022), "Experimental study on the effect of wall roughness on heat transfer characteristics of supercritical carbon dioxide in vertical tubes", *International Journal of Heat and Mass Transfer*, Vol. 196, pp. 123258, https://doi.org/10.1016/j.ijheatmasstransfer.2022.123258.

Guzonas, D., R. Novotny, S. Penttilä, A. Toivonen and W. Zheng (2018), *Materials and Water Chemistry for Supercritical Water-cooled Reactors*, Woodhead Publishing, Elsevier Ltd.

Hales, J.D., R.L. Williamson, S.R. Novascone, G. Pastore, B.W. Spencer, D.S. Stafford, K.A. Gamble, D.M. Perez and W. Liu (2014), " Cladding corrosion model", Chapter 15, *BISON Theory Manual, The Equations Behind Nuclear Fuel Analysis*, INL/EXT-13-29930 Rev. 1.1, BISON Release 1.1, Idaho National Laboratory, https://inldigitallibrary.inl.gov/sites/sti/sti/5842314.pdf.

Kiss, A., T. Vágó and A. Aszódi (2015), "Numerical analysis on inlet and outlet sections of a test fuel assembly for a supercritical water reactor", *Nuclear Engineering and Design*, Vol. 295, pp. 415-428, http://dx.doi.org/10.1016/j.nucengdes.2015.10.008.

Lee, D., B. Elward, P. Brooks, R. Umretiya, J. Rojas, M. Bucci, R.B. Rebak and M Anderson (2021), "Enhanced flow boiling heat transfer on chromium coated Zircaloy-4 using cold spray technique for accident tolerant fuel (ATF) materials", *Applied Thermal Engineering*, Vol. 185, https://doi.org/10.1016/j.applthermaleng.2020.116347.

Pioro, I.L., R.B. Duffey and T. Dumouchel (2004), "Hydraulic resistance of fluids flowing in channels at supercritical pressures (survey)", *Nuclear Engineering and Desig*n, Vol. 231/2, pp. 187-197, https://doi.org/10.1016/j.nucengdes.2004.03.001.

Pioro, I.L. and R.B. Duffey (2007), *Heat Transfer and Hydraulic Resistance at Supercritical Pressures in Power Engineering Applications*, ASME Press, https://doi.org/10.1115/1.802523.

Razumovskiy, V.G., E.N. Pis'mennyi, K. Sidawi, I.L. Pioro and A.E. Koloskov (2016), "Experimental heat transfer in an annular channel and 3-rod bundle cooled with upward flow of supercritical water", *Journal of Nuclear Engineering and Radiation Science*, Vol. 2/1, https://doi.org/10.1115/1.4031818.

Wu, G., H. Wang and Q.C. Bi (2022), "Experimental investigation of the pressure drop and friction factor of supercritical water in a 2 x 2 rod bundle", *Annals of Nuclear Energy*, Vol. 166, pp. 108732, https://doi.org/10.1016/j.anucene.2021.108732.

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 Alternative Energies and Atomic Energy Commission, France; the Department of Energy, United States; the Joint Research Centre, Euratom; the Japan Atomic Energy Agency (JAEA); the Ministry of Science and Information and Communication and Technology, Korea; the China National Nuclear Corporation; Rosatom, Russia; and the Department for Business, Energy and Industrial Strategy, 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 the 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 are highly compatible with sodium, 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. Table 3.2 summarizes the typical design parameters of the SFR concept being developed in the framework of the Gen IV System Arrangement. Plant sizes ranging from small modular systems to large monolithic reactors are being considered.

Several system options that define the general classes of SFR design concepts have been identified for Gen IV SFR research collaboration: 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 SFR (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, modular construction, compact systems and components, advanced fuels and materials, and refined safety systems. They are thus used to guide and assess Gen IV SFR R&D collaborations.

Table 3.2. Typical design parameters for the Gen IV sodium-cooled fast reactor

Reactor parameters	Reference value
Outlet temperature	500-550°C
Pressure	~1 atmosphere
Power rating	30-5 000 MWth (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: MWth = megawatt thermal; Mwe = megawatt electrical; ODS = oxide dispersion-strengthened; GWD/MTHM = gigawatt days per metric tonne of heavy metal.

Industry engagement and near-term demonstrations

The SFR SSC hosted a panel session for SFR demonstrations at the G4SR conference GIF Industry Forum in October 2022 in Toronto and invited international SFR companies to present and discuss potential collaboration opportunities with the GIF to accelerate the demonstration of Gen IV SFRs. There was significant interest from the three companies who attended (TerraPower, Oklo and ARC Clean Technology) in collaborating through existing or new GIF projects. The common topics of interest included: sharing experimental data, simulation benchmarks, fuel qualification, closing the fuel cycle, utilising research infrastructure, instrumentation and energy storage/conversion systems. Follow-up discussions with the individual SFR companies have taken place to outline a concrete path forward for collaboration with the GIF. More details of this industry engagement meeting are available at: https://gif2022.process.y-congress.com/ScientificProcess/Schedule/?setLng=en.

In the near term, SFR demonstrations continue to be developed in several GIF countries. In China, two CFR-600 units, which are demonstration SFR plants generating 600 MWe each, are being constructed in Xiapu County, Fujian Province. Construction of Unit 1 started in 2017 with anticipated grid connection in 2023. Construction of Unit 2 started in early 2021 (WNN, 2020).[1] In the United States, the first

1 https://world-nuclear-news.org/Articles/China-starts-building-second-CFR-600-fast-reactor.

Natrium demonstration plant by TerraPower is planned for construction in Kemmerer, Wyoming, with anticipated submission of the plant's construction permit application to the US Nuclear Regulatory Commission in 2023.[2] Natrium is a 345 MWe SFR with a molten salt-based energy storage system to be started up with high assay low-enriched uranium fuel.

Technical highlights – System Integration and Assessment Project

Through a systematic review of the technical projects and relevant contributions on design options and performance, the System Integration and Assessment (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.

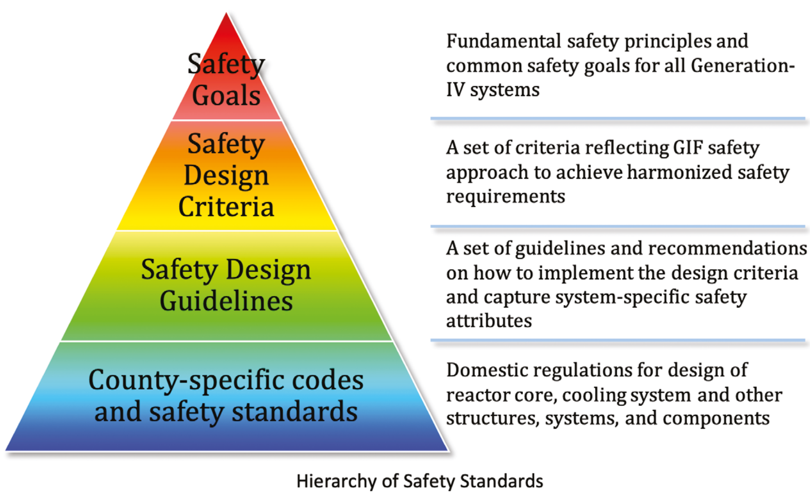
From 2010 to 2019, the CEA developed the ASTRID-600 SFR demonstration reactor. One of its high-level objectives was to fulfil the Gen IV safety requirements. In parallel, the GIF issued a set of criteria reflecting the GIF safety approach to achieve harmonized safety requirements for SFRs (safety design criteria, SDC) and a set of guidelines on how to implement the design criteria for SFRs (safety design guidelines, SDG) (Figure 3.16). In 2022, the CEA started examining the application of the GIF SDC and SDG to the ASTRID-600 Reactivity Control Systems. This work will be continued in 2023-24 with an examination of SDC and SDG applications to the decay heat removal systems and the prevention of significant energy release during a core damage accident. The objectives of this work are to draw recommendations on how to apply the GIF SDC-SDG to a specific SFR design and to highlight the potential gaps between the ASTRID concept and the GIF recommendations. This work is performed in collaboration with the JAEA, which will examine the application of the SDC and SDG to the JSFR systems.

Euratom reported on the developments within European SFR projects. The Horizon 2020 ESFR-SMART project (European Sodium Fast Reactor Safety Measures Assessment and Research Tools) ended recently and a follow-up project, ESFR-SIMPLE (Safety by Innovative Monitoring, Power Level flexibility and Experimental research), recently began. The aim is to further enhance the safety and performance of the ESFR concept.

Korea contributed an SFR core concept – SALUS (Small, Advanced, Long-cycled and Ultimate Safe SFR). The long fuel cycle was studied for electricity generation based on the Prototype Generation-IV Sodium Fast Reactor (PGSFR) design approach. To achieve ~20 years' fuel cycle length, the power rating was decreased from 150 MWe to 100 MWe and the fuel inventory was increased by adopting a longer active core length and larger fuel assembly within the same PGSFR barrel. For effective fuel breeding without blankets and peak burnup reduction, several enrichment zoning configurations were sought and fuel performance analysis to predict cladding failure. Core thermal-hydraulic design was performed to satisfy the fuel design limit during ~20-year operation by the iteration of whole pin cumulative damage fraction evaluation and flow rate distribution. A Monte Carlo code, McCARD, was selected to reduce the cumulative error of the depletion calculations. Using

2 https://natriumpower.com.

Figure 3.16. Hierarchy of safety standards established by the GIF



this code, the parameters of the core characteristics such as reactivity coefficients and control rod worth were produced for safety analysis.

The US national laboratories, on behalf of the US DOE, performed time-dependent fuel cycle scenario simulations to inform on the transition of the current US LWR nuclear fleet to a future reactor fleet that includes Gen IV SFRs. The scenarios were modelled using fuel cycle systems' codes that informed on the fuel cycle facility capacity and fuel material requirements to support a large-scale deployment of SFRs under various fuel cycle assumptions (once-through HALEU, recycling, etc.).

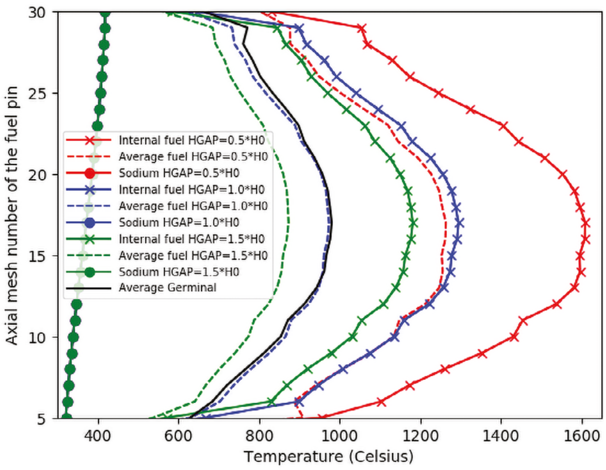
The JAEA investigated a pool-type reactor with a JSFR core as one of the variations of the system configuration. A combination of the JSFR core and a pool-type reactor vessel were evaluated and sketches of major components were provided. Based on those sketches, the construction cost of the alternate pool-type reactor system with the JSFR core was evaluated and found to be similar to that of the loop configuration JSFR.

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 programmes 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 301/302R tests and Phenix dissymmetric test – which began in the last quarter of 2019. The first phase of the benchmark analysis was a "blind phase" without comparison with experimental data. After the blind phase, an "open phase" has begun to compare the experimental data for phenomenological understanding for the EBR-II benchmark study. Argonne National Laboratory, the CEA, the JAEA and KAERI completed 1-D calculations by 2022. The CEA and the JAEA started additional calculations for EBR-II analysis using a 1-D calculation code coupled with a 3-D CFD code in 2021. For the Phenix dissymmetric test benchmark, the JAEA completed the blind-phase calculation and started the open-phase calculation in 2022.

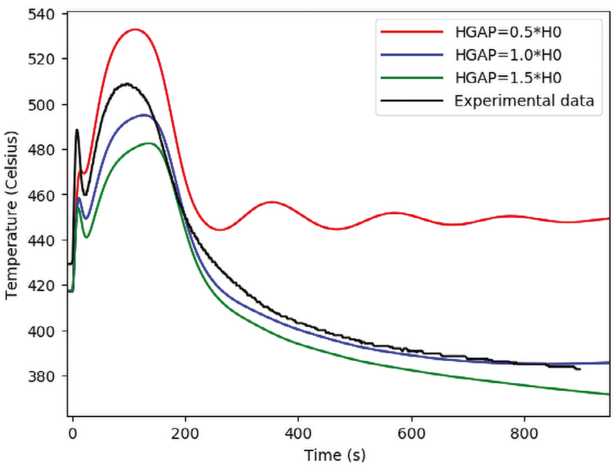
The CEA presented a study regarding the influence of the fuel pellet-clad heat exchange coefficient on a loss-of-flow without scram transient in the Fast Flux Test Facility (FFTF) reactor, simulated with the CATHARE3 system code. In the FFTF safety test, primary pumps are tripped while the core power is 50% of the nominal core power and the flowrate is 100% of the nominal flowrate; the reactor scram is disabled and secondary pumps are kept in operation. Data are from an IAEA benchmark. In the CATHARE3 core model, the gap between the fuel pellets and the clad is represented by a radial mesh with a width $e(z)$ and a heat exchange coefficient $h_{gap}(z)$ that are calculated with the GERMINAL V2 code for each group of sub-assemblies. h_{gap} is in W/(m^2.K). The objective of the study was to assess the influence on the transient results of the h_{gap} value at the beginning of the transient and of time-dependent h_{gap} value during the transient. Figures 3.17 and 3.18 present the results of the sensitivity study of the h_{gap} value for the FFTF transient. For this sensitivity study, three values of the h_{gap} at the beginning of the transient are considered: H0 (value computed by GERMINAL V2), 0.5*H0 (minimization of the heat transfer coefficient) and H0*1.5 (maximization of the heat transfer coefficient).

Figure 3.17. Fast Flux Test Facility fuel maximum and mean axial temperature for three different values of the h_{gap} (fuel-clad heat exchange coefficient)



Source: CEA.

Figure 3.18. Sodium core outlet temperature evolution during the Fast Flux Test Facility loss-of-flow without scram transient for three different values of the h_{gap}



Source: CEA.

With a smaller hgap value (0.5*H0) at the beginning of the transient, the fuel temperature is higher than with the normal value (H0). The fuel temperature drop is more important at the beginning of the transient (the core power decreases at the beginning of the transient thanks to reactivity feedbacks). Doppler and fuel dilatation feedbacks are higher (reactivity insertion in this kind of transient), so that the core total reactivity and the resulting power are higher. The sodium temperature at the core outlet is higher than the temperature with the normal value of the fuel-clad heat exchange coefficient. In contrast, with a higher initial hgap value, the sodium temperature at the core outlet is lower than the temperature with the normal value of the fuel-clad heat exchange coefficient. Two cases were considered for the sensitivity study of the fuel-clad heat exchange coefficient evolution during the FFTF LOFWOS transient. In the first case, the h_{gap} was kept constant during the transient (at its normal value at the beginning of the transient); in the second case, the h_{gap} was decreased during the transient (as computed by the GERMINAL V2 code). This sensitivity study showed that time-dependent hgap effect was not significant in the FFTF LOFWOS transient because of the gas expansion modules that lead to a fast decrease of the reactivity at the beginning of the transient. To conclude, an accurate modelling of the fuel-clad gap thermal conductance in the initial state of the transient and during the transient is important because of its influence on the thermal behaviour of the core as well as on the neutron feedbacks (in particular, the Doppler and fuel axial expansion effects).

Euratom performed an assessment of the transition from forced to natural circulation during selected accidents with analyses of the ESFR behaviour in protected loss of flow and protected station blackout accidents. The first part of the study aimed to evaluate numerically the thermal-hydraulic behaviour of the ESFR primary and secondary systems under accident conditions initiated by a trip of primary and secondary pumps and reactor scram. The second part of the study evaluated the ability of system codes used for safety analysis to predict the complex flow patterns during the evaluated transients by comparing their predictions with more accurate CFD codes. Indeed, the pool-type configuration of the ESFR primary system makes the 3D flow pattern and temperature stratification during transition from the full-power forced convection regime to the decay-heat natural convection conditions important and therefore imposes corresponding requirements to the calculation tools used for the numerical analysis. Furthermore, the study has shown that the ESFR behaviour in protected loss-of-flow and protected station blackout transients are likely to result in a smooth transition from forced to the natural convection, allowing a long-term safe decay heat removal.

In an additional study devoted to the primary pumps' operation and capability of preventing boiling sodium, Euratom performed assessments of the performance of ESFR primary pumps, the decay heat removal system, the reactor passive shutdown systems and the primary system as a whole under selected accident conditions. The study aimed to investigate the capability to prevent sodium boiling in the core upper plenum region and the potential benefits of enhanced natural circulation of sodium, e.g. in the secondary and decay heat removal loops, using the innovative thermal pump features. Further assessments were carried out to analyze the performance of the innovative concept of hydraulic diode installed at the outlets of the primary pumps or inside the diagrid. The extensive simulations performed within this activity have provided a broad assessment of the capability of the refined primary pumps to prevent sodium boiling in the core upper plenum region and new insights of the sodium fluid dynamics in both primary and secondary circuits.

The JAEA has conducted a Level 1 probabilistic risk assessment for external vessel storage tank of Japan sodium-cooled fast reactor in scheduled refuelling. Spent fuels are transferred from a reactor core to a spent fuel pool through an external vessel storage tank filled with sodium in SFRs in Japan. Based on the design information, the probabilistic risk assessment has identified initiating events, event and fault tree analyses, human error probability analysis, and quantification of accident sequences. Fuel damage frequency of the external vessel storage tank was evaluated at approximately 10^{-6}/year. By considering the secondary sodium freezing, the fuel damage frequency was twice increased. The dominant accident sequence resulted from the common cause failure of the damper opening and/or the human error for the switching from standby to operation mode in the three standby cooling circuits. The second dominant accident sequence following the secondary pump trip is sodium freezing caused by the failure of air blower trip in the air cooler due to the common cause failures of secondary sodium flowmeter failure or erroneous opening of the air cooler damper.

The JAEA has also improved the steam generator tube failure propagation analysis code LEAP for evaluating overheating rupture. The JAEA developed an evaluation model for overheating rupture in the existing code to evaluate the rate of water leakage due to wastage-type tube failure propagation and confirmed its predictive capability through numerical analysis of tube failure propagation experiments.

As an extension of the STELLA-2 test conducted in 2021, KAERI performed decay heat removal system cooling capability tests for representative accident conditions of PGSFR. In the case of accidents such as loss of flow, loss of heat sink and pipe break, the system transients according to decay heat removal system capacity were simulated. The experiment and analysis results, as well as the operation experience of the large-scale experimental equipment, will be shared.

In KAERI, the open phase analysis for the EBR-II benchmark problem is being performed using the GAMMA+ (General Analyzer for Multi-component and Multi-Dimensional Transient Application) code to verify the domestic safety analysis code and methodologies. In 2023, open phase analysis results and methodologies will be shared.

Technical highlights – Advanced Fuels Project

The Advanced Fuels project aims to develop and demonstrate minor actinide-bearing (MA-bearing) high burn-up fuels for SFRs. The R&D activities of the Advanced Fuels 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/ducting 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 oxide dispersion-strengthened steels.

The Advanced Fuels project consists of three work packages: 1) SFR non-MA-bearing driver fuel evaluation, optimization and demonstration; 2) MA-bearing transmutation fuel evaluation, optimization and demonstration; and 3) high-burn-up fuel evaluation, optimization and demonstration.

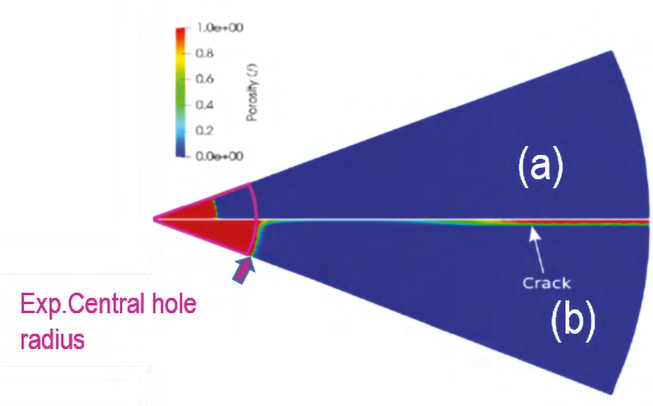
The French CEA is continuing to work on improving the simulation and optimization of SFR fuel and subassembly. Concerning the oxide fuel simulation, a new engineering-scale model describing the migration of porosity in a fuel pellet under thermal gradient has been proposed. The pore-advection equation is coupled with thermal equilibrium via the dependency of the fuel thermal conductivity and of the volumetric heat source on the local porosity. Stabilization techniques are used to reduce spatial oscillations of the solutions. The model is applied to analyze the contribution of as-fabricated and crack-induced porosities in determining fuel restructuring and central hole formation (Figure 3.19). Effect of crack-induced porosity must be considered for simulating these phenomena.

Concerning the hexagonal duct manufacturing, a finite-element model has been used to understand the force peak observed during the first industrial hot extrusion test. A constitutive model has been selected from the literature and surface properties have been considered time- and space-dependant to represent glass lubrication. Model predictions are compared to experimental measurements, suggesting that the difficult start of the glass melting and flow along a cooled die affects the force peak. Model improvements, in particular bimaterial and mesh adaptations, have been proposed to go further in the understanding and prediction of lubrication effect on hot extrusion.

Metal fuel has been developed for the PGSFR in KAERI, where the fuel assembly is being designed to satisfy its requirements for core performance and safety. Stress limits for each event category were established for this. Also, the structural analysis of the fuel assembly was carried out in the case of normal operating conditions. The details of the design analysis results of normal operating conditions for PGSFR fuel assembly was described. Sintered $LaYO_3$ pellet was being fabricated using cold isostatic press followed by a sintering process to develop reusable crucible for metal fuel cast. Sintering temperature conditions were controlled to increase the density of the pellet and fabricate perovskite $LaYO_3$ phase structure. Interaction preventing the effect of the sintered body has been investigated using a sessile drop test, where U-Zr-RE melt was fabricated for the sessile drop test. Interaction behaviours and defects of the ceramic body were analyzed after the droplet test.

KAERI also develops barrier cladding tube technology for suppressing fuel-cladding chemical interactions for the use of minor actinide-bearing metal fuel. Electroplating is considered to be one of the most probable options on account of its applica-

Figure 3.19. Final results of the porosity distribution, considering an initial condition having both as-fabricated and crack-induced porosity (b) and only as-fabricated porosity (a)



Note: Each circular sector represents a separate case.

Source: Barani et al. (2022).

bility to the cladding inner surface. Chromium has been selected because of its higher resistance to fuel-cladding chemical interaction as well as its higher technical maturity. The work has been focused by optimising the Cr plating process to suppress the incipient cracks inside the Cr layer to enhance barrier integrity. Optimization of the plating parameters along with the post-plating treatment was described in this work.

The JAEA measured thermal conductivity for near stoichiometric $(U_{0.7-z}Pu_{0.3}Am_z)O_2$ (z = 0.05, 0.10, and 0.15) between room temperature and 1 473 K under oxygen partial pressure-controlled atmosphere. The JAEA conducted experiments on innovative component technologies for SFR MOX fuel fabrication, such as microwave denitration with flashing prevention and field-assisted sintering.

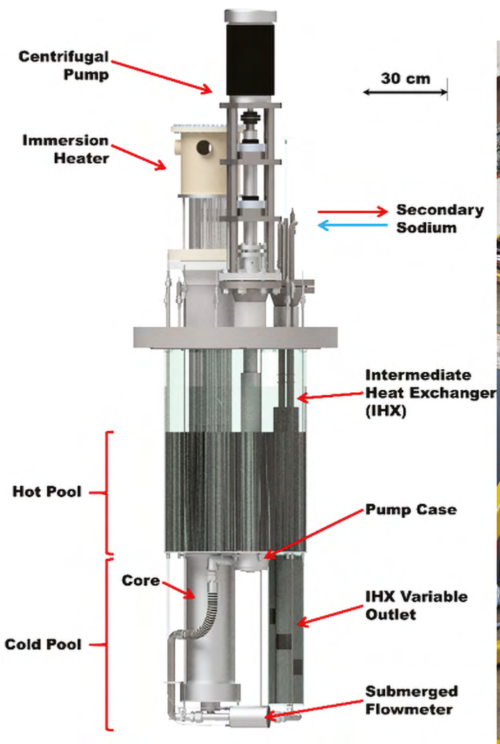
The China Institute of Atomic Energy has established a programme on key technology research of U-Pu-Zr metal fuel on the base of U-Zr metal fuel research. The programme carries out research of U-Pu-Zr metal fuel and subassembly design, metal fuel model research and fuel performance code development, and metal fuel fabrication to support the construction of an integral fast reactor in the future. The metal U-Pu-Zr mental fuel model has been investigated and studied. The U-Pu-Zr metal fuel assembly has been preliminarily designed, including the preliminary structure design of fuel assembly and the basis analysis of fuel pin and the fuel assembly.

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 in-service inspection and repair 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.

In 2022, the DOE described the initial operations of a new experimental test article called the Thermal Hydraulic Experimental Test Article (THETA), an electrically heated pool plant that will be used for validating thermal-hydraulic codes. The THETA 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 (Figure 3.20). In addition, also in 2022, the DOE shared the development of an inductive-level sensor that was tested at the METL facility.

Figure 3.20. THETA test article schematic and assembly on test stand



Source: Argonne National Laboratory (ANL).

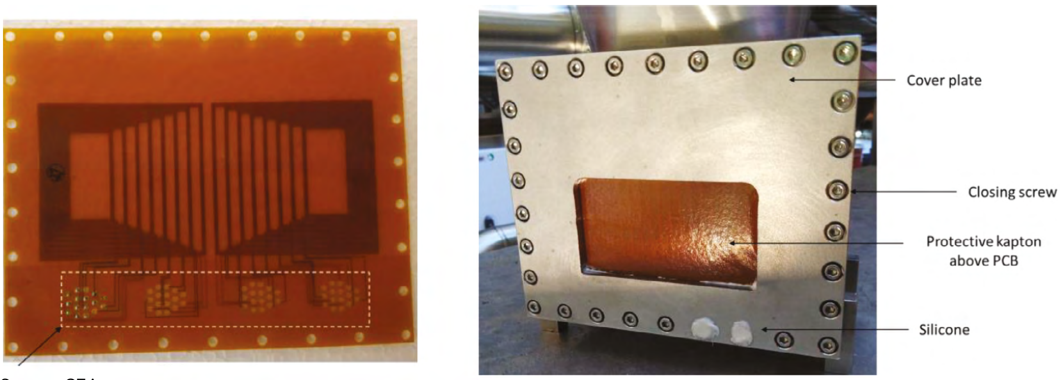
The CEA studies the optimization of an ultrasonic linear phased array probe for selected under sodium viewing applications (opened cracks detection, subassembly identification, lost piece detection) using full matrix capture and total focusing method reconstruction imaging. Optimization was studied by simulating the imaging performances on five different target cases using the CIVA software. Several criteria were used to evaluate the quality of the reconstructed images; a probe design with 64 elements was identified for the under sodium imaging. The CEA has continued with the development of electromagnetic acoustic transducer (EMAT) phased arrays for SFR inspection (Figure 3.21). An EMAT, whose technology is based on the use of a magnetic field coupled with eddy currents, is very difficult to implement because it is necessary to make many compromises on:

- the size of the coils, which must be large at transmission but rather narrow at reception;
- the size and spacing of the magnets in order to generate a magnetic field of high amplitude but as homogeneous as possible between the magnets;
- the pitch, which must be small enough to avoid the creation of grating lobes but whose minimum size is quickly limited by the technology;
- the aperture of the sensor, which must be large enough to allow for distant focusing while keeping the pitch small enough.

A progressive adjustment of these different parameters has allowed the development of various EMAT prototypes over time, which have benefited from the improvement of the multi-element technology. Thus, the feedback and the help of simulation (the models of the CIVA software platform evolving in parallel) have enabled us to move from commercial single-element sensors to 16-element multi-element sensors whose performances in the laboratory and in sodium are very encouraging. At the same time, the electronic chain required to drive the sensor has evolved to drive 16 elements and be able to adapt the impedances of each coil. Important work also consisted in adapting the case to make it resistant to temperature and sodium and above all watertight to avoid any penetration of sodium and thus any possible degradation of the sensor's active elements (coil, magnet, cable, etc.).

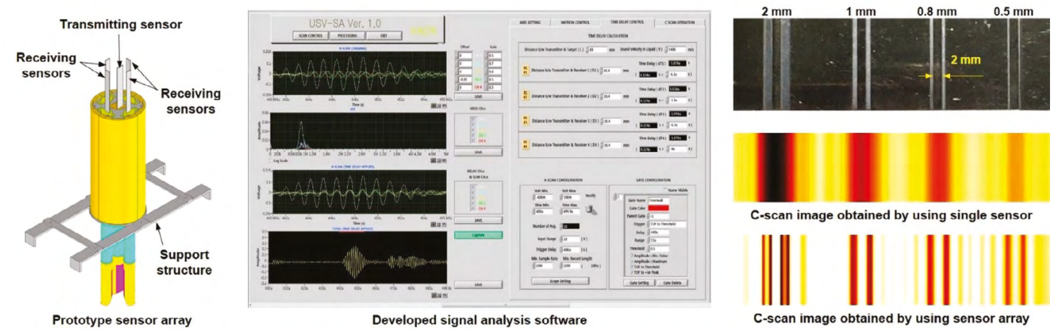
In 2022, as the second step in the development of the plate-type ultrasonic waveguide sensor array, KAERI designed and fabricated a prototype sensor array consisting of a single transmitting waveguide sensor and four receiving waveguide sensors based on the preliminary study results conducted in 2021. KAERI also developed signal analysis software adopting the time delay function to process multiple signals simultaneously measured by the prototype sensor array for a given target point. Then, several underwater viewing tests were conducted to evaluate the feasibility of the prototype sensor array and signal analysis software (Figure 3.22).

Figure 3.21. Photos of the Kapton and the EMAT 2019



Source: CEA.

Figure 3.22. Underwater feasibility test of prototype sensor array and signal analysis software



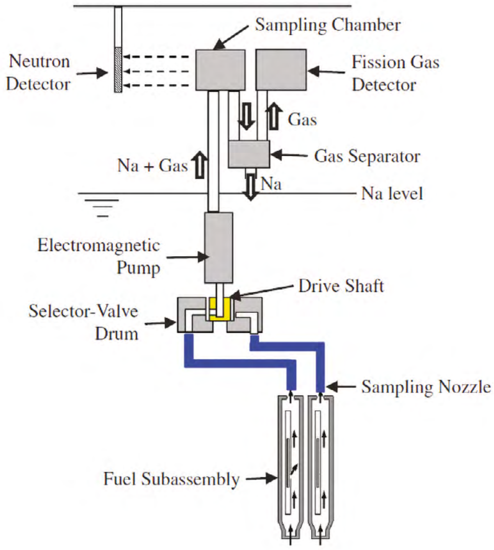
Source: KAERI.

KAERI continued a study to develop advanced sodium instrumentation techniques based on high-fidelity distributed temperature sensor and ultra-sonic technology. For the purpose of measuring and monitoring various process variables in a sodium system, a concept for a high-fidelity distributed temperature sensing system has been provided and a prototype reference component for that has been fabricated.

The JAEA conducted the development of failed fuel detection and location (FFDL) system for the early detection of fuel failure and precise identification, which is important for the safe operation of SFR (Figure 3.23). The selector valve type FFDL system samples sodium from each fuel subassembly outlet using a selecting mechanism of the sampling lines during reactor operation. The identification of failed fuel subassembly is made by detecting fission product gas or delated neutron precursors for each sampling line. The endurance of the selector valve is important for this type of FFDL system. To demonstrate the manufacturability and endurance of selector valve, a full-size mock-up was manufactured, and the endurance experiment under sodium was conducted.

Figure 3.23. Schematic diagram of selector valve FFDL System

Source: JAEA.



Yoshitaka Chikazawa
Chair of the SFR SSC,
with contributions from SFR members

References

Barani, T., I. Ramiere and B. Michel (2022), “Analysis of fabrication and crack-induced porosity migration in mixed oxide fuels for sodium fast reactors by the finite element method,” *Journal of Nuclear Materials*, Vol. 558, Article 153341, January 2022, https://doi.org/10.1016/j.jnucmat.2021.153341.

WNN (2020), “China starts building second CFR-600 fast reactor”, *World Nuclear News*, https://world-nuclear-news.org/Articles/China-starts-building-second-CFR-600-fast-reactor.

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 in 2016 for ten years. The current signatories are Australia, Canada, China, Euratom, France, Japan, Korea, Switzerland, the United Kingdom, and the United States. The VHTR System Research Plan outlines four active projects with the following members and observers:

- VHTR fuel and fuel cycle: China, Euratom, France, Japan, Korea and the United States as members.
- VHTR materials: China, Euratom, France, Japan, Korea, Switzerland and the United States as members. Canada and the United Kingdom will become members pending the approval of the 3rd Project Arrangement in 2023.
- VHTR hydrogen production with China, Canada, Euratom, France, Japan, Korea and the United States as members.
- VHTR computational methods validation and benchmarks with China, Euratom, Japan, Korea and the United States as members. Canada and the United Kingdom are currently observers. The Project Arrangement became active for the first time in December 2022.

High core outlet temperatures enable high efficiencies for power conversion and hydrogen production, as well as co-generation of high steam qualities (superheated or supercritical). Current VHTR R&D focuses on the demonstration of inherent safety features and high fuel performance, hydrogen production, the validation of new computational methods and code developments, coupling with process heat applications, co-generation of heat and power, and the resolution of potential conflicts between these challenging goals.

Main characteristics of the system

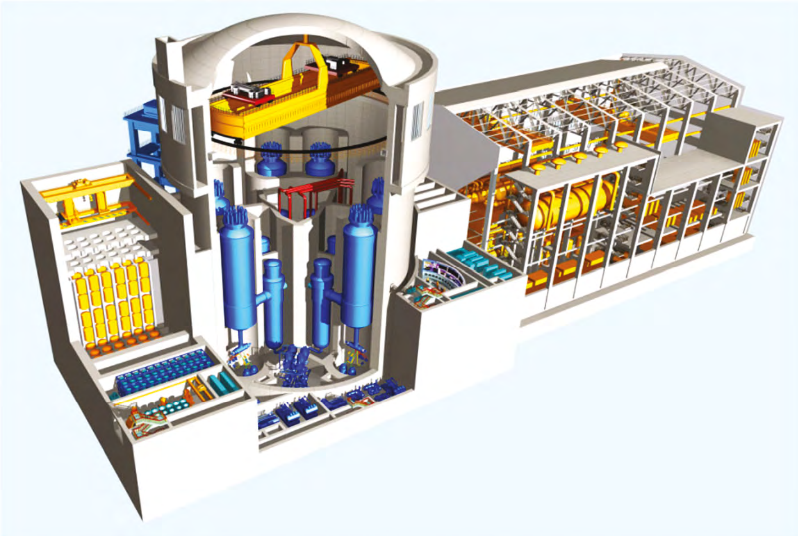
High-temperature gas-cooled reactors (HTRs) are helium-cooled graphite-moderated nuclear fission reactors using fully ceramic tri-structural isotropic (TRISO)-coated particle-based fuels. HTRs are characterized by inherent safety features, excellent fission product retention in the fuel and high-temperature operation suitable for generation 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 greater efficiency and more versatile use by further increasing the helium outlet temperature to 1 000°C or higher. Above 1 000°C, however, HTRs will require new structural materials, especially for the intermediate heat exchanger.

The initial driver for VHTR development in the GIF was thermo-chemical hydrogen production with the sulfur-iodine cycle requiring a core outlet temperature of approximately 950°C. The operational envelope of HTRs can be adapted to specific end-user needs, and recent experience at the 2022 GIF Forum indicated that a large near-term market exists for process steam of approximately 400-550°C, achievable with lower temperature HTR designs. Inherent safety in accident conditions is assured by the low power densities and high thermal inertia of typical HTR designs, and the potential for high fuel burnup (150-200 gigawatt days per metric tonne of heavy metal [GWd/tHM]), high efficiency, as well as modular construction, all constitute advantages favouring commercial HTR deployment.

The HTR standard fuel form is based on UO_2 TRISO-coated particles (UO_2 kernel, buffer/inner pyrocarbon/SiC/outer pyrocarbon coatings) embedded in a graphite matrix, which is then formed either into 6 cm diameter pebbles or cylindrical compacts embedded in hexagonal fuel blocks. This fuel form exhibits a demonstrated long-term temperature tolerance of 1 600°C in accident situations with sufficient safety margin. Recent research has shown that the safety performance may be further enhanced through the use of a uranium-oxycarbide fuel kernel, a zirconium carbide (ZrC) coating layer instead of silicon carbide (SiC), or the replacement of the graphite matrix material with SiC. The current HTR fuel cycle is once-through, very high burnup, low-enriched uranium fuel cycle, but solutions to adequately manage the back end of the fuel cycle are under investigation, and significant research is being performed internationally on TRISO and graphite waste processing. 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 codification of new materials and manufacturing methods.

Industry engagement and demonstration projects overview

The VHTR SSC hosted a panel session for HTGR demonstrations at the G4SR conference GIF Industry Forum in October 2022 in Toronto, with the objective of assessing the potential scope for multilateral collaboration programmes that can benefit industry and the GIF, and how to build bridges between the GIF VHTR system R&D community and industry efforts. There was significant interest from the four companies who attended (BWXT, U-battery, USNC, X-energy) in collaborating through existing or new GIF projects. The session was attended by approximately 80 HTR experts. Potential collaboration topics included benchmark studies for TRISO particle performance, fission product transport codes, fuel qualification method development, access to metallic and graphite data contained in the *Materials Handbook*, and demonstration of hydrogen technologies. Industry also showed interest in obtaining access to existing test facilities, qualification of advanced design features, e.g. helium/molten salt heat exchangers, and access to existing TRISO particle performance and graphite R&D data. Follow-up discussions with the individual HTGR companies have taken place to outline a concrete path forward for



Source: Zhang et al. (2022).

Figure 3.24. 3D design of the HTR-PM600 nuclear power plant

collaboration with the GIF. More details of this industry engagement meeting can be found at: https://gif2022.process.y-congress.com/ScientificProcess/Schedule/?setLng=en.

In the near term, several demonstration projects are being pursued to meet the needs of current industries interested in early applications. Following the first grid connection in December 2021, the high-temperature gas-cooled reactor – pebble-bed module (HTR-PM) reached its initial full power with stable operation on 9 December 2022. Significant progress was also made in 2022 on the detailed design of the HTR-PM600, a 600 MWe commercial plant with six modules (Figure 3.24), with the preliminary safety analyses report submitted to the regulatory body for safety review.

The High Temperature Test Reactor (HTTR) in Japan was restarted in 2021 after a decade of shutdown following the Great East Japan Earthquake in 2011. In addition to the reactor restart in 2021, a simulated loss-of-coolant test was performed in 2022 in which all cooling systems were shut down with the reactor at 30% power and no control rod actuation. The reactor power decreased spontaneously and the fuel temperature did not rise abnormally, providing an effective demonstration of HTR passive safety behaviour with no operator intervention. A further loss of forced coolant test is planned for 2023 with the reactor at 100% power.

In the United States, the US DOE is supporting several HTR-related projects. While the DOE's Advanced Gas Reactor Fuel Development and Qualification Program (Demkowicz and Hunn, 2020) has provided the technological foundation for uranium-oxycarbide TRISO fuel performance that will be leveraged by these HTR developers, variations of the TRISO fuel form are being explored to accommodate diverse reactor design requirements. Some key HTR demonstration and fuel development activities include:

- X-energy is designing a 200 MWt pebble-bed HTR (Xe-100) and a <10 MWt gas-cooled microreactor (Xe-Mobile).[1] The subsidiary company TRISO-X has started construction of a TRISO fuel fabrication facility in Oak Ridge, Tennessee (WNN, 2022a).
- Kairos Power is developing a 320 MWth fluoride salt-cooled, HTR design fuelled with spherical pebbles containing TRISO fuel and a small 35 MWth Hermes demonstration reactor (WWN, 2021). In support of these designs, the company has announced a partnership with Los Alamos National Laboratory to fabricate fuel in New Mexico (WNN, 2022b).
- BWX Technologies (BWXT) is currently developing an 80 MWth commercial microreactor design, BANR (BWXT, 2021), as well as a <10 MWth mobile nuclear power plant for the US DOE (BWXT, 2022a), both utilising TRISO fuel. The BANR fuel deviates from conventional TRISO fuel forms by utilising UN kernels in a SiC matrix. BWXT AT is expanding its TRISO fuel fabrication capabilities to support these projects as well as coated fuels for NASA for its space nuclear propulsion and has announced the start of fuel production (BWXT, 2022b).
- Ultra Safe Nuclear Corporation (USNC) is seeking to commercialize its 15-30 MWth micro modular reactor,[2] utilising "FCM" fuel that incorporates

1 https://x-energy.com.

2 www.usnc.com/mmr (accessed on 9 January 2023).

TRISO fuel particles in a SiC matrix. Its pilot fuel manufacturing facility in Oak Ridge, Tennessee was opened in August 2022 and will produce TRISO fuel for testing and qualification in support of micro modular reactor development (WWN, 2022c).

Technical highlights – Fuel and Fuel Cycle Project

The Fuel and Fuel Cycle Project Management Board (PMB) held its annual meeting in Cadarache in October 2022 and welcomed observers from Canada and the United Kingdom. The annual meeting was accompanied by a special two-day workshop on Material Properties of Tristructural Isotropic Fuels. The workshop (the sixth in an ongoing series organized by the Fuel and Fuel Cycle PMB) garnered significant participation. It included 24 technical presentations by attendees from 5 different countries and 9 different institutions. Topics included coating layer thermal and mechanical property measurement, SiC microstructure evaluation and irradiation behaviour, TRISO fuel modelling and simulation studies, and fission product transport behaviour in TRISO fuel.

The Fuel and Fuel Cycle project is intended to provide demonstrated solutions for the VHTR fuel (design, fabrication and qualification) and for its back-end management, including novel fuel cycle options. TRISO-coated particles, which are the basic fuel concept for the VHTR, need to be qualified for relevant service conditions. Furthermore, its standard design – a uranium dioxide (UO_2) kernel surrounded by successive layers of porous pyrocarbon, dense inner pyrocarbon, SiC and, finally, an outer pyrocarbon – could evolve along with the improvement of its performance through the use of a uranium oxycarbide kernel or a ZrC coating for enhanced burnup capability, minimized fission product release and increased resistance to core heat up accidents (>1 600°C).

Fuel characterization work, post-irradiation examinations (PIE), safety testing, fission product release evaluation, as well as the measurement of chemical and thermomechanical material properties in representative conditions will feed a fuel material database. Further development of physical models enables assessment of in-pile fuel behaviour under normal and off-normal conditions. The fuel cycle back-end encompasses spent fuel treatment and disposal as well as used graphite management. An optimized approach for dealing with graphite needs to be defined. Although a once-through cycle is envisioned initially, the potential for deep burn of plutonium and minor actinides in a VHTR, as well as the use of thorium-based fuels, will be accounted for as an evolution towards a closed cycle.

PIE of the US AGR-5/6/7 irradiation is currently in progress and includes non-destructive examination of the fuel compacts and irradiation capsule components as well as destructive examination of fuel compacts to assess the particle coating integrity and the extent of fission product release during the irradiation. A major focus of the early PIE was examination of Capsule 1 – which exhibited high fission gas release starting roughly halfway through the irradiation – to determine the cause of the apparent particle failures. The initial examination concluded that the particle failures were caused by nickel corrosion of the SiC layers, which was precipitated by degradation of the thermocouples in the capsule due to unexpectedly high temperatures during operation. The remaining four capsules did not experience operational anomalies.

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). Three HTR-10 spent fuel elements have undergone burnup evaluation. This included non-destructive gamma spectrometry as well as pebble deconsolidation, particle dissolution and isotopic analysis. The results indicated generally good agreement between measured cesium activity determined from the various methods. Pre-irradiation characterization of legacy HTR-10 fuel elements was also performed, concentrating on the effect of high-temperature annealing on the SiC layer microstructure. Results indicated some coarsening of the SiC grain size after heating at 1 800°C and 2 000°C for one hour.

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. The AGR-3/4 irradiation experiment in the United States was dedicated to investigating fission product transport. PIE and testing were extensive, involving numerous heating tests and destructive examination of approximately 20 fuel compacts, as well as a detailed examination of fission product distribution in the components of the 12 irradiation capsules. With the experimental work completed, current efforts involve analysing the data and comparing them to computational fission product transport models.

Another critical area of research with regard to TRISO fuel qualification is fuel behaviour in oxidizing environments, which can be experienced in an HTR core during certain accident scenarios. The United States continues to develop the capability to heat irradiated TRISO fuel specimens to temperatures as high as 1 600°C in atmospheres containing air or steam while measuring the release of gaseous and condensable fission products in real time. The system is expected to begin operation in 2023. At KAERI, studies have continued on the behaviour of fuel elements and graphite under air or steam ingress conditions. Data collected include mass loss and oxidation region thickness as a function of time and temperature.

Several PMB members are pursuing the development of advanced TRISO fuel. KAERI is performing fundamental research on advanced, composite ZrC-SiC coatings ("QUADRISO") for particles to improve fuel performance at higher burnup, higher outlet temperature and extended core life. Modelling and simulation studies have examined the theoretical impact of fission product corrosion on predicted particle failure rates using the ZrC-SiC double layer, the stresses that develop in the layers of the QUAD-

RISO particle, and the impact of kernel migration on particle performance over very long irradiation times. Experimental investigation has included simulated irradiation tests using Ar ion irradiation at room temperature and 700°C (Figure 3.25). The results demonstrated good layer interface integrity at up to 6.7 dpa and have provided data on the defect evolution in ZrC. Further development of the internal gelation process for kernel fabrication is also underway to support kernel compositions beyond UO_2. China is also pursuing sol-gel fabrication of uranium-oxycarbide kernels and ZrC coating layer deposition and subsequent heat treatment.

In China, fabrication of HTR-PM pebbles with 8.5% ^{235}U enrichment is in progress. The combined thermal output of the two HTR-PM modules reached 200 MWth power in December 2022 using the first core fuel pebbles with 4.2% ^{235}U enrichment. Work also continues on a new fuel manufacturing line with greater capacity.

Technical highlights – Hydrogen Production Project

In addition to the two main processes for splitting water for hydrogen production (the sulphur/iodine thermo-chemical and high-temperature steam electrolysis [HTSE] cycles), the recent Project focus has been on the hybrid copper-chlorine thermo-chemical and the hybrid sulphur cycles (Figure 3.26). R&D efforts in this PMB address material development, feasibility, optimization, efficiency and economic evaluation for industrial-scale hydrogen production. Performance and optimization of hydrogen production processes are being assessed through integrated test loops, from laboratory scale through pilot and demonstration scale, and include component development.

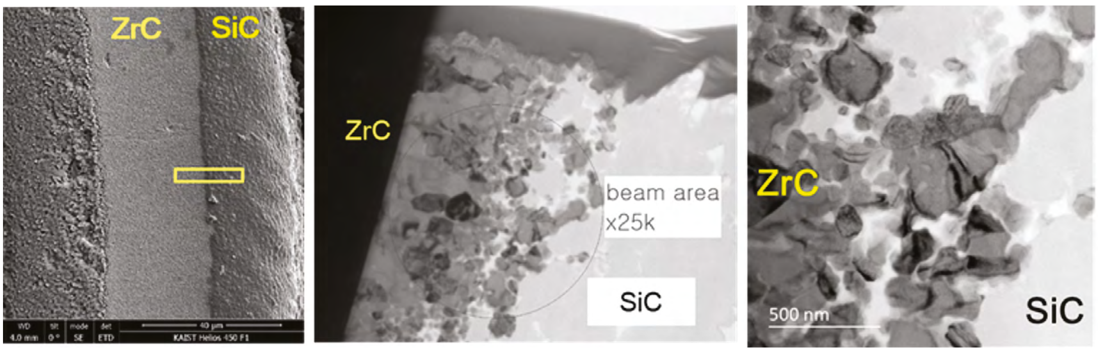
In 2022, research activities focused on HTSE and thermo-chemical processes, as well as other processes potentially coupled to renewables. Notably, countries involved in R&D activities noted a potential role for low-carbon hydrogen from nuclear energy to produce synthetic fuels or address demands in industrial processes. In Canada, a lab-scale demonstration of the copper-chlorine process was completed producing 50 L/h H_2. Research is continuing on HTSE powered by nuclear or renewables, improving the longevity of HTSE cells, and on issues related to co-electrolysis. Work is also advancing on synthetic fuels manufacturing using clean hydrogen and Syngas, which aligns with Canadian federal initiatives related to hydrogen production and SMRs.

In contrast, research in China has focused on the development of the sulphur/iodine and hybrid sulphur processes and chemical reactors. Specifically, prototype chemical reactors have been fabricated, including an H_2SO_4 and a HI decomposer producing an H_2 flow of 0.85 m^3/h. A series of experiments to test the chemical reactors is planned, utilising a high-temperature helium loop currently nearing completion. Research on SO_2 depolarised electrolysis (SDE) also continued in 2022 with kinetics reaction tests conducted on a 5x5 cm^2 SDE cell to prepare for scaling-up the SDE system (Figure 3.27).

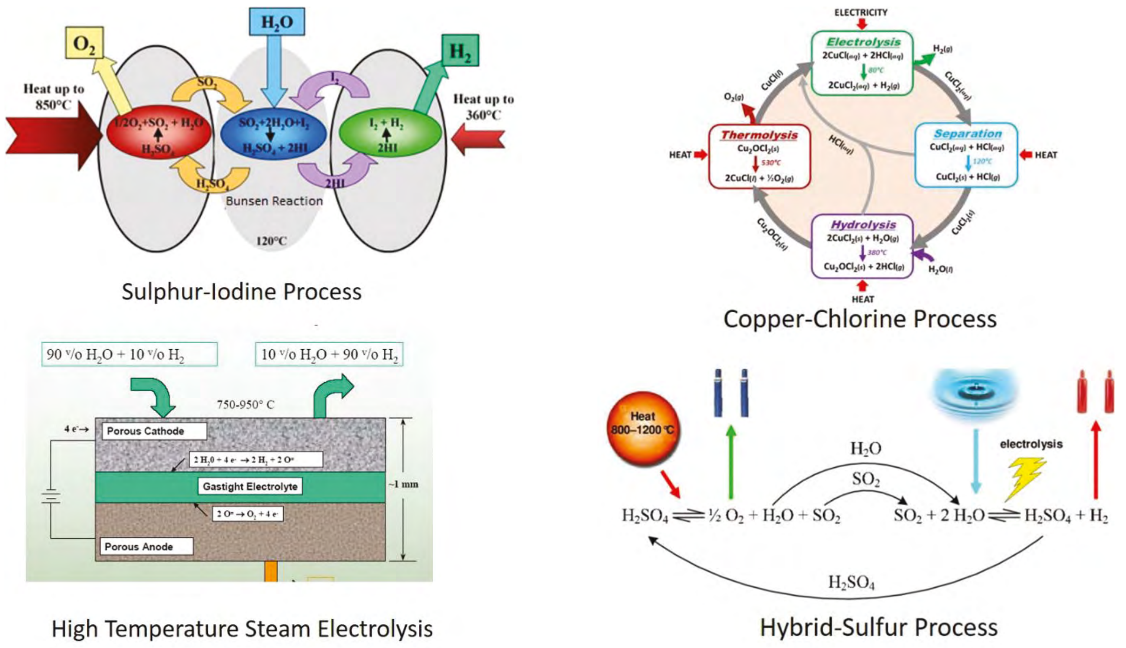
Research in the European Union has focused on storage and heat transfer media in solar thermochemical sulfur cycles, thermal energy from solar for HTSE, the use of iron as an energy carrier through electrochemical reduction, and an economic assessment of green hydrogen and solar-thermochemical jet fuel. This work aligns with broader initiatives in the European Union, such as the H_2 Mobility Plan, which aims to establish the necessary infrastructure to guarantee nationwide hydrogen-powered mobility in Germany and Europe.

France is advancing research in support of national industrial and hydrogen policies, such as the French National Hydrogen Strategy and the French Recovery Plan, which aims to provide significant support to hydrogen production at scale, and the coupling of hydrogen production with nuclear energy. Research in France included the development of a design

Figure 3.25. Micrographs showing the ZrC-SiC coating interface region after Ar irradiation up to 10^{18} ions/cm^2 at KAERI



Source: Zhang et al. (2022).



Source: Presented at the GIF Industry Forum by Sam Sappiah, Canadian Nuclear Laboratories, Canada.

Figure 3.26. Hydrogen production pathways from Generation IV nuclear technologies

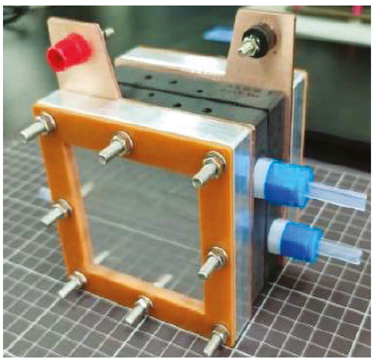
and process for an HTSE plant, experimental testing to improve HTSE cells and their durability, and additional activities to design and scale hydrogen production stacks. An assembly of 3 subsets of 25 larger cells (a 200 cm^2 active area) was successfully tested with similar results obtained compared to an earlier 100 cm^2 cell test. These results are encouraging for the future testing of a 75-cell stack.

With the operating hydrogen production test facility at the JAEA, research in Japan is focused on the thermochemical iodine-sulphur (IS) process. The facility is undergoing component performance testing to confirm the predicted process functionality, and long-term corrosion data are being produced by extended H_2 production tests. Technology development is underway to improve plant maintenance technologies and automate plant operation systems. The decomposition of hydrogen-iodine was theoretically and experimentally investigated to assess the reactor's suitability for thermochemical hydrogen production. The JAEA is working on coupling a carbon-free hydrogen production plant to the HTTR and is planning to complete the development of the coupling technologies by 2030.

Korean hydrogen production research efforts in 2022 focused on an HTSE solid-oxide electrolyser cell integral test, coupled with a helium gas loop and steam supply, towards kW-scale demonstrations over the next few years. KAERI is evaluating the performance of HTR-based hydrogen and electricity co-generation, and is working with the Research Institute of Industrial Science & Technology on a project to decarbonize the industrial sector by coupling nuclear power to hydrogen production and other industrial processes.

The United States is conducting research to advance technology development and commercialization of hydrogen produced from HTSE. This includes integrated systems testing utilising the Dynamic Energy Transport and Integration Lab, Human Systems Simulation Laboratory and Real-Time Digital Grid Simulation Connection experimental facilities. Manufacturing and testing of solid oxide electrolyser cells is ongoing, and multiple short stacks of different sizes have been assembled and tested. Through the US "Earthshot" initiative, commercialization of hydrogen production stacks utilising HTSE is accelerating, and a 100 kW solid oxide electrolyser cells stack module exceeded 2 000 hours of stable and transient testing. Multiple well-sealed cassettes with large 300 cm^2

Figure 3.27. 5x5 cm^2 membrane SO_2 depolarized electrolysis testing cell at the INET Lab



Source: Ma et al. (2022).

active area electrode-supported SOEC cells have been successfully produced, and performance and longevity will continue to be explored.

As part of its Advanced Modular Reactor (AMR) activities, the United Kingdom aims to achieve demonstration of HTR technology by the early 2030s. An objective of the AMR RD&D Programme is to demonstrate that AMRs can produce high-temperature heat which could be used for low-carbon hydrogen production. Special research activities related to these goals include work on the sulphur-iodine thermochemical process, HTSE and technologies to enable the coupling of nuclear energy to hydrogen production.

Technical highlights – Materials Project

Although the original VHTR Materials Project Plan term was completed in 2012, the Materials Project Arrangement continued through 2021 under its 2nd amendment of the project arrangement, which became effective on 27 April 2020. It incorporated a new project plan for technical activities and planned contributions from 2018 to 2022 and added Australia as an additional signatory. It also extended the term of the project arrangement through April 2030. In 2021, the United Kingdom provided a proposal for joining the VHTR Materials Project Arrangement and Canada provided its proposal to rejoin the project arrangement after its earlier withdrawal. The proposals containing their formal planned contributions were accepted by the Materials Project Management Board (PMB) and an updated project plan was prepared for 2018-24. Contributions to the updated project plan were provided by all existing signatories (Australia, China, the European Union, France, Japan, Korea, Switzerland and the United States), as well as by Canada and the United Kingdom. The updated project plan was unanamously approved by the VHTR Materials PMB and the VHTR System Steering Committee in September and October 2021, respectively. The NEA prepared a 3rd amendment to the project arrangement, based on the updated project plan, which was distributed for formal approval early in 2022.

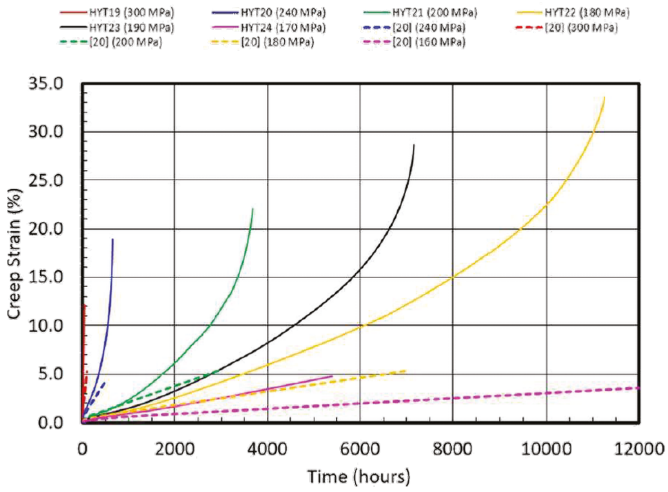
Materials development and qualification, design codes and standards, as well as manufacturing methodologies are essential for VHTR system development. The primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in operating environments. For core coolant outlet temperatures up to 950°C, it is envisioned to use existing materials; however, safe operation under off-normal conditions possibly involving corrosive process fluids will require the development and qualification of new materials to reach the VHTR goal of 1 000°C. Multi-scale modelling is needed to support improved design methods. In addition to high-temperature heat exchangers, additional attention is being paid to metal performance in steam generators, which reflects the current interest in steam-based process applications in the 750-850°C range. Structural materials are considered in three categories: 1) fuel matrix and core structures graphite; 2) very/medium-high-temperature metals; and 3) ceramics and composites. A materials database (refered to as the Gen IV *Materials Handbook*) has been developed and is being used to efficiently store and manage materials data, facilitate international R&D coordination, and support modelling to predict damage and lifetime assessment. As a recent development pursued by several signatories, emerging advanced manufacturing techniques such as additive manufacturing, laser fusion, consolidation of metal powders, direct deposition, etc., will create new material classes, with increased design potential and possibly new solutions to the challenges described above. New approaches to synthesize novel high-temperature structural materials were also explored. The qualification of these advanced manufacturing methods and the resulting materials have been added as a new task in the 3rd Project Arrangement amendment.

In 2022, research activities focused 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. By the end of 2022, more than 498 technical reports and over 30 000 materials test records, including contributions from all signatories, had been uploaded into the *Materials Handbook*. This reflects members' outstanding technical output that has now been shared to support system design and codes and standards development.

Several members performed additional characterization and analysis of selected graphite baseline data and its inherent scatter. Mechanical, physical and fracture properties behaviour were examined for numerous candidate 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 assessed, and examination and validation of graphite multi-axial loading responses from dimensional changes and seismic events using large-scale experiments on graphite blocks continued in 2022. Data to support graphite model development were generated in the areas of microstructural evolution, irradiation damage mechanisms and creep. Support was provided for the development of both American Society for Testing and Materials and American Society of Mechanical Engineers codes and standards required for the 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 Alloys 800H and 617 database and its extension through aging, creep, creep fatigue and creep crack growth rate testing to 950°C. Welding studies on 617, 800H and dissimilar welds of T22 to 800H were performed as well as qualification testing of new metallic materials (e.g. alloy 709, high entropy alloys and oxide dispersion-strengthed alloys) for construction of high-temperature nuclear components. Enhanced diffusion-bonding techniques for construction of compact heat exchangers showed very promising results, and extensive modelling and testing of compact heat exchangers are laying the groundwork for their qualification in VHTRs. Figure 3.28 shows an example of the 2022 UK contribution (Jacobs) for the creep strain deformation behaviour of PM/HIP 316L austenitic stainless steel at 600°C in air with various applied stresses (solid lines). The dotted lines show the predicted values for conventional forged 316L steel under similar temperature and applied stresses. This comparison essentially suggests that the PM/HIP material has slightly reduced creep performance compared with conventional forged material.

In the near/medium term, metallic alloys are considered as the main option for control rods and internals in HTR projects, which target outlet temperatures below 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 the capabilities of metallic materials. This is especially true for control rods, reactor internals, thermal insulation materials and fuel cladding. In 2022, work continued to examine the thermomechanical properties of SiC and SiC-SiC composites, including irradiation -creep effects and oxidation in carbon-carbon composites. Studies to evaluate radiation damage and fracture behaviour of composites began, as did 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 examining irradiation effects on ceramic composites for these types of applications. Figure 3.29 shows a recent study conducted at ANSTO, using a finite-elment modelling approach to gain a more comprehensive understanding of the elasto-plastic and fracture behaviour of C/C composites. A large number of SEM microstructure images were used to create a representative volume element, which was subsequently used to simiulate the behaviour of C/C composites. The model effectivly captures the elasto -plastic behaviour and fracture of the matrix, while the de-cohesion of the carbon fibers is accounted for using cohesive elements.



Note: The solid lines show creep strain deformation of PM/HIP 316L austenitic stainless steel at 600°C in air compared to the conventionally forged material.

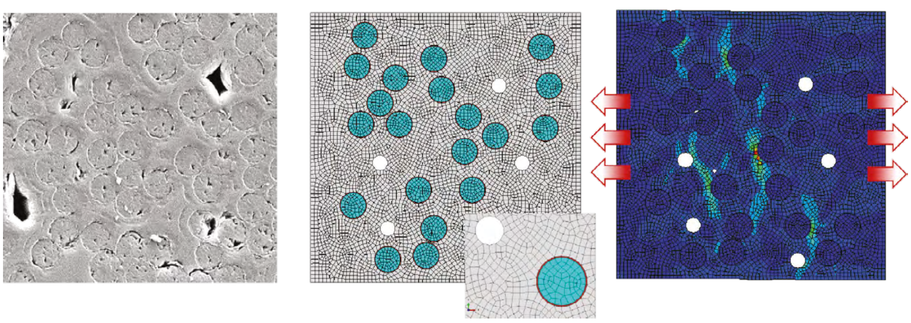
Source: Data from Jacobs (United Kingdom). New results (not yet published) presented at the VHTR MAT PMB meeting in September 2022.

Figure 3.28. Creep strain deformation of PM/HIP 316L austenitic stainless steel at 600°C in air compared to the conventionally forged material

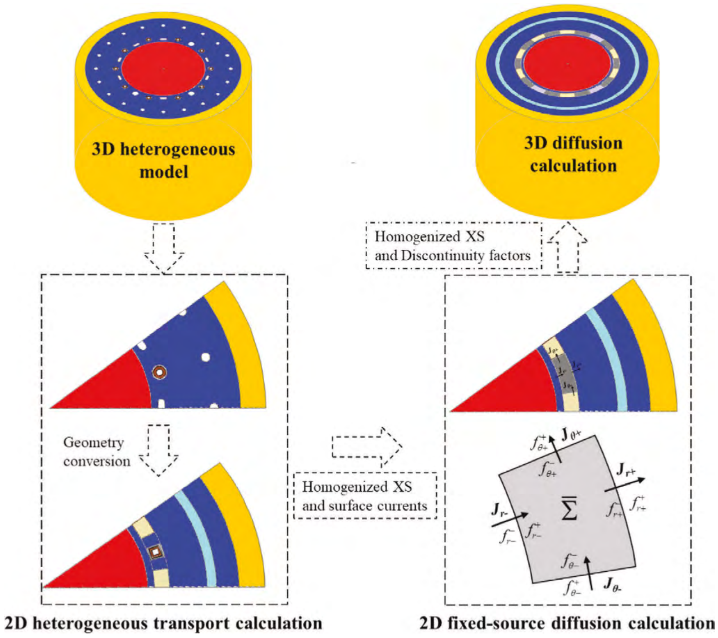
Technical highlights – Computational Methods Validation and Benchmarks Project

Validation of new computational methods and codes in the areas of HTR 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 the current HTTR, High Temperature Reactor (HTR)-10 and High Temperature Gas-Cooled Reactor – Pebble-bed Module (HTR-PM) test data or historic German and US operational data (AVR, 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.

Figure 3.29. ABAQUS FEM model of a C/C composite (center and right), based on SEM microstructure detail (left)



Note: Study performed at ANSTO, Australia. New results (not yet published) presented at the VHTR MAT PMB meeting in September 2022.



Source: She et al. (2022).

Figure 3.30. Treatment of strong absorbers with PANGU code

At the Chinese INET, a state-of-the-art 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 activities continued in 2022. The 3D calculation capability of the PANGU code was enhanced, especially in the treatment of strong absorbers, and the code was successfully applied to prediction calculations for the HTR-PM criticality experiments, as well as the simulation of subsequent run-in-phase operations. In March 2022, the PANGU and NUIT codes were submitted to the National Nuclear Safety Administration for licensing review (Figure 3.30). The new codes are used as design verification tools in HTR-PM600 and HTR-PM1000 projects and are expected to replace the legacy codes after licensing from the National Nuclear Safety Administration.

In Japan, the JAEA carried out a planned HTTR safety demonstration test on 31 January 2022 under the framework of an NEA project. There are also plans to perform various tests concerning safety, core physics, thermal-fluid characteristics, fuel performance, etc. including heat application test coupling HTR and hydrogen production. The JAEA's R&D in code and calculation methodology developments are expected to contribute to computational methods, validation and benchmarking activities, such as a benchmark activity using the US Advanced Test Reactor TRISO irradiation data. As part of this benchmark activity, the JAEA has been developing a model of the Advanced Test Reactor using the JAEA-developed Monte Carlo simulation code MVP (Nagaya et al., 2017).

The VHTR R&D programme in Korea aims at improving design code development and assessment of very high-temperature system key technologies. A sub-project focuses on the development of coupled analysis technologies between the very high-temperature system and the HTSE hydrogen production system. Activities in 2022 included modification of the GAMMA+ code to support a water-cooled reactor cavity cooling system, and a coupled calculation (CAPP/GAMMA+) to simulate transients such as control rod ejection, coolant temperature drops and flow rate increases in the secondary loop. GAMMA+ has also been updated to perform uncertainty assessments and extended to non-LWR applications (SFR, MSR). A tritium diffusion model from the TRISO kernel via the compact and moderator graphite was also developed to accurately predict tritium release into the coolant. In addition, TRISO particle failure models were investigated for normal operation and core heat-up and air-ingress transient conditions.

In the European Union, most of the current/recent (V) HTR-related activities are taking place in the Horizon Europe Framework Program project "GEMINI 4.0" ("GEMINI For Zero Emission"), a follow-up project of GEMINI+, which started June 2022 and will run until May 2025. The main focus areas are optimization of safety and competitiveness, decarbonization of industry, fuel and fuel cycle (including back end and alternative fuels), licensing readiness (including validation of the GEMINI+ safety options), and socio-economic impact. Some additional national HTR-related projects will also deliver contributions to the Computational Methods Validation and Benchmark (CMVB) Project: the Polish projects GOSPOSTRATEG – HTR, which was completed in 2022, the GOSPOSTRATEG-HTR follow-up NCBJ-HTGR (2021-24) and the NOMATEN Centre of Excellence (2021-26). The NCBJ-HTGR focuses consecutively on pre-conceptual design, conceptual design, basic design and detailed design of a 30 MWth research and demonstration HTR. The conceptual design phase was completed in September 2022.

The US DOE funds several activities related to the production of experimental validation data and software for the analysis of HTRs. The Natural Convection Shutdown Heat Removal Test Facility experiment continued to produce valuable data on off-normal conditions for a water-based reactor cavity cooling system (e.g. boil-off until inventory depletion), while the High Temperature Test Facility integral test datasets produced up to 2021 are currently being reviewed to select the most representative data for the validation of system and CFD codes. Several DOE awards to US universities were also assessed for their potential to provide future code validation benchmark data (Qin et al., 2022) that could be shared within the CMVB Project.

US HTR core modelling and simulation activities in 2022 included the successful validation of the Serpent Monte Carlo models used to generate HTTR cross-section libraries (Laboure et al., 2022), which will lead to improved Griffin code results. At Idaho National Laboratory, a novel high-resolution Monte Carlo-based run-in scenario for a generic pebble bed HTR was obtained, and reduced-order models were trained using data from the Griffin/Pronghorn equilibrium core simulations to perform a sensitivity analysis on the effect of input parameters (e.g. pebbles' recirculation speed, discharge burnup) on power peaking factors (Prince et al., 2022). A SAM model of a pebble bed reactor was also developed at Argonne National Laboratory using local point-kinetic parameters generated by Griffin/Pronghorn to provide a fast-running model for pebble bed reactors' core analysis.

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Chair of the VHTR SSC,
with contributions from
VHTR members

References

BWXT (2021), "Terrestrial Micro RX", web page, www.bwxt.com/what-we-do/advanced-technologies/terrestrial-micro-rx.

BWXT (2022a), "BWXT to build first advanced microreactor in United States", www.bwxt.com/news/2022/06/09/BWXT-to-Build-First-Advanced-Microreactor-in-United-States (accessed on 9 January 2023).

BWXT (2022b), "BWXT starts production of TRISO fuel for first U.S. Generation IV microreactor", www.bwxt.com/news/2022/12/07/BWXT-Starts-Production-of-TRISO-Fuel-for-First-US-Generation-IV-Microreactor (accessed on 9 January 2023).

Demkowicz, P.A. and J.D. Hunn (2020), "Two-decade DOE investment lays foundation for TRISO-fueled reactors in the US," *Nuclear News*, Vol. 63.

Kim, E.S. (2022), "Status and progress of TRISO fuel R&D in KAERI", presentation at the 18th official meeting of the VHTR Fuel and Fuel Cycle Project Management Board, Cadarache, France, 18 October 2022.

Laboure, V.M. et al. (2022), *FY22 Status Report on the ART-GCR CMVB and CNWG International Collaborations*, INL/RPT-22-68891 Rev 0, Idaho National Laboratory, http://dx.doi.org/10.2172/1893099.

Ma, X., et al. (2022), "Application of Pt loaded graphite felt in SO2-depolarized electrolyzer", *International Journal of Hydrogen Energy*, Vol. 47/74, pp. 31 575-31 586, https://doi.org/10.1016/j.ijhydene.2022.07.086.

Nagaya, Y. et al. (2017), *MVP/GMVP Version 3: General Purpose Monte Carlo Codes for Neutron and Photon Transport Calculations Based on Continuous Energy and Multigroup Methods*, JAEA-Data/Code 2016-018, Japan Atomic Energy Agency, https://jopss.jaea.go.jp/pdfdata/JAEA-Data-Code-2016-018.pdf.

Prince, Z.M. et al. (2022), *Reduced Order Models Generation for HTGRs Pebble Shuffling Procedure Optimization Studies*, INL/RPT-22-68865 Rev 0, Idaho National Laboratory, https://inldigitallibrary.inl.gov/sites/sti/sti/Sort_63229.pdf.

Qin, S. et al. (2022), *High-temperature Gas-cooled Reactor Research Survey and Overview: Preliminary Data Platform Construction for the Nuclear Energy University Program*, INL/RPT-22-68771 Rev 0, Idaho National Laboratory, http://dx.doi.org/10.2172/1887092.

She, D. et al. (2022), "An improved homogenization method for the treatment of strong absorbers in pebble-bed HTGR", *Annals of Nuclear Energy*, Vol. 170, 108983, https://doi.org/10.1016/j.anucene.2022.108983.

WNN (2022a), "X-energy marks start of construction at US fuel plant", *World Nuclear News*, www.world-nuclear-news.org/Articles/Groundbreaking-marks-start-of-construction-at-US-f (accessed on 9 January 2023).

WNN (2022b), "Kairos Power, Los Alamos collaborate to make TRISO fuel", *World Nuclear News*, www.world-nuclear-news.org/Articles/Kairos-Power,-Los-Alamos-collaborate-to-make-TRISO (accessed on 9 January 2023).

WNN (2022c), "Pilot TRISO fuel manufacturing plant opens in Tennessee", *World Nuclear News*, www.world-nuclear-news.org/Articles/Pilot-TRISO-fuel-manufacturing-plant-opens-in-Tenn (accessed on 9 January 2023).

WNN (2021), "Kairos Power plans Hermes demonstration reactor at Oak Ridge", *World Nuclear News*, www.world-nuclear-news.org/Articles/Kairos-Power-plans-Hermes-demonstration-reactor-at (accessed on 9 January 2023).

Zhang, Z.-Y. et al. (2022), "600-MWe high-temperature gas-cooled reactor nuclear power plant HTR-PM600", *Nuclear Science and Technology*, Vol. 33/101, https://doi.org/10.1007/s41365-022-01089-9.

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. a lower levelized unit cost of energy);
- to have a level of financial risk comparable to other energy projects (i.e. 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 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 that demonstrate the EMWG methodology for the economic assessments of Gen III and Gen IV systems, as well for co-generation 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 and the Hydrogen Economic Evaluation Programme. The results were published in peer-reviewed publications. Lessons learnt from the benchmarking exercise and users' feedback has informed the refinement of the G4ECONS tool. The latest version, G4ECONS v3.0, was released in October 2018 with an improved user interface. In 2022, the EMWG launched a survey to collect user feedback on G4ECONS v3.0 and identify potential model improvements. The unique aspects of small modular reactors (SMRs), non-electric applications and embedded sensitivity analysis were identified as areas of the greatest interest for future developments. In addition, many users expressed an interest in additional training sessions on the G4ECONS tool.

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 EMWG has worked collaboratively with the GIF Senior Industry Advisory Panel (SIAP) to investigate challenges and opportunities for the deployment of Gen IV systems in electricity markets with a significant penetration of renewable energy resources and to produce a position paper for the GIF Policy Group.

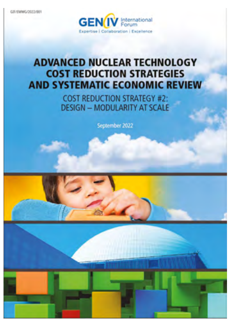
An abridged version of the EMWG position paper on the impact of increasing shares of renewables on the deployment prospects of Gen IV systems was presented at the 4th GIF Symposium (2018) and an executive summary was posted on the GIF website.

In 2018, the terms of reference for the EMWG were amended to incorporate the expanded mandate so as to inform the GIF Policy Group and Experts Group on the policies and R&D needs for the future deployment of Gen IV systems.

In 2021, the EMWG worked with the finance industry to produce a report, *Nuclear Energy: An ESG Investable Asset Class*. The report considered the nuclear industry's ability to report against a broad range of environmental, social and governance data collection and accounting metrics (ESG). The report was required to help the finance industry to start to consider nuclear as an investable asset class. Each project can then be considered on its own merits. For a nuclear project to access finance (and particularly climate finance), the project itself will need to be able to report well against a broad range of ESG. Enabling nuclear to be considered as an ESG-investable asset class is the first step to enabling the flow of financing. The report will be maintained and updated regularly to enable it to be used by the finance industry as a reference guide for nuclear as an asset class.

Also in 2021, the *Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review* (ANTSER) process was developed to provide a methodological framework for evaluating nuclear cost-reduction strategies. This initial report provided an example strategy on "functional confinement". In 2022, a second ANTSER cost-reduction strategy was developed for "modularity at scale".

EMWG activities in 2022



In 2022, the EMWG issued a major report: *Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review, Cost Reduction Strategy #2: Design – Modularity at Scale*.

The study adds to the ANTSER framework through an investigation of modularity for nuclear energy applications at different scales. Modularity can include the incorporation of all major safety-significant systems – within one module, standardized modules and factory fabricated modules – the capacity to add modules to increase power output and the consolidation of components resulting in less on-site construction. Modularity is a topic of interest in the commercial nuclear energy sector with the emergence of SMRs. Modularity applications for small- to medium- and micro-scale plants are less well-known given their limited deployment.

The report focuses on the use of modularity approaches to reduce the cost of Gen IV nuclear technologies. The study surveys the literature on modularity and describes the different ways that modularity has been used or considered for nuclear plants. Lessons learnt from previous uses of modularity in the nuclear industry are used to inform readers on the challenges and opportunities involved in extending uses to advanced nuclear technologies. Modularity approaches are surveyed for the highest potential to reduce costs for large-, small-, medium- and micro-scale nuclear reactors. Successful modularization approaches from other industries are considered from the perspective of their potential transferability to the nuclear sector. The functional containment design approach is explored as a means for designers to rethink how modularity is used in advanced nuclear technologies. Modular approaches, from large to very small scales, and balance-of-plant options are described in terms of their cost reduction potential; their technical readiness; and their research, development and demonstration needs.

In October 2022, the EMWG participated in the GIF Industry Forum, leading the EMWG Session: "Economic Challenges and Opportunities for Gen IV Reactors". The technical session was an engaging and dynamic discussion looking at the economic challenges and discussing how the impact of how projects are established can considerably influence the economic and financing of projects.

In 2023, the EMWG will continue the development of ANTSER cost-reduction strategies and extend work on nuclear financing. It will also embark on a multiannual effort to update and refresh the 2007 *Cost Estimating Guidelines* report and G4ECONS v3.0 model. In 2023, the EMWG plans to engage with the IAEA, SIAP and other GIF working groups/task forces on suggested updates, particularly with a view toward evaluating small reactors. The EMWG will also collaborate with key industry players who are developing SMRs to inform the cost-estimating guidelines. In 2025, once the updated guidelines and G4ECONS v4.0 are released, the EMWG will work with the Education and Training Working Group (ETWG) to launch an improved training series for both existing and new users of these EMWG tools.

David Shropshire
Co-Chair of the EMWG

Megan Moore
Co-Chair of the EMWG

Fiona Reilly
Co-Chair of the EMWG

References

GIF (2022), *Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review, Cost Reduction Strategy #2: Design – Modularity at Scale*, Generation IV International Forum, Paris, www.gen-4.org/gif/jcms/c_204177/antser-2-final.

GIF (2007), *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems*, Generation IV International Forum, Paris, www.gen-4.org/gif/upload/docs/application/pdf/2013-09/emwg_guidelines.pdf.

Education and Training Working Group

The Generation IV International Forum (GIF)-ETWG serves as a platform to enhance open education and training as well as communication and networking of people and organizations in support of the GIF. Seventy-two webinars have been recorded and archived on the GIF portal since September 2016 with 13 focusing on operational experiences[1]. A panel session entitled the "Role of Nuclear Energy in Reducing CO_2 Emissions" featuring representatives from the GIF, the IAEA and the NEA was organized on 19 April 2022. Panellists offered their perspectives on the increasing recognition of nuclear energy not only as climate-friendly low-carbon energy option, but also as an enabler of a broader, more resilient energy source. The first-place winner of the 2021 Pitch your Gen IV Research competition, Dr. Flore Villaret from EDF, France received her prize with a fully sponsored participation in the GIF Industry Forum in Canada in October 2022. The second and third place winners of this competition presented a GIF webinar in March and May 2022.

Participation at the GIF Industry Forum, October 2022, Toronto, Canada

The ETWG organized two panel sessions during this event.

The objectives of the first panel session, "Innovators' Panel", were for the panelists to share experiences on advancing concepts beyond the academic phase, to reflect on their experience with Gen IV/SMR systems, share their career paths, discuss the challenges they faced in moving a concept forward and highlight the rewards of working in the advanced nuclear field. Five speakers made a short presentation, providing experience-based insights for the early career attendees on how innovation can lead to breakthrough opportunities in the nuclear field. This was followed by a panel discussion moderated by John Kelly.

The second panel session "International Knowledge Management and Preservation on SFR" assembled an international Sodium Fast Reactor (SFR) Panel which discussed the lessons learnt on the design, construction and operation of SFRs (Phénix/Superphénix, Monju/Joyo and Fast Flux Test Facility) and how the transfer of knowledge was passed on in an international context to companies planning on building SFRs ranging in power from 300 MWe to 1 200 MWe. The international panel consisted of three senior SFR experts from France, Japan and the United States who have worked on past SFRs or current SFR projects as well as two engineers from Canada and the United States involved in designing/constructing the Natrium reactor from Terrapower and the ARC-100 SMR from ARC Clean Energy. The three senior panellists presented their experience, how their respective country has preserved the knowledge from these SFRs and how it is transferred to the next generation. The two industry panellists explained the project they are working on; they also provided their vision for SFRs over the coming decades and the challenges faced by their respective company, such as people and infrastructure. This was followed by a panel discussion moderated by Patricia Paviet.

1 www.gen-4.org/gif/jcms/c_84279/webinars.

Table 4.1. GIF webinars in 2022

Presenter	GIF webinars from January 2022 to December 2022
Joel Guidez, CEA, France	ESFR SMART: A European Sodium Fast Reactor Concept Including the European Feedback Experience and the New Safety Commitments Following Fukushima Accident
Nawal Prinja, Jacobs, United Kingdom	Artificial Intelligence in Support of NE Sector
Benjamin Jourdy, CEA, France	Scale Effects Analysis on the Thermal Hydraulic Behavior of Impinging Jets in Sodium Fast Reactors
Dianne Cameron, NEA; Wei Huang, IAEA; Shannon Bragg-Sitton, Idaho National Laboratory, United States	Role of Nuclear Energy in Reducing CO_2 Emissions
Jiho Shin, KAIST, Korea	Development of Nanosized Carbide Dispersed Advanced Radiation Resistant Austenitic Stainless Steel (ARES) for Generation IV Systems
John Vienna and Brian Riley, PNNL, United States	Nuclear Waste Management Strategy for Molten Salt Reactor Systems
Junghyun Bae, Purdue University, United States	A Gas Cherenkov Muon Spectrometer for Nuclear Security Applications
Danrong Song, Nuclear Power Institute of China, China	China's Multi-purpose SMR-ACP100 Design and Project Progress
Shijeru Takawa, JAEA, Japan	Development of In-Service Inspection Rules for Sodium-Cooled Fast Reactors Using the System Based Code Concept
Jewhan Lee, KAERI, Korea	Sodium Integral Effect Test Loop for Safety Simulation and Assessment (STELLA)
Mark Deinert, Colorado School of Mines, United States	Geospatial Analytics for Energy and Resilience Analysis
Derek Kultgen, Argonne National Laboratory, United States	The Mechanisms Engineering Test Loop (METL) Facility at Argonne National Laboratory

Notes: CEA: French Alternative Energies and Atomic Energy Commission; NEA: Nuclear Energy Agency; IAEA: International Atomic Energy Agency; KAIST: Korea Advanced Institute of Science & Technology; PNNL: Pacific Northwest National Laboratory; JAEA: Japan Atomic Energy Agency; KAERI: Korea Atomic Energy Research Institute.

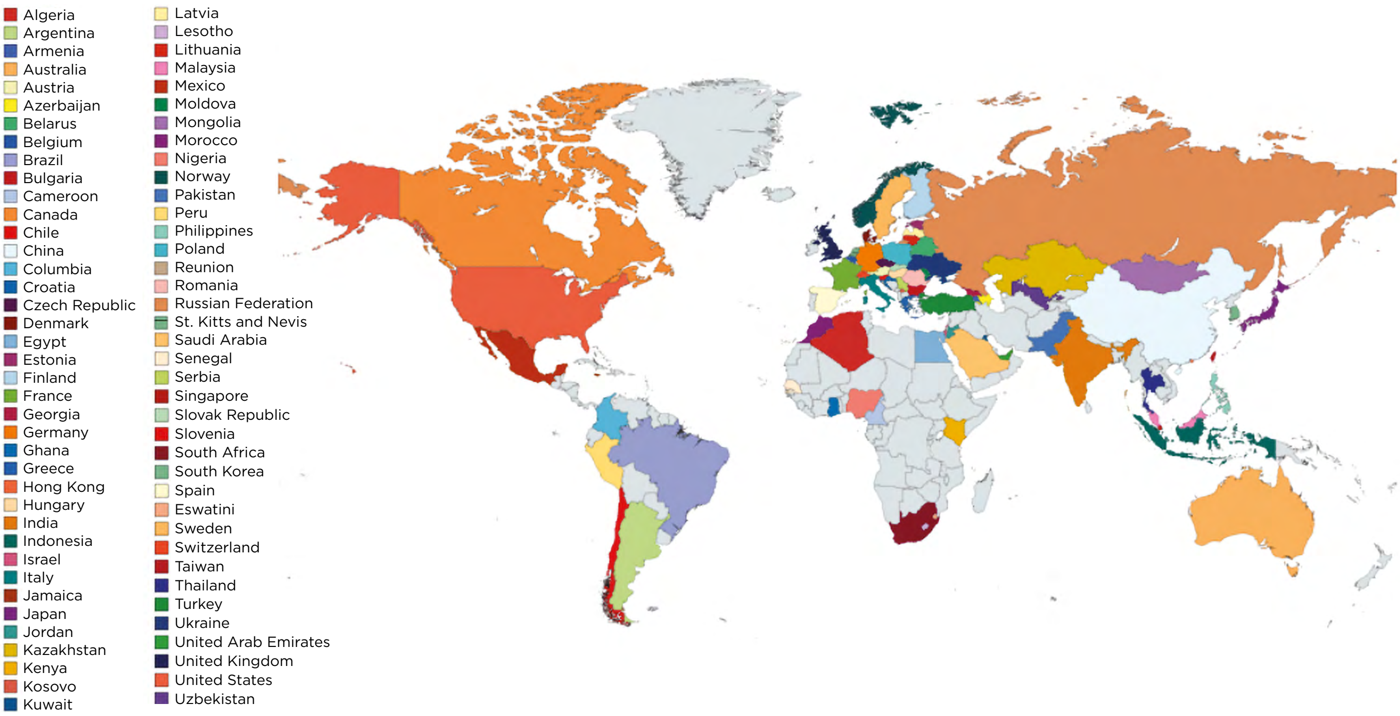


Figure 4.1. International participation in the attendance of the GIF webinar series with 78 countries

Main achievements

Twelve webinars, depicted in Table 4.1, were presented live in 2022 and are archived on the GIF portal.

As of December 2022, attendance during the live webcasts totalled 6 594. The number of viewings of recorded webinars in the online archive was 7 123. Total webinar viewing was 13 717 in 6 years. The webinars have reached scientists and engineers across 78 countries (Figure 4.1).

Members of the GIF ETWG participated in several summer schools; Patricia Paviet gave two lectures on the "Overview of the Nuclear Fuel Cycle" and the "Current and Prospects of Recycling" at the Modelling, Experimentation, Validation (MeV) summer school organized by the Oak Ridge National Laboratory in July 2022. Students appreciated the content of each class, and a very dynamic Q&A session took place after each class. Christian Latge and Konstantin Mikityuk participated in the ESFR SMART summer school on 11-12 July 2022 in Riga, Latvia, where lectures focused on thermos pumps, start-up and operation of a dedicated sodium loop.

The GIF ETWG launched a virtual "Pitch your Gen IV Research" competition in February 2021, and the three winners were invited to present a GIF webinar in December 2021, March 2022 and May 2022. The first winner of this competition participated in the GIF Industry Forum on 3-6 October 2022 in Toronto, Canada. The 2023 Pitch your Gen IV Competition was launched on 1 December 2022. All participants are initially required to submit a one-page executive summary of their research to be reviewed by a panel of experts. Up to 30 outstanding research projects that are original, significant and relevant for the GIF will be selected and the author will be invited to record a video pitch in first quarter of 2023. The recorded videos will be publicly posted (e.g. YouTube) for 30 days to allow world-wide viewing and voting. The public viewing of the pitches will begin in April 2023. A GIF expert panel (GIF Jury) will also judge the video pitches based on their creativity, communication effectiveness and technical quality. The best pitches will be selected based on: the GIF Jury; and the popular voting. The winners will be selected in both the popular vote category and the GIF Jury category. Winners will be announced in May 2023.

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

Proliferation Resistance and Physical Protection Working Group

The Proliferation Resistance and Physical Protection Working Group (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 proliferation resistance (PR) and physical protection (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." [1]

The methodology provides designers and policy makers with a technology-neutral framework and a formal comprehensive approach to evaluate, through measures and metrics, the PR and PP characteristics of advanced nuclear systems. As such, the application of the evaluation methodology offers opportunities to improve the PR&PP robustness of system concepts throughout their development and deployment cycle. The working group released the current version (revision 6) of the methodology for general distribution in 2011 (GIF, 2011a) and translations in Japanese and Korean of the methodology report have been produced for national use.

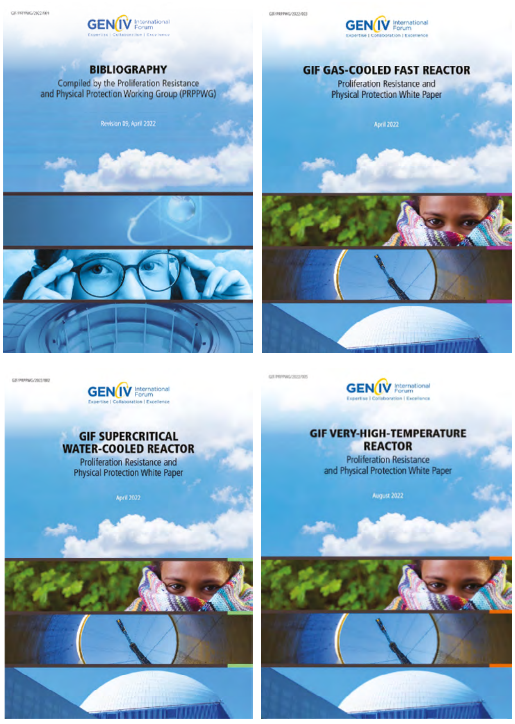
Since 2018, the main focus of the PRPPWG has been on updating the white papers on proliferation resistance and physical protection robustness of the six GIF design concepts. The past decade has seen several new advanced reactor vendors receiving funding from private and public investment – these white papers provide recommendations to improve the safeguards and security of advanced reactor designs. This is a joint effort with the system steering committees (SSCs) and provisional pSSCs of the six Gen IV technologies. The first versions of these white papers were produced between 2008 and 2011 (GIF, 2011b). The papers have been 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 enhanced intrinsic PR&PP features.

Individual white papers, after endorsement by both the PRPPWG and the responsible SSC/pSSC, are transmitted to the Experts Group for approval and published as GIF documents. 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 in the white papers such as cybersecurity. In addition to the LFR and SFR white papers published in 2021, as of December 2022, three additional white papers (GFR, SCWR, VHTR) were finalized and publicly available for download on the GIF website.[2] The following white papers were still in the process of being updated as of the end of 2022:

- MSR – paper incorporated feedback from the pSSC and the PRPPWG in a new draft. The MSR pSSC is currently reviewing the latest draft.
- Cross-cutting topics – the document is finalized and in the publication pipeline.

The PRPPWG's work was presented at the G4SR-4 and GIF Industry Forum in Toronto, Ontario, 3-7 October 2022. It received several positive comments and suggestions on the value of the work and areas where PR&PP analyses may be useful in the near future. These ideas include PR&PP aspects of transportable and floating reactors; spending more time on siting issues (remote versus near population centres); and current challenges reactor vendors face, including difficulties in working with multiple regulatory bodies, security costs and the need for people trained in these areas. The PRPPWG will be using this feedback to inform future efforts.

The update of the six GIF reactor technologies white papers and the related cross-cutting topics were presented by the PRPPWG at the 2022 IAEA Symposium on International Safeguards: Reflecting on the Past and Anticipating the Future (Cipiti et al., 2022).



1 www.gen-4.org/gif/jcms/c_9502/generation-iv-goals.

2 The GFR white paper is available at: www.gen-4.org/gif/jcms/c_200149/gfr-prpp-white-paper-2022-final-full-cover-page; the LFR white paper is available at: www.gen-4.org/gif/jcms/c_196730/lfr-prpp-white-paper-2021-final-22102021-clean2; the SCWR white paper is available at: www.gen-4.org/gif/jcms/c_200150/scwr-prpp-white-paper-2022-final-full-cover-page; the SFR white paper is available at: www.gen-4.org/gif/jcms/c_196731/sfr-prpp-white-paper-2021-final-18102021v8; and the VHTR white paper is available at: www.gen-4.org/gif/jcms/c_205621/2022-vhtr-prpp-white-paper.

The group maintains an annually updated bibliography of official publications, of publications referring to the PR&PP methodology, and of relevant issues. The latest edition, revision 9, was published in April 2022. It is available on the GIF website.[3]

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. Several PRPPWG members support the INPRO efforts, in particular updating the INPRO *Manual on Proliferation Resistance*. An IAEA representative participates regularly in the PRPPWG activities. The PRPPWG made a presentation at the 15th GIF-IAEA Interface meeting on 12-13 July 2022, highlighting collaboration on the INPRO PR methodology that the IAEA is updating, as well as on emerging safeguards issues related to the deployment of SMRs and micro reactors and the interfaces between safety, security and safeguards (3S).

Collaboration with the Risk and Safety Working Group (RSWG) was strengthened through exchanges at each group's meetings. PRPPWG representatives attended the 2022 RSWG semi-annual meetings, and RSWG representatives have been invited to attend the 33rd PRPPWG annual meeting.

After a series of preparatory activities, the PRPPWG, the RSWG and the VHTR SSCs formed an informal subgroup to investigate the interfaces between safety, security and safeguards in Gen IV reactors adopting a bottom-up approach. The group will investigate the interfaces between the 3S of a representative pebble-bed VHTR small modular reactor design, with the objective to comprehend potential technology-neutral guidelines for identifying 3S interfaces in Gen IV reactors. The main beneficiary of these guidelines will be designers aiming at incorporating the requirements of the 3S regimes into their systems during the very early design stages, supporting a potential 3S-by-design reactor development. The subgroup held its kick-off meeting in September 2022; the activity is scheduled to last approximately two years.

The PRPPWG is also engaged with the EMWG in exploring areas of potential collaboration. One area of mutual interest is the addition of safeguards and security costs to economic analysis of Gen IV reactor systems.

The PRPPWG holds monthly teleconferences to report on the progress of group and member activities and the summary records are filed in the PRPPWG archive for documentation and retrieval. The group will hold its 33rd annual meeting in person in Vienna on 25-27 January 2023. providing an opportunity to exchange information with the IAEA on topics of mutual interest and potential areas for strengthening the collaboration and synergies between the PRPPWG's and the IAEA's activities.

Lap-Yan Cheng
Co-Chair of the PRPPWG

Guido Renda
Co-Chair of the PRPPWG

Ben Cipiti
Co-Chair of the PRPPWG

References

Cipiti, B.B. et al. (2022), "Proliferation resistance and physical protection white papers of the six GIF Generation IV nuclear energy systems and cross-cutting topics: 2021-2022 Update", in: *IAEA Symposium on International Safeguards: Reflecting on the Past and Anticipating the Future*, 31 October-4 November 2022, International Atomic Energy Agency, Vienna, https://media.superevent.com/documents/20221030/ad0584a9cfd355f5a01a90308d0a38a9/id-117-cipiti.pdf.

GIF (2011a), *Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems*, GIF/PRPPWG/2011/003, Generation IV International Forum, www.gen-4.org/gif/jcms/c_40413/evaluation-methodology-for-proliferation-resistance-and-physical-protection-of-generation-iv-nuclear-energy-systems-rev-6.

GIF (2011b), *Proliferation Resistance and Physical Protection of the Six Generation IV Nuclear Energy Systems*, GIF/PRPPWG/2011/002, Generation IV International Forum, www.gen-4.org/gif/jcms/c_40414/proliferation-resistance-and-physical-protection-of-the-six-generation-iv-nuclear-energy-system.

3 www.gen-4.org/gif/jcms/c_199670/gif-prppwg-bibliography-2022.

Risk and Safety Working Group

Since its inception in 2005, the RSWG developed the safety principles and attributes of Gen IV systems and the Integrated Safety Assessment Methodology (ISAM) as a technology-neutral toolkit to support design and evaluate risk and safety. In later years, the RSWG supported the application of ISAM, in coordination with the SSCs, to select Gen IV design tracks as documented in a series of system-specific white papers. Also in co-operation with the SSCs, the RSWG supported the preparation of system safety assessments as a summary of high-level safety design attributes and remaining R&D needs to promote a consistent approach to safety, risk and regulatory issues between six Gen IV systems. The latest output of this effort was the approval, in 2022, of the gas-cooled fast reactor (GFR) system safety assessment by the GIF Expert Group (GIF, 2022a). In 2022, RSWG membership included representatives from Canada, China, the European Union, France, Japan, Korea, the Russian Federation, South Africa, the United Kingdom and the United States. The IAEA Safety Department also participated as observer. All these previously completed reports are available on the GIF RSWG web page: www.gen-4.org/gif/jcms/c_9366/risk-safety.

The development of safety design criteria (SDC) and guidelines (SDG) for specific systems is an ongoing collaborative effort between the RSWG and the SSCs to establish the minimal requirements for the design, fabrication, construction, inspection, testing and operation of Gen IV prototypes. These SDC guideline reports are intended to fill the gap between the high-level GIF safety goals (fundamental safety principles) and the country-specific codes and standards (domestic regulations for the design of the reactor core, cooling systems and other structures, systems, and components). The activity was initiated by the SFR SDC Task Force, which gathers representatives from the RSWG and the SFR SSC, who produced the SDC and two SDGs on Safety Approach and Structure, System and Components (SSC) for the SFR concept. The work was then included within the RSWG's activities and extended to the other Gen IV concepts. In 2022, the SFR SSC Safety Design Guidelines report was submitted to the GIF Experts Group for approval. The extensive Experts Group member comments and suggested modifications required additional revision before its publication on the GIF public portal.

The interim LFR SDC report has been reviewed by the IAEA and members of the NEA Committee on Nuclear Regulatory Activities Working Group on New Technologies (formerly the Working Group on the Safety of Advanced Reactors). The comments are being analyzed by LFR pSSC members before discussing the outcome of the LFR pSSC considerations with the RSWG.

Also in 2022, the GIF Experts Group reviewed and approved the GFR Safety Design Criteria Report (GIF, 2022b). Based on SFR SDC as its starting point, the GFR SDC introduces specific revisions that capture the impact of the use of helium as a coolant with no thermal inertia and very low density. As for the previous SDC documents, the report was sent to the IAEA and NEA Working Group on New Technologies for revision.

The first revision of the VHTR Safety Design Criteria interim report was prepared and submitted for review. The VHTR report was developed by an informal subgroup of RSWG and VHTR SSC experts largely based on revisions of the IAEA SSR 2/1 requirements for high-temperature gas-cooled reactors. The VHTR report clarifies the requirements for design extension conditions, confinement function (vs. the conventional containment structure), unique coolant and decay heat removal system designs, and introduces new requirements for the coated fuel particles, non-electric applications and multi-module plant designs.

Another major RSWG accomplishment in 2022 was the development of a risk-informed framework for selection of licensing basis events and safety classification of systems, structures and components as an overlay of ISAM. A position paper that introduces the foundational concepts and main elements of such a framework was completed and distributed to the GIF Experts Group and Senior Industry Advisory Panel members. The approach aims to establish the event sequence categories considered in design to integrate the deterministic input and risk insights, define a generic frequency-consequence target structure to categorize the event sequences against the regulatory requirements, outline a process to classify the plant equipment based on their risk significance and role in plant safety, and support a selection of design basis accidents and design extension conditions consistent with safety classification of the responding plant equipment. Following its review, the paper was revised and resubmitted to the GIF Experts Group for approval.

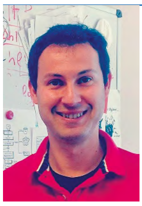
The RSWG contributed to the organization of the GIF Industry Forum in Toronto with the organization of two sessions dedicated to licensing. Speakers from regulatory and designer organizations discussed readiness plans for Gen IV systems and licensing approaches and experience. A workshop addressing ISAM was also organized by RSWG Canadian representatives as a side event.

International collaboration with the NEA continued in 2022 with the participation of RSWG representatives at the meetings of the Committee on Nuclear Regulatory Activities Working Group on New Technologies. The working group provided a review for the LFR SDC as well as the GIF risk-informed approach document. The RSWG was also invited to the kick-off meeting of the new Committee on the Safety of Nuclear Installations Expert Group on SMRs (EGSMR) to present the "Basis of Safety Approach for Generation IV Systems".

The RSWG also maintains a technical interface with the IAEA. Besides participation at the annual GIF-IAEA interface meeting (held virtually on 12-13 July 2022), the RSWG contributed to several relevant IAEA meetings. RSWG representatives were members of the International Advisory Board of the FR22 Conference and coordinated the GIF's contribution. The RSWG was also involved in the organization of the GIF-IAEA High Temperature Gas-Cooled Reactors and MSR technical meeting, where it presented its activities in support of VHTR and MSR systems. At the GIF-IAEA LMR Safety Workshop, the RSWG presented its work on the Safety Design Criteria and Guidelines for SFR and LFR designs. Finally, the RSWG provided papers on ISAM (Ammirabile, 2022a), risk-informed approach (Sofu, 2022), SDC and SDG for SFR (Futagami, 2022), and safety-security interface (Ammirabile, 2022b) to the Topical International Conference dedicated in 2022 to innovative systems and SMR. Pivotal was also the participation of RSWG experts to the IAEA initiative on the development of safety standards for novel advanced reactors in identifying specific safety features and assessing the applicability of design requirements and recommendations of non-LWRs systems.

Looking ahead, a new activity started in collaboration with the PRPPWG and the VHTR SSC for establishing 3S interfaces for the VHTR system. The initiative aims to gain insights of the interaction among the 3S by means of performing a pilot application with a design track identified within the VHTR SSC. The kick-off meeting was held in September with general agreement and enthusiasm to move on with this proposal. A common repository has been set, where individual subgroup members share safety/security and safeguards information. The effort is expected to develop over two years, resulting in a joint white paper on the VHTR 3S interface assessment.

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



Luca Immirabile
Co-Chair of the RSWG, with contributions from RSWG members

References

Ammirabile, L., T. Sofu, P. Gauthè (2022a), "GIF Integrated Safety Assessment Methodology (ISAM) and guidance for its implementation for novel advanced reactors", International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs, Vienna.

Ammirabile, L. and G.L. Fiorini (2022b), "Application of the objective provision tree tool for the safety-security interface assessment", International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs, Vienna.

Futagami, S. (2022), "Development of safety design criteria and safety design guidelines for Generation IV sodium-cooled fast reactors", International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs, Vienna.

GIF (2022a) *Gas Fast Reactor System Safety Assessment*, GIF/GFR-RSWG/2022/001, Generation IV International Forum, September 2022.

GIF (2022b), *Gas Fast Reactor Safety Design Criteria*, GIF/GFR-SDC/2022/001, Generation IV International Forum, September 2022.

Sofu, T. (2022), "GIF framework for risk-informed approach for design and licensing of novel advanced reactors", International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs, Vienna.

5 Task force reports

Advanced Manufacturing and Materials Engineering Task Force

Recent years have seen a very substantial resurgence of interest in advanced reactors and, in particular, small modular advanced reactors. Although there have been a number of predicted nuclear renaissances over the past 20 or 30 years, the present interest is significantly different in that it is based on the development of novel future technology rather than the refinement of present technology. Although some small modular reactors (SMR) under development are based on simplified light water reactor technology, many are based on Gen IV technology and use the inherent safety capability that can arise in non-water cooled reactors.

Key to the successful mass deployment of SMRs is the idea that the large-scale manufacture of components and structures in factory-like environments will reduce the capital cost of new nuclear deployment. This requires innovation in the nuclear supply chain, particularly in the areas of advanced manufacturing and materials engineering. However, nuclear design codes dictate that only qualified materials and processes can be used and in the nuclear industry, qualification can be a long and tortuous process. In addition, novel advanced manufacturing developments 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 was formed to see how collaborative R&D could be used to accelerate the implementation of advanced manufacturing into the nuclear supply chain to reduce the time to deployment of Gen IV and comparable advanced reactors.

In 2019, the task force undertook a survey of advanced nuclear reactor vendors, supply chain specialists, regulators and other experts. Advanced manufacturing methods were considered opportunities by more than 90% of respondents. The survey also identified that the greatest obstacle to adoption was perceived to be the creation and approval of appropriate codes and standards.

In February 2020, just before the COVID-19 pandemic, the task force held an international workshop at the NEA 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 are available at: www.gen-4.org/gif/jcms/c_115848/workshop-on-advanced-manufacturing.

The purpose of the workshop was to identify where collaboration could reduce the time to deployment of advanced manufacturing for advanced reactors. Attendees from reactor vendors, nuclear supply chain firms, regulators, national laboratories, and R&D providers enabled broad representation from across the nuclear industry and the cross-functional break-out sessions were particularly engaging and successful.

The output from the six break-out groups was discussed in the final session of the workshop and final conclusions were made. The recommendation of the workshop was that collaborative activities should be actively encouraged in the following areas:

- qualification: including new modality (e.g. real-time process qualification);
- design and modelling: including sharing data for design optimization and modelling and simulation (M&S) benchmarks.

The COVID-19 pandemic curtailed further progress in 2020, but in the first half of 2021 the task force undertook a second survey to investigate if and how the community's activities and opinions on these issues had changed. As in 2019, the survey was sent to nuclear reactor vendors, supply chain specialists, regulators, national laboratories and potential end users. The intent was again to gain the opinions of people actively engaged in the design and manufacture of advanced nuclear reactors. The size of the distribution list had doubled since 2019, providing anecdotal evidence that the community working on the manufacture and deployment of advanced reactors had continued to grow. Consequently, the number of survey responses also concomitantly increased.

Comparing the responses of the 2021 and the 2019 surveys, there was an increase in active interest in advanced manufacturing, with significant increases in reported efforts to support codes and standards and to secure regulatory approval. There was also a substantial decrease in the number of respondents who had adopted a "wait and see" attitude. There was also a change in the focus on the reactor components being considered for advanced manufacturing, with an increase in interest in reactor internals and heat transfer infrastructure (e.g. pipes, valves and heat exchangers) and a decrease in interest in reactor vessels, fuel cladding and fuel assemblies. There was also an increased focus on advanced manufacturing techniques of traditional nuclear materials, with most effort on stainless steels, low alloy steels and nickel alloys.

The community's focus on qualification and M&S as a potential enabler for qualification of advanced

23 June (Thu)	GIF AMME Workshop on Qualification of 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, Chair AMME-TF
13:10 - 13:30	How do new materials and processes get into codes? Cecile Petesch, RCC-MRx Sub-Committee Chair, CEA (15mins+5min questions)
13:30 - 13:50	The eVinci proposed composite materials qualification process Jurie Van Wyk, Westinghouse (15mins+5 min questions)
13:50 -14:10	WGSAR's activities on Materials Qualification and Lifetime Performance Raj Iyengar, NRC (15mins+5 min questions)
14:10 - 14:30	Aerospace experience of qualifying advanced manufacturing? Richard Russel, NASA (15mins+5min questions)
14:30 -14:50	Question for groups: what should happen next to enable qualification of advanced manufacturing for High Temp Reactors
Session 2 – Group activity 1	
14:50 - 15:00	BREAK
15:00 -16.30	Attendees split into allocated groups, which undertake the following activities with the group Moderator/Rapporteur: a. what could happen next b. analyse multiple ways forward c. Prioritise, things that could happen next d. Agree communication for Rapporteur to give to meeting (Can develop presentation in break if necessary)
16:30 —17:00	Break
Session 3 – Final Group Reporting and Meeting Outcomes	
17:00 - 17:30	Each Group present their findings and recommendations
17:30 - 18:00	Summarise discussion and consensus of meeting
18:00	End of Meeting

Table 5.1. Structure of the June 2022 workshop on qualification of advanced manufacturing

manufacturing and materials was confirmed. Enthusiasm for engagement with the task force had also grown, with 54% of respondents reporting a high or very high interest in a Modelling and Simulation Workshop and 84% reporting a high or very high interest in a Qualification Workshop.

Thus, the task force developed a series of workshops focusing on both qualification and how M&S can be used to accelerate qualification. As the outputs of the M&S were seen to be a key input to the qualification workshop, a specific virtual workshop on how M&S can enable the qualification of advanced manufacturing was held on 8-9 November 2021.

As in the 2020 workshop, the 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 at: www.gen-4.org/gif/jcms/c_82829/workshops.

The first half of the workshop provided an overview of the potential for M&S to support the nuclear qualification of advanced manufacturing. In the second half, attendees were assigned to groups and asked identify opportunities for M&S to accelerate qualification of advanced manufacturing, then to prioritize, by likelihood of success, the best opportunities for deployment.

The M&S R&D opportunity areas identified were:

- collaborative M&S R&D to control the process;
- collaborative M&S R&D to predict the microstructure and/or the material/component properties;
- collaborative R&D to "share" M&S models, software and data;
- collaborative M&S R&D focused on codes and standards.

However, identifying specific R&D activities to be prioritized was more challenging, with a wide range of possible actions identified. Consequently, a significant analysis of the group's outputs was subsequently undertaken and the findings of this analysis presented in a paper at G4SR-4 in Toronto, Canada, on 3-6 October 2022 (van Rooyen et al., 2022). This analysis proposed machine learning as a promising method to solve the specific M&S activities suggested and the following potential machine learning collaborative activities were identified:

- Demonstrate machine learning and correlate processing history to material structure.
- Develop and disseminate benchmark problems for applying models to accelerate material qualification.
- Identify:
  - (i) existing datasets for use in training/validating models;
  - (ii) common and/or open-source software;
  - (iii) existing datasets for use in training/validating models;
  - (iv) key ranges of operating parameters for models to target (temperature, stress, fluence, etc.).

Following the analysis of the November 2021 workshop, the task force organized two workshops focused on collaboration in 2022. Table 5.1 shows the structure of the first virtual workshop held in June 2022.

Session 1 was designed to inform, stimulate and help define the workshop area and purpose. It contained presentations from informed experts from the nuclear industry, standards organizations and regulatory bodies together with a talk from NASA, which has already implemented qualification procedures for advanced manufacturing components.

Like previous workshops, Session 2 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.

As in previous workshops, there was a large virtual international participation drawn across the whole nuclear industry and its regulation and compliance systems and nearly all attendees made contributions. The broad results confirmed conclusions of previous workshops, but discussion did not converge and it was not possible to narrow down a large list of potential solutions to a few actionable items. It probably did not help that the format of the workshop had changed from two days to one day.

However, the workshop provided the task force with valuable insights on how to focus its second workshop in 2022 to enable the identification of distinct actions. Thus, the decision was taken to use the June 2022 workshop as a template for the design of the task force's advanced manufacturing workshop at the 2022 GIF Industry Forum, which was held in conjunction with the G4SR-4 Conference in Toronto in October 2022. This joint workshop was co-organized with the Canadian Advanced Manufacturing in Nuclear Alliance and held on 4 October 2022.

The workshop was designed to develop and disseminate knowledge of the steps required to successfully introduce advanced manufacturing to the supply chain of advanced nuclear reactors. As such, it was aimed at a broader audience than the previous workshops and advertized as being of interest to:

- Gen IV reactor developers looking to use advanced manufacturing methodologies to reduce the time and cost of advanced reactor development and manufacture;
- supply chain companies seeking a competitive advantage in advanced manufacturing innovations and knowledge in factory manufacturing;
- personnel involved in nuclear regulation and nuclear standards looking to gain knowledge on advanced manufacturing and the innovation it can bring to advanced reactor deployment;
- researchers seeking information on an advanced manufacturing research roadmap, innovation discoveries and enabling infrastructure, such as a modern advanced manufacturing research institute.

To address the objectives of the above-mentioned target audiences and the needs of the task force, a hybrid structure was devised.

Session 1 consisted of presentations by speakers with a focus on how the adoption of advanced manufacturing technologies can reduce the cost and time to deployment of advanced reactors and the challenges still to be met regarding its widespread adoption for advanced high-temperature reactors.

Session 2 involved two break-out session:

- Track 1: Advanced Manufacturing Demonstrations at McMaster Manufacturing Research Institute, designed for an audience with a broader interest in advanced manufacturing and keen to see the demonstrations of advanced manufacturing innovations at work while engaging further discussion with experts from the McMaster Manufacturing Research Institute during demonstrations on the various topics presented in Session 1.
- Track 2: a participatory physical interactive workshop in the style of previous task force workshops, where participants worked collectively with a focus on the qualification of advanced processes and manufactured components, which has been identified as a key barrier to advanced technology adoption.

Attendance at this workshop was rather less than the previous virtual workshops but, as ever, the participants' commitment was excellent and despite (or maybe because of) the restricted attendance and time available, a large list of potential solutions was narrowed down to a few actionable projects. Importantly, attendees were eventually asked to answer the simple question: What should be done next to accelerate the qualification of advanced manufacturing for use in Gen IV reactors? The data obtained from this October 2022 workshop are still being analyzed but a broad consensus enabled the following recommendations for future actions:

- Disseminate a "short-form" survey to vendors every year. Limit questions to try to promote regular response aiming to identify key manufacturing issues and how they change over time.
- Identify geometries, loading conditions, materials and the corresponding critical flaws/limit states that are broadly representative of a wide range of likely advanced manufacturing Gen IV components (minimum requirements).
- Round robin benchmark studies for accelerated testing, M&S, staggered qualification, and in situ monitoring approaches – demonstrating they can replace long-term testing.
- Set up a forum for summarising/sharing/co-ordinating /harmonising work at codes and standards bodies.

The task force will report on the conclusions drawn from all its workshops and formulate plans for its future activities in 2023.

References

van Rooyen, I. et al. (2022) "Accelerating the adoption of advanced manufacturing technologies for Gen IV nuclear reactors through international collaboration," International Conference on Generation IV and Small Reactors, G4SR-4, Toronto, Canada, 3-6 October 2022.

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

Non-Electric Applications of Nuclear Heat Task Force

The decarbonization of electricity generation alone is insufficient to meet the challenging CO_2 emissions reduction targets. Energy demand from the industrial and transport sectors offers significant potential for further emissions reductions through the direct use of nuclear-generated heat and/or process intermediates that may be produced using nuclear heat and electricity (e.g. hydrogen).

While nuclear plants' primary energy product is heat, most plants were previously dedicated to electricity generation because the primary heat was of low value relative to what could be provided by fossil fuel combustion. As restrictions are placed on emissions and fossil fuel prices are increasing in many regions, nuclear technologies promise to be more competitive, particularly for higher temperature Generation IV reactor technologies.

The International Atomic Energy Agency (IAEA) introduces several broad applications of nuclear beyond electricity (Figure 5.1). Each of those applications include additional complexities, including several possible technologies for implementation, as well as further breakdowns of each type of application on the basis of various parameters, for example the temperature required.

Beginning in November 2020, the GIF facilitated an open exchange of expert views on Gen IV systems regarding applications of nuclear fission-generated heat beyond the electric grid. These activities suggested a path forward for the GIF to leverage work being conducted internationally and identify the benefits that Gen IV reactor systems could bring to the non-electric energy sector in the context of future energy markets.

At the 51st GIF Policy Group meeting held on 20-21 May 2021, the GIF Policy Group members decided to establish a new Task Force on Non-Electric Applications of Nuclear Heat (NEANH TF), which was officially launched in October 2021 for a period of 24 months.

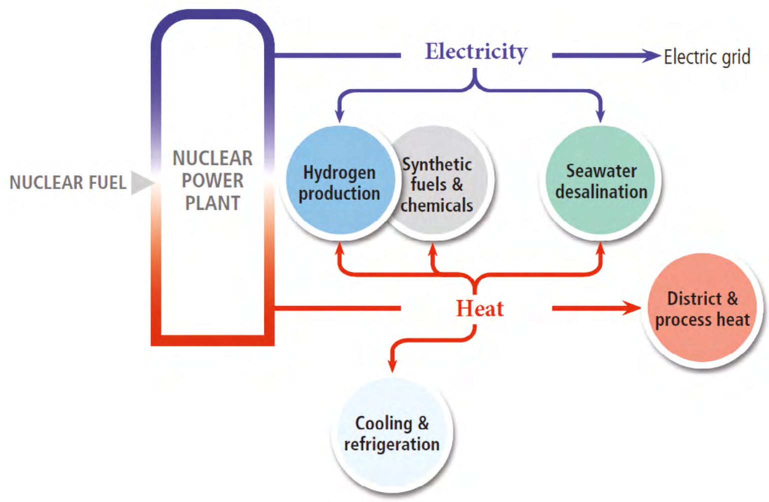
The NEANH TF is a unique initiative, as it connects a range of stakeholders involved in the development of Gen IV systems for a range of applications, including industrial end users, technology developers, stakeholders in GIF signatory countries, and international organizations such as the IAEA and the NEA.

Under the NEANH TF, a virtual workshop was held in July 2022 to exchange knowledge and perspectives among task force and GIF members and to align members in advance of engaging the end-use community. This virtual workshop identified commonalities, key differences and gaps among work being performed by participating organizations.

Objectives of the GIF NEANH Task Force

GIF-type reactor technologies can be employed for co-generation and integration into energy markets with high fractions of renewables, providing ancillary services to support grid stability, enhanced flexibility and high-quality heat. The NEANH TF will identify and review these systems and develop key performance indicators, such as technology readiness levels, timeliness, adaptability to geographical conditions, CO_2 emissions reduction potential, cost and boundary conditions for economic viability.

Figure 5.1. Five families of non-electric applications of nuclear energy, inspired by the IAEA



Source: adapted from IAEA.

The objectives of the NEANH TF are to:

- Articulate the GIF's position regarding the coupling of non-electric applications to nuclear energy systems, focusing on GIF systems and associated technologies, in both the short and long term.
- Improve the general level of knowledge of GIF members regarding recent and ongoing research activities on NEANH coupled with Gen IV systems, and ongoing research for non-electric applications using water-cooled systems that may be leveraged for Gen IV applications.
- Enhance the general level of knowledge of GIF members on non-grid applications of nuclear systems.
- Highlight relevant system configurations, including system feasibility, emissions impacts, etc.
- Develop a network to connect the GIF to the high-temperature community outside the nuclear field.
- Explore systems analysis with regard to key performance indicators. Systems analysis may include the integration within a system of other clean energy options, and analysis related to time-of-use of production, transmission, end-of-life costs, and the interactions between electricity and heat markets.
- Provide input to decision makers, industry, licensing authorities, investors, etc. on configurations that will support achieving policy goals, and R&D efforts to be undertaken.

The GIF position on non-electric application of nuclear heat

To achieve energy supply security whilst meeting the constraint of net zero CO_2 emissions in 2050, all potential solutions must be considered for assessment. Some of the solutions to be deployed by 2050 are still in their infancy and require further development. This need offers great opportunities for development and innovation across all energy sectors, including the nuclear industry.

The GIF signatories agree that Gen IV systems could and should provide diversified service offers ranging from electricity to numerous heat applications at the required large scale to achieve a significant societal impact in terms of greenhouse gas emission reduction, security of energy supply, and energy affordability.

The GIF signatories are determined to utilize their collective skills, knowledge, and expertise to propose and evaluate relevant coupling and cogeneration options for the short and medium to long term. The Task Force will identify major obstacles that may arise from the energy system combinations under consideration, contribute to their resolution, and define a portfolio of realistic, technically and economically feasible NEANH solutions coupled to Gen IV reactors to help accelerate decarbonization.

Figure 5.2. NEANH Workshop, Toronto, Canada, 3 October 2022

NEANH position paper

To solidify the GIF's position regarding the coupling of NEANH to GIF systems, the NEANH Task Force, in collaboration with the broader GIF community, produced a position paper. The position paper was published in November 2022, and highlights the relevance of the NEANH in the near term and in the transition to a future energy landscape. The paper also addresses considerations, opportunities and challenges related to coupling NEANH with Generation IV nuclear reactor systems.

GIF Industry Forum

A full-day open workshop was also held in Toronto, Canada on 3 October 2022 in conjunction with the GIF Industry Forum and co-organized by the International Framework for Nuclear Energy Cooperation.

The primary objective of this full-day workshop on NEANH was to connect the GIF to the high-temperature community outside the nuclear field, and to bring together stakeholders to establish connections between the research community and industry, engaging both nuclear technology developers and energy end users.

Through panel discussions and interactive dialogue, deployment opportunities were discussed in which nuclear energy systems could be used to support heat and electricity demands outside the power sector.

- The research community spoke about computational tools and facilities that could support systems analysis and demonstrations to accelerate the path to commercial advanced reactor deployment, particularly for non-traditional applications.
- Members of the energy end-use community shared details of their energy needs and requirements and raised potential issues regarding the integration of nuclear energy to drive these processes.
- Nuclear technology developers shared their expected performance capabilities and deployment timelines for their systems.

The event was well-attended, with more than 150 participants, including Gen IV reactor developers, energy system modellers, industrial energy users, researchers and other stakeholders. Key points identified by the panellists and ensuing discussions are highlighted below.

- It is technically possible to couple these heat-intensive processes with nuclear energy, and there are successful precedents.
- There is a significant need to develop and share detailed data by relevant parties, including validation of expected cost and performance data via demonstration projects.
- Certain reactor types are capable of delivering steam of 550°C as feedstock for standard industrial processes, representing a very sizable market in most industrialized countries. An additional, equally large market for even higher temperatures can be supported by specific high-temperature reactors and temperature-boosting technologies.
- In addition to primary heat transfer from the reactor core to power conversion that is standard in nuclear reactor designs, many advanced reactor developers propose an additional heat transfer loop with thermal energy storage to isolate the nuclear reactor from the heat customer. This would grant greater flexibility and enhanced safety of the overall integrated system.
- Nuclear energy can be a "drop-in" solution expected to directly replace fossil fuel-powered steam supply operations, but it may not be able to replace direct fossil fuel-fired operations.
- Hydrogen has promising applications, but its role should be considered on a case-by-case basis that considers the benefits it offers as a feedstock and/or energy carrier with respect to the intended energy use sectors, in addition to its potential for decarbonization.
- There are a range of options for owner-operator models. Energy end users do not desire to own and operate a reactor themselves, but are customers for nuclear-generated heat, steam and electricity.
- Cost is a significant driver for adoption, as are energy security, reliability and social acceptance. Regulatory processes are viewed as a significant barrier for adoption; clarity is needed regarding interaction across the regulatory bodies for nuclear systems and industrial processes.

Next steps

There is an opportunity to facilitate early interaction between nuclear energy and conventional industry regulators and to familiarize industrial end users with nuclear energy. This includes identifying relevant stakeholders that would benefit from ongoing conversation, including the licensees, regulators and industrial groups such as mining, oil and gas, and chemicals production. There is also a need to evaluate drivers outside of North America, including by exploring issues related to energy security or increased stringency with emissions caps.

The members of the NEANH Task Force continue to meet on a monthly cadence to advance the task force's goals. The task force is currently developing a database to capture key performance indicators from ongoing and historic work related to NEANH systems.

Additional information on the GIF NEANH Task Force is available on the Generation IV International Forum website at: www.gen-4.org/gif/jcms/c_207893/neanh.

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

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

Summary of the October workshop and artificial intelligence session from the GIF Industry Forum

The Gen IV Industry Forum, a new event replacing the traditional GIF Symposium in 2022, provided a more industry-oriented approach to foster collaboration in Generation IV systems between the private and public sectors. The event was held on 3-7 October 2022, in Toronto, Canada, alongside Canada's 4th International Conference on Generation IV and Small Reactors (www.g4sr.org). The event disseminated the GIF's most recent work relevant to the nuclear industry and for enhancing collaboration with leading small modular reactors (SMR) developers. Discussions included future work that the GIF could undertake to accelerate demonstration and deployment of advanced reactor technologies.

To explore these opportunities, a special Senior Industry Advisory Panel (SIAP) session on SMRs was organized on the margins of the Gen IV Industry Forum. The session on 6 October 2022 brought together 15 SMR vendors from 7 countries, who provided feedback on GIF products and made requests for future GIF work.

During this session, the following GIF papers on key topics of interest were presented:

- materials: Overview of GIF VHTR Materials R&D and the Gen IV *Materials Handbook* (William Corwin, Advanced Reactor Materials LLC, United States);
- physical protection: Proliferation Resistance and Physical Protection: Insights for GFR, SFR, LFR, SCWR, MSR, and VHTR (Ben Cipiti, Sandia National Laboratories, United States);
- economics and cost reductions: Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review (Antoine de la Chevrotière, NRCan, Canada).

The papers were met with great interest and were found useful for SMR vendors. It was agreed to continue this practice of disseminating the GIF's work to industry. The next SIAP session with industry is tentatively planned on the margins of the World Nuclear Exhibition 2023 in Paris, France.

A technical session on Artificial Intelligence for Nuclear was held on 4 October 2022 to discuss how artificial intelligence (AI) can help in the development of Gen IV reactors. International experts from industry, national laboratories and regulators were invited to discuss the following topics:

- how best to use AI-powered technologies to accelerate and derisk GIF projects;
- the priority for short- and long-term AI strategies;
- whether smaller subgroups can be created to collaborate on different subject matters such as: AI security, AI ethics, AI for nuclear reactor solutions, AI for asset management;
- understanding current challenges (regulatory acceptance, data sharing) and capabilities;
- sharing AI success stories/use cases from the private and public sectors;
- identifying collaboration opportunities to share data and develop, train and test AI models.

The session was chaired by Professor Nawal Prinja, SIAP Vice Chair, with the following six presentations:

- "How AI is Empowering the Future of the UK Nuclear Industry – A Position Paper" (Caroline Chibelushi, Nuclear Institute, United Kingdom);
- "Application of AI within Data-centric Engineering and Autonomous Systems" (Nigel Tate, Rolls Royce, United Kingdom);
- "The Challenge of Autonomous Operation and Automated Reasoning as an AI Enabler" (Robert Ponciroli, Argonne National Laboratory, United States);
- "AI-powered Cognitive Search for Information for Nuclear Knowledge Management" (Thomas Devraj, S & P Global, United States/United Kingdom);
- "Development of the Regulation of AI for Nuclear Applications in the UK" (David Smeatham, Office for Nuclear Regulation, United Kingdom);
- "The IAEA's role in the Deployment of AI for Nuclear Power" (Aline Des Cloizeaux, IAEA).

During the discussion moderated by the chair, the NEA presented a view that AI is part of the digital transformation that is taking place in the industry and requires work on data management, interfaces and standards. Overall, delegates felt that there is great potential in using AI technology to help accelerate and derisk R&D for Gen IV reactors.

It was agreed that another similar event needs to be held, and the SIAP should work closely with the IAEA for the initiatives being taken to industrialize the use of AI technology.

Recent 2022 highlights, including industry perspective on GIF engagement

Nuclear energy could play a significant role in attaining the net zero targets to which an increasing number of OECD countries are committed. The International Energy Agency's *Net Zero by 2050* assumes an increase in electricity generated by nuclear power plants from 2 698 terawatt hours (TWh) in 2020 to 5 497 TWh in 2050, which corresponds to an increase of 415 gigawatts (GW) of capacity in 2020 to 812 GW in 2050 (IEA, 2021). Continued operation of the existing fleet, as well as new builds 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.

A further contribution to decarbonizing the world's energy sector can be made by using heat (steam) and electricity from nuclear reactors for non-power applications: district heating, production of hydrogen and synthetic fuels, or desalination, which are all processes that today mainly operate using fossil fuels (coal, oil, gas) or biomass. Another important strength of nuclear power is its stable production cost, and the potential to offer price stability to industrial consumers. Co-generation can also dramatically reduce primary energy resource consumption by greatly increasing the efficiency in energy use, from a global average of 37% for conventional power generation to 80% for the co-generation of heat and power.

Applications of nuclear thermal energy to date have been limited to low-temperature applications such as desalination and district heating, which require thermal energy at temperatures up to a maximum of 200°C, which can be supplied by the current generation reactors. All nuclear co-generation to date has used less than 1% of the total thermal energy output of the world's nuclear fleet. There have been few applications of nuclear thermal energy for industrial processes. The advanced nuclear reactors that are under development as Gen IV reactors and several types of SMRs have higher outlet temperatures and are therefore better suited for supplying heat to industrial processes.

Studies performed in 2022 (NEA, 2022) show that nuclear co-generation is part of the solution to achieve the global energy decarbonization goals set under the IEA's net zero scenario. Co-generation is an integral part of the future of nuclear energy, as it allows further reductions in CO_2 and air pollution due to fossil fuel burning (including biomass), improves the efficiency of the plant and limits thermal pollution.

The potential market for nuclear co-generation is significant – even targeting a fraction of the heat market could translate into a high number of new reactors – provided the business case is favourable. SMRs could be more suited for desalination without emitting greenhouse gases, as the demand for desalinated water is growing rapidly. The advanced nuclear reactors that are under development as Gen IV reactors and several types of SMRs will have higher outlet temperatures and could therefore be better suited for supplying heat to industrial processes.

The report calls for important decisions to be taken concerning nuclear power in different areas, including the extent of government support for advanced modular reactors. These technologies still under development hold the potential to expand markets of nuclear power beyond electricity and foster decarbonization in hard-to-abate sectors. The current and next decade will be critical to demonstrate the commercial viability of Gen IV systems in order to achieve sufficient market uptake to remain relevant and aligned with net zero targets.

There is also a need to better inform the public and policy makers – as well as industry at large – of the potential of nuclear co-generation. Economic and environmental benefits are decision drivers, but relevant stakeholders need to be involved at the early planning stage in order to build public acceptance.

The GIF therefore has a role to play in advancing Gen IV concepts towards the construction of demonstration and/or first-of-a-kind units through well-designed collaboration with private actors. 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 chairs, task forces and working groups, and with the guidance of the members of the GIF Policy Group.

SIAP continues to advise and support the GIF in building a better connection with the private sector and increasing the involvement of industrial partners. It also continues to provide industrial insight for GIF activities, create opportunities for the industry to bring together different viewpoints, and share its experience in developing and applying new technologies for Gen IV concepts.

In particular, SIAP will focus on how to enhance industry engagement opportunities with the GIF and how to make GIF products more visible and even more instrumental to end users.

References

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

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

NEA (2022), *Beyond Electricity: The Economics of Nuclear Cogeneration*, OECD Publishing, Paris, www.oecd-nea.org/jcms/pl_71699/beyond-electricity-the-economics-of-nuclear-cogeneration.

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Appendix 1. Country reports 2022

Australia

Australia continues to be a committed and cooperative member of the Generation IV International Forum for the joint development of the next generation of nuclear technology, which is vital for the future of the nuclear energy industry and for the sustainable development of the planet.

While Australian Government policy continues to prohibit the civilian use of nuclear energy in the country, it continues to recognise that nuclear energy is a mature technology used to deliver reliable electricity in many countries, with zero greenhouse gas emissions at the point of generation and low lifecycle emissions. The Australia Government supports the peaceful use of nuclear science and technology and maintains the highest standards of safeguards and transparency measures to ensure the non-proliferation, safety, and security of nuclear material and technology.

On 14 March 2023, the Prime Minister of Australia, the Prime Minister of the United Kingdom, and the President of the United States announced the optimal pathway for Australia to acquire a nuclear-powered submarine (NPS) capability. The announcement followed the conclusion of an 18-month study. Australia will acquire Virginia class NPS by the early 2030's, with trilaterally developed SSN-AUKUS vessels from the early 2040's. ANSTO continues to play a role in supporting the AUKUS program and implementation of the optimal pathway, including advice to the Australian Government.

As mentioned in previous reports, the Australian Government announced in late 2021 funding for the design of a new, world-leading nuclear medicine manufacturing facility to be constructed at ANSTO's Lucas Heights campus. Since our last report, ANSTO has completed the business case for the delivery of a facility that will support the supply of crucial diagnostic and theranostic medical radioisotopes and ensure Australia's sovereign production capability, as well as enabling international supply. This work has been presented to the Government for decision.

Australia continues to develop its planned National Radioactive Waste Management Facility for the permanent disposal of low-level radioactive waste and temporary storage of intermediate-level radioactive waste managed by the Australian Radioactive Waste Agency (ARWA). ARWA recently updated the national inventory of radioactive waste for the first time since 2018. The national inventory will be used to develop the safety case and design of the planned National Radioactive Waste Management Facility.

As part of its overall nuclear medicine production facilities, Australia is building a first of a kind waste treatment facility. The SyMo plant will use ANSTO's Synroc technology to treat the intermediate level waste stream arising from the extraction of molybdenum-99. Works are continuing with recent installation of the key technology in the process – a hot isostatic press used to produce a durable, disposal ready waste form product. Commissioning and operational readiness activities will continue over the next few years.

As mentioned in previous reports, Australia, through ANSTO, is the Chair of the Regional Cooperative Agreement for Research, Development and Training Related to Nuclear Science and Technology for Asia and the Pacific (RCA) in 2023. We look forward to hosting the National Representatives Meeting in May this year and continuing the focus of the members on the importance of meeting international obligations to cooperate in the peaceful uses of nuclear science and technology.

Australia's OPAL reactor will undergo a planned long shutdown from 18 March 2024 to 5 July 2024, to undergo necessary upgrades and scheduled maintenance. A key driver of this planned shutdown is replacing the reactor's unique cold neutron source, which has an operational life of 15 years. The replacement cold neutron source offers increased scientific performance. The shutdown of OPAL will result in a short-term disruption to the production of some nuclear medicines, irradiations, such as silicon, and research activities, however, these upgrades will mean ANSTO can continue to provide a safe and reliable nuclear medicine supply into the future. OPAL is essential to ANSTO's production of nuclear medicine for supply to hospitals and clinics around Australia and overseas. ANSTO will work with supply chain partners to ensure where possible, any disruption to the supply of nuclear medicine is minimised.

A focus for Australia in its role as a member of the Generation IV International Forum continues to be on the mutual benefits reaped from international cooperation in programs that underpin the next generation of nuclear technology. We are pleased to provide the majority of our contributions in support of high temperature reactor (VHTR) projects. We also undertake some research activities in support of molten salt reactor (MS) projects and continue to support the movement of GIF MSR research from a provisional to a full system steering committee in due course.

Finally, GIF members may have seen media reports of a civilian use radioactive source that was lost for some time in a remote area in Western Australia. The Cs-137 capsule measuring about 6 mm in diameter by 8 mm high was lost on a greater than 1 000 km long stretch of highway. After over a week of searching specialised support from ANSTO was called in and using CORIS-360 – a position sensitive gamma-ray detector technology developed by ANSTO – the source was found within 24 hours.

Canada

Update on government policy and actions

Since the launch of Canada's SMR Action Plan (2020), 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 119 partner organizations from across the country as of September 2022. It includes 520 concrete actions that partners are taking to advance the development, demonstration and deployment of small modular reactor (SMR) technologies in Canada. Under the Action Plan, the Government of Canada has established an SMR Leadership Table, convening the Canadian nuclear family, from mining to production, as well as indigenous peoples. The Indigenous Advisory Council of the SMR Action Pan, created in 2021 to enable a co-ordinated indigenous leadership voice from across Canada on the development of SMRs, co-chaired the most recent SMR Leadership Table.

In Budget 2022, the Government of Canada announced new funding for SMRs, including almost CAD 70 million for Natural Resources Canada to support research to minimize waste generated from SMRs, support the creation of a fuel supply chain, strengthen international nuclear co-operation agreements, and enhance domestic safety and security policies and practices. Also, CAD 250 million over four years was allotted to Natural Resources Canada to support pre-development activities of clean electricity projects of national significance, such as inter-provincial electricity transmission projects and SMRs. The Canadian Nuclear Safety Commission received over CAD 50 million to build capacity to regulate SMRs and work internationally on regulatory harmonization. In addition, the Canadian Infrastructure Bank now includes an expanded mandate to facilitate decarbonization, including SMR technology. In line with its new mandate, the Canadian Infrastructure Bank committed CAD 970 million towards Ontario Power Generation's 300 megawatt SMR that is to be built at the Darlington Nuclear Power Plant.

In 2022, four provincial governments (Alberta, New Brunswick, Ontario and Saskatchewan) released their strategic plan for the advancement of SMRs. The plan highlights how SMRs can provide safe, reliable, zero-emissions energy to power growing economies and populations, while creating new opportunities to export Canadian knowledge and expertise around the world.

Utilities, vendor and regulatory SMR updates

In March 2022, Canada's Minister of Innovation, Science and Economic Development announced CAD 27.2 million to Westinghouse Electric Canada to support its SMR, the eVinci micro-reactor. This technology has the potential to help communities that rely on diesel fuel to transition to a cleaner source of energy.

In July 2022, Ontario Power Generation and X-energy signed a framework agreement to pursue opportunities to deploy Xe-100 SMRs for industrial applications in Canada. By providing both high-temperature steam and power production, the Xe-100 high-temperature gas-cooled reactor can support heavy industry, including oil sands operations, mining applications and other industrial processes.

In June 2022, Saskatchewan power utility SaskPower also selected General Electric-Hitachi's BWRX-300 SMR for potential deployment in the province in the mid-2030s after an assessment process in which it looked at several SMR technologies. SaskPower is considering two sites for the potential construction of Saskatchewan's first nuclear power facility and aims to take a decision in 2029 on whether to build an SMR.

In August 2022, Canada's Terrestrial Energy and Invest Alberta (the Government of Alberta's crown corporation promoting high-value investments) signed a memorandum of understanding to support the commercialization of Terrestrial Energy's Integral Molten Salt Reactor Generation IV SMR plant in Western Canada. Terrestrial Energy's IMSR plant has the potential to supply heat and power to many industrial activities, including those in the Alberta oil and gas and petrochemical sectors.

Industry action on climate change and SMRs

In June 2022, two oil sands industry groups combined to form the Pathways Alliance: Canada's Oil Sands Innovation Alliance, which includes Canadian Natural Resources, Cenovus Energy, Conoco Philips Imperial, MEG Energy and Suncor Energy, that together operate about 95% of Canada's oil sands production. SMRs have been identified as a possible pathway and an SMR Working Group has been created. Oil sands operations represent the largest industrial source of greenhouse gas emissions in Canada.

Other activities and developments

The refurbishment of CANDU reactors at Darlington is on track for overall completion by the end of 2026. The refurbishment will enable the production of clean, safe, low-cost and emission-free electricity from Darlington for an additional 30+ years. In September 2022, OPG announced that it will continue operating its Pickering plant one year longer than planned, until 2026, and will launch a study to examine the feasibility of refurbishment.

Bruce Power's Unit 6 Major Component Replacement Project also continues to proceed on track. The project focuses on the replacement of key reactor components in units 3-8, including steam generators, pressure tubes, calandria tubes and feeder tubes.

The Hydrogen Strategy for Canada identified synergies with nuclear energy as it can be used to produce low-carbon hydrogen via electrolysis or high-temperature processes. The Nuclear Working Group established four task forces to explore specific topics related to hydrogen production opportunities; technology and infrastructure; safety, regulations, codes and standards; and economics, financing and policy. The analytical work performed by the task forces will inform and highlight specific opportunities to produce hydrogen with nuclear energy in Canada and will build on Canada's unique history and capabilities.

The Canadian Space Agency, in partnership with Natural Resources Canada, is studying concepts for space nuclear power systems, including fission power systems for the lunar surface in the order of 30 kilowatt electrical (kWe). Through the Federal Nuclear Science and Technology programme, the Government of Canada is supporting a project to evaluate current and past Canadian nuclear technologies suitable for space applications in support of identifying Canada's contribution to international priorities in space. Canada also participates in the International Space Exploration Coordination Group, which has identified space nuclear power systems within a list of critical technologies that need to be developed or matured to advance the group's mission scenarios.

International update

In October 2022, Minister Wilkinson participated in the IAEA's International Ministerial Conference on Nuclear Power in the 21st Century in Washington, DC, where he stressed the importance of enhanced international partnerships to diversify and secure the global market for nuclear fuel, services and technologies. This included delivering Canada's National Statement and meeting with key counterparts to discuss opportunities to enhance cooperation in areas of mutual interest.

In October 2022, the Canadian Nuclear Society hosted the 4th International Conference on Generation IV and Small Reactors (G4SR-4) in Toronto in cooperation with international partners in industry and government. The GIF Industry Forum was co-located and ran in parallel to the conference.

Canada's participation in the GIF

Canada participates in the supercritical water-cooled reactor (SCWR), very high-temperature reactor (VHTR) and molten salt reactor (MSR) systems. In 2022, Canada expanded its participation with the CNL joining the VHTR Materials project and the MSR provisional System Steering Committee.

China (People's Republic of)

Nuclear energy policy

- On 29 January 2022, the National Energy Administration issued the Modern Energy System Program during the Fourteenth Five-Year Plan, stressing the need to actively and orderly promote the construction of coastal nuclear power projects; actively promote the advanced reactor demonstration projects; promote the comprehensive use of nuclear energy; promote the optimization, upgrading and demonstration application of key technologies of Gen III nuclear power, support early R&D of the controlled nuclear fusion as well as actively partake in international cooperation in nuclear energy. By 2025, the installed capacity of nuclear power in operation will reach about 70 GW.

Nuclear Energy Development and Application

- In 2022, the State Council approved the commencement of a total of ten units of five nuclear power projects. As of 31 December 2022, there were 54 nuclear power units in operation, with a rated installed capacity of 55.8 GW and 22 units under construction, with a rated installed capacity of 25.2 GW on the Chinese mainland.
- On 25 March 2022, Unit 6 of the Fuqing Nuclear Power Plant was put into commercial operation, and on 18 April 2022, the K3 unit in Karachi Pakistan passed the provisional acceptance. So far, the first batch of four units of "Hualong One" at home and abroad have been fully completed and put into operation.
- On 9 December 2022, the dual reactors of Shidao Bay HTR-PM reached initial full power operation for the first time, achieving stable operation under the mode of "two reactors with one turbogenerator", verifying that all systems of the demonstration project meet the design functions.
- On 30 November 2022, the nuclear island installation of "Linglong One" multi-purpose modular small reactor in Changjiang Hainan was officially started, 75 days ahead of schedule.
- On 27 May 2022, the industrial steam supply project in Tianwan Jiangsu was fully started. It uses the steam of Units 3 and 4 of the Tianwan nuclear power plant as the heat source and transmits the steam to Lianyungang Petrochemical Industrial Base. It is expected to be put into operation for steam supply by the end of 2023.

The GIF's activities in China

- **Sodium-cooled fast reactor (SFR)**: Chinese experimental fast reactor (CEFR) is still shutdown; the licensing for future operation and irradiation application is underway. CFR-600 is commissioning; the primary and secondary circuits had been filled with sodium by the end of 2022.
- **Very high-temperature reactor (VHTR)**: HTR-PM reached initial full power on 9 December 2022 for both modules. It is scheduled to finish the commissioning test in mid-2023. The R&D in Fuel and Materials Projects is going as planned, joining the Hydrogen Production Project, which is still ongoing.
- **Supercritical water-cooled reactor (SCWR)**: The fuel assembly design of CSR 1000 has been improved with higher hydraulic performance. Its integral design scheme was well-prepared and is ready for international peer review. The preliminary design scheme of the small SCWR, known as CSR 150, has been completed. In addition, a reactivity control method has also been proposed. The thermal-hydraulics experimental database of SCW

has been established, including SCW and flow in multiple geometric channels and SCW critical flow. Two international benchmark studies on SCWR thermal-hydraulic characteristics were held virtually, respectively in April and December 2022. New alumina forming austenitic steels are developed, showing better corrosion resistance in SCWR operating conditions. A round-robin test was organized, focusing on obtaining reliable corrosion rate data and reaching agreement on corrosion mechanisms and materials testing standards.

- **Lead-cooled fast reactor (LFR)**: To explore advanced UN powder preparation process, a simulation experiment and theoretical calculation are being carried out. To improve corrosion resistance of 15-15Ti steel for cladding tube, the Grain Boundary Engineering technology is used. Steady-state and nature circulation experiments are carried out in the large pool-type lead-bismuth cooled non-nuclear reactor. Domestic technical standards for lead-based reactors are being developed.

Euratom

The Russian military aggression against Ukraine is an act of war and constitutes a violation of international law, undermining European and global security and stability. It is an attack on elementary values of freedom, democracy and self-determination, on which cultural expression, academic and scientific freedom, and scientific cooperation are based. The European Union stands by Ukraine and its people.

The current crisis is causing many member states to (re)consider their energy mix and its evolution in view of ensuring appropriate autonomy and independence of energy-related supplies. In some cases, this includes renewed interest in nuclear energy. The decisions on the national energy mix remain the competence and responsibility of each member state.

Complementary Climate Taxonomy Delegated Act

The Taxonomy Regulation is an essential part of the EU sustainable finance framework in the context of the transition towards climate neutrality. The regulation establishes conditions to be fulfilled by nuclear activities in order to comply with the taxonomy criteria. Its scope with respect to nuclear applications covers both existing and advanced systems. Time limits are set for the approval of new projects related to long-term operation (2040) and new-builds of Generation III+ (2045). Requirements and conditions for nuclear energy include adopting best-available safety technologies and having a geological repository for high-level waste and/or spent fuel in operation by 2050. The regulation emphasizes the important long-term role of Gen IV systems, indicating "closed fuel cycles or fuel self-breeding concepts ... that minimise the production of high-level radioactive waste" as relevant features on which research, development, demonstration and deployment are envisaged.

Euratom contributions

The Joint Research Centre (JRC) and Euratom member states' representatives proactively contribute to the six systems in steering committees, projects, working groups and task forces. Their contributions to the GIF consist of:

- **Indirect actions**, which are typically three- to four-year multi-partner projects co-funded by the European Commission's Directorate-General for Research and Innovation. Current indirect action projects cover Gen IV-relevant activities, including around EUR 100 million as Euratom research and innovation co-funding. The work programme for 2021-22 addresses, among others, safety of advanced and innovative nuclear designs, including SMRs, spent fuel multi-recycling, advanced structural materials, high-temperature gas-cooled reactor co-generation, digitalization, harmonization of licensing procedures, codes and standards for future fission and fusion plants. The work programme for 2023-25 was in an active development stage at the time of writing.

  A relevant field of growing interest is concerned with the availability, complementarity and integration of nuclear research infrastructures in support of the needs associated with advanced systems (project OFFERR) and of adequate capabilities and synergies towards a European nuclear education and training competence area (project ENEN2Plus). The NEA's contribution as an associated partner in these projects and the opportunities for synergies with corresponding initiatives co-ordinated by the NEA (FIDES, Nuclear Economics Support Tool) are also acknowledged. Ukraine's association agreement to the Euratom Programme is in force. Ukrainian representatives are to be invited to Programme Committee meetings with observer status.

- **Direct actions**, which correspond to parts of the JRC research programme; JRC mainly contributes along three dimensions:
  - coordination and management (including interfacing at various levels);
  - system design integration;
  - supply of data from experiments performed in JRC research facilities on inactive structural materials and active/irradiated fuels and compounds. The prioritization and reassessment of the JRC's activities and dedicated resources is reflected in the nuclear Direct Action Work Programme 2023-24, which was near finalization at the time of writing.

Initiatives and events

SMR. Several member states' active interest envisaging the construction of an SMR for power generation and non-electric applications continues. The concepts considered encompass both "light water" SMRs and advanced SMRs. The research supported by the Euratom Programme addresses in particular safety, security and safeguards (3S by design); decarbonization potential; non-electricity applications; and licensing and regulatory aspects.

FISA/EURADWASTE. The Euratom research and training conferences on fission safety of reactor systems (FISA 2022) and on radioactive waste management (EURADWASTE '22)[1] were held in Lyon, France, 30 May-3 June 2022. The results of almost 80 Euratom research and training projects were presented. The proceedings will be published in early 2023. Significant events completed the scene, including the SNETP 2022 Forum and the ENEN PhD Event & Prize.

The 15th **European Nuclear Energy Forum** was held in Prague, Czech Republic, on 10 and 11 November 2022. Around 200 participants discussed the future of nuclear energy in the European carbon-free power system and the opportunities and challenges of the nuclear sector.

France

Nuclear energy policy

In February, President Emmanuel Macron outlined the new French energy policy. The main objective is to reduce energy consumption while increasing carbon-free energy production capacity. Notably, in the next 30 years, the country must be able to produce up to 60% more electricity than today to replace part of the fossil fuels and cope with the planned oil and gas exit. The multi-pronged strategy involves both renewables and nuclear energy development, betting on these two pillars at the same time.

Two major decisions have been taken regarding nuclear energy. First, the operation of all existing reactors should be extended without compromising safety. The French nuclear operator, EDF, is therefore requested to study the conditions for extending the reactors' lifetime operation beyond 50 years. Second, President Macron announced the launch of a programme of new reactors, picking up the thread of the great French nuclear industrial adventure. Benefiting from the lessons learnt from the EPR, the new reactor design EPR2 is simplified and on-site building and mounting oriented. The whole programme consists of six EPR2s, which will be built two by two starting in 2028, and the studies on the location of eight additional units, which will come later. Building reactors by pair is a key factor to ensure cost control and on-time commissioning.

In addition, EUR 1 billion will be made available through the France 2030 reindustrialization plan. Two items are being addressed: 1) the NUWARD small reactor project detailed design; 2) innovative and advanced reactors, with an ongoing call for tender dedicated to new players so as to push new ideas forward, with the help of existing players in the background.

Considering the whole picture, this new programme on large nuclear power plants, SMRs and advanced reactors could lead to the commissioning of 25 GW of new nuclear capacity by 2050 in France.

In the wake of the announcement of this new energy policy, the French government has announced its objective to simplify and accelerate the administrative procedures that precede the first concrete casting of the foundations of the reactor and fuel buildings.

Innovation incentive programme

The government's plan for reindustrialization that is referred to as France 2030 dedicates EUR 1 billion for two nuclear innovation items: the NUWARD small reactor project detailed design and innovative and advanced reactors. For this second item, new players are invited to submit their proposals, pushing new ideas forward, with a private funding threshold. The French Alternative Energies and Atomic Energy Commission (CEA) working with emerging companies to assess their project and help them present their proposal.

Growing support to nuclear energy in France

According to an Ifop poll, three-quarters of the French population have a positive opinion of nuclear power, an unprecedented score. More than 80% of respondents consider nuclear power as a safe energy source, essential for the country's energy security. This opinion crosses the political spectrum. Nevertheless, beyond political consensus, those 65 years old support nuclear by more than 20 points compared to 18-24 year-olds, 84% versus 64% respectively. Finally, more than 65% of the French population is in favour of reviving the construction of nuclear power plants. This is up 14 points compared to 2021.

Nuclear power generation

In 2021, total power generation was up 4.5% in France compared to 2020, rising from 500 terawatt hours (TWh) to 523 TWh. Nuclear power amounts to 361 TWh, up 8% compared to 2020, accounting for 69% of the total power generation. Such figures remain below 2019 figures, which reveals the lingering effect of the COVID crisis on the reactor fleet maintenance management. The low-carbon electricity generation rate remains steady at 93%.

France faced an exceptional energy situation in 2022, which should persist in the beginning of 2023. Three main factors have triggered this situation. First, the gas crisis which appeared in the second half of 2021 with tensions on energy supply and demand following the post-COVID global economic recovery, then amplified by the Russian invasion in Ukraine. Second, the nuclear fleet availability has been 15 GW below the nominal situation, out 63.1 GW installed in France, resulting in a 25% drop of anticipated 2022 production compared to 2016-19 on average. This stems from the postponement of significant main-

1 Euratom conferences: www.sfen.org/evenement/fisa-2022-euradwaste-22.

tenance, linked to the ten-yearly inspections, due to the pandemic, and from an anomaly detected on some reactors, due to stress corrosion cracking, which has led to the shutdown of 12 reactors. Finally, a significant summer drought has resulted in relatively low hydraulic stocks. Despite this situation, tensions have not been to deplore on the electricity grid during the fall of 2022. In a context of massive energy price increases in Europe, the French government has been able to put in place a price cap called the "tariff shield". The competitiveness of nuclear power generation has made this decision possible.

Declaration of public utility for CIGEO

The declaration of public utility for the Cigéo project received a favourable response in July. This recognition of the public's interest of the project allows Andra to acquire the missing land by expropriating the land needed to build the storage centre, compensating farmers and carrying out preliminary work. This is a major milestone for the Cigéo project, which aims to store the most radioactive waste in a deep geological layer.

Highlights on the French R&D programme on advanced reactors

An outstanding R&D accomplishment is of note concerning SFR codification: the end a long-lasting thermal creep testing. A 9Cr steel sample has broken after 34 years under steady constraints at 500 MPa and 190°C, providing major data for codification for SFR steam generators. Such data are also of interest for other industrial or R&D communities, including the oil and gas industry, pipes and turbines suppliers, and fusion. Beyond scientific interest, such a result also demonstrates the capability of carrying out long-lasting, decades-long tests, keeping steady conditions despite a changing environment due to staff turnover, software replacement or the evolution of data treatment methods, for instance. Another test has been going on since 1987 (more than 35 years) at a higher temperature of 600°C and a lower pressure of 70 MPa.

Regarding MSR, R&D is growing and accelerating, with a partnership between several private and academic players, with a partial government funding. The Innovative System for Actinide Conversion project was displayed in the GIF MSR meeting in Toronto.

Japan

- The government has expressed plans for a "Carbon Neutral Policy": -46% in 2030 compared to 2013, and carbon-neutral in 2050
- 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 comment and others, Cabinet approved the Sixth Strategic Energy Plan on 22 October 2021.

For stable use of nuclear power

- Restart of operation with safety as a top priority: launch of restart acceleration task force; bringing human resources and knowledge together; and maintaining and improving technological capability.
- 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 have been made for the completion and operation of the Rokkasho Reprocessing Plant by a public-private partnership to get the engagement of the relevant municipalities and international society; and further promotion of plutonium-thermal (MOX fuelled) 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 for various challenges in proceeding with long-term operation with secured safety.
- Fulfilling conservation activities and considering various issues depending on the public and private sectors' respective roles.
- Public understanding: interactive dialogue including regions where electricity is consumed; and easy-to-understand public relations/hearings.

For promoting R&D

By 2030, while making the most of the private sector's knowledge and experience, the development of fast reactors will be steadily promoted through international cooperation; SMR technology will be demonstrated through international cooperation; and component technologies related to hydrogen production at high-temperature gas-cooled reactors will be established; R&D of nuclear fusion will be promoted through international collaboration such as the ITER Project.

- The Nuclear Energy X Innovation Promotion programme is ongoing.
- The private sector has proposed several types of reactors (LWR base SMR, SFR, HTGR, MSR). The government provides financial support and the Japan Atomic Energy Agency (JAEA) provides technical support.
- Prime Minister Kishida ordered a discussion of the development of next-generation nuclear power plants at the second meeting of Japan's GX (Green Transformation) Implementation Council on 24 August 2022. The government planned to discuss the topic by the end of 2022.

The Advanced Reactor Working Group made up of experts and others in the council is presently discussing nuclear energy policy. In July, the working group drafted a timetable for technological development, in which it stated that it would develop "innovative light water reactors" and the "small reactors" and "fast reactors".

In the GX Implementation Council, the Japanese government confirmed its policy to restart an additional seven nuclear reactors in 2023, in addition to the ten that have already been restarted:

- Light water reactors: 9 units restarted, 7 certified and 11 under examination on the new nuclear regulation.
- The HTTR, a 30 MW experimental high-temperature gas-cooled reactor (JAEA) restarted its operation on 30 July 2021 after being shut down following the great earthquake of 2011. While the HTTR itself was not severely damaged, regulatory requirements were enhanced in view of the lessons learnt from the accident at TEPCO's Fukushima Daiichi Nuclear Power Station. Following a safety review by the Nuclear Regulation Authority, which stated that the reactor has been in conformity with the new regulatory requirements for more than five years, 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 the NEA project. The JAEA also has a plan to conduct various tests to confirm safety, core physics and thermal-fluid characteristics, fuel performance. Furthermore, the demonstration plan of hydrogen production by the HTTR is under discussion.
- The experimental reactor, "Joyo", has been approved for development for large production of medical radioisotopes, which is seen as essential to Japan's growth strategy from the perspective of economic security. Additionally, a new "research reactor" is planned to be built at Tsuruga.

Korea

Nuclear power in Korea

Twenty-four nuclear power plants (21 PWRs and 3 CANDUs) were in operation in Korea as of October 2022. The installed nuclear capacity accounts for 17.3% (23 250 MWe) of the country's total capacity. The nuclear power plants provided 158 015 gigawatt hours (GWh) of electricity, corresponding to 27.4% of total domestic production in 2021. Two PWRs, Shin-Kori Units 5 and 6, are under construction. Shin-Hanul Units 1 and 2 are awaiting certification for commercial operation.

Nuclear energy policy

With the inauguration of the new government in May 2022, major changes were made to the national policy as it pertains to nuclear energy. The new government completely abandoned the previous nuclear plan to phase out nuclear power and announced a new policy to enhance the country's nuclear strategy. These major nuclear policy changes included extending the lifetimes of currently operating nuclear power plants and the aim to increase the number of nuclear power plants by 2030.

The government announced the draft version of the "10th National Plan for Electricity Supply and Demand" in August 2022. The plan contained Korea's electricity demand forecast and electricity supply plan, targeting the year 2036. The government proposed to resume the construction of new nuclear power plants (Shinhanul Units 3 and 4), which had been suspended during the previous regime. According to this plan, nuclear power generation in 2030 is expected reach 32.8% of Korea's total power generated.

Nuclear R&D prospect

The nuclear R&D policy will focus on future nuclear technologies, such as innovative SMRs, Gen IV reactor systems and nuclear-power-linked hydrogen production. The government is raising interest in the Gen IV SMR, which can be used for multiple purposes in a range of industrial fields. Additionally, the construction of the Gijang reactor (15 MWth) to produce medical radioactive isotopes started in August 2022.

Sodium-cooled fast reactor

With regard to SFR development, recent R&D activities have focused mainly on the development as an SFR-based power generation system (called SALUS), referring to spin-off technology from the Prototype Generation-IV Sodium Fast Reactor. The SALUS programme facilitates global exports in the SMR market. The electric power output of SALUS is 100 MWe, with a flexible operation and a long-life fuel cycle.

Very high-temperature gas-cooled reactor

The project to develop VHTR key technologies entitled "Development of Essential Technologies for Hydrogen Production Coupled with VHTR" was launched in April 2021. This project focuses on the development of coupled technologies between VHTR technology and the high-temperature steam electrolysis (HTSE) process. Hydrogen production testing will be completed with the 6 kWe HTSE module by 2024.

Molten-salt reactor

The Korea Atomic Energy Research Institute (KAERI) has initiated a new Gen IV reactor project for the molten-salt reactor (MSR)-based SMR. The objective of this project is to develop key technologies related to the MSR. The project recognizes the importance of cooperation with private companies to accelerate the demonstration of the MSR in the near future. The new MSR project will, therefore, be promoted through cooperation between KAERI and nuclear industries from its initiation.

Russian Federation

Russian nuclear power plants (affiliates of the Rosenergoatom Concern, the Electric Power Division of the State Corporation Rosatom) increased electricity generation by 2.02% over the first eight months of 2022 compared to the same period of the previous year.

Generation from January to August amounted to 145.86 billion kilowatt-hours. The operation of all Russian nuclear power plants for the first eight months of 2022 made it possible to prevent more than 73 million tonnes of CO_2-equivalent emissions of greenhouse gases into the atmosphere.

The share of electricity generation by nuclear power plants in Russia is 19%. In total, 37 power units are in operation at 11 nuclear power plants, with a total installed capacity of over 29.5 GW, including the floating power unit comprised of two reactor facilities.

The development of nuclear fuel for modernized floating power units (MFPU) intended for the Baimsky Mining and Processing Plant has begun. Each MFPU is supposed to be equipped with two RITM-200S reactors. To provide electricity to the Baimsky Mining and Processing Plant, Rosatom State Corporation proposed four MFPUs with two new RITM-200S reactors at each power unit with an installed capacity of at least 106 MW each. The commissioning of the first two power units scheduled by early 2027, the third by early 2028 and the fourth by early 2031.

Most of the MFPU equipment is similar to that of the Akademik Lomonosov floating nuclear power plant (FNPP). The core will be able to work up to five years without reloading. Unlike the FNPP, the MFPU is not equipped with reloading complex. There are fewer staff at the MFPU than at the FNPP.

The assembly works of the first RITM-200 reactor facility for the project 22220 nuclear icebreaker Chukotka began in 2022, as did the development of the technical design of the reactor plant and the main technological equipment for the first-of-a-kind small nuclear power plant based on the SHELF-M reactor, to be completed by 2024.

On 23 August 2022, a complex technological process was completed for the loading and installation of two steam-generating units of the RITM-200 reactor plant with a thermal capacity of 175 MW each at the universal nuclear icebreaker Yakutia.

As part of the implementation of the project for a new thermal reactor VVER-S, an R&D programme was established to justify technical solutions for a power unit of a nuclear power plant with a medium-power VVER.

Work on foreign projects is proceeding according to schedule, apart from the Hanhikivi-1 nuclear power plant, the contract for the construction of which was unilaterally terminated by Finland.

At power Unit No. 1 of the Akkuyu nuclear power plant in Türkiye, the dome of the inner containment was installed in the design position and thereby completed the formation of the sealing loop.

At power Unit No. 3 of the Kudankulam nuclear power plant in India, welding of the main circulation pipeline has begun.

As a result of the most recent nuclear fuel reloading, the entire core of the BN-800 reactor was fully converted to uranium-plutonium MOX fuel for the first time. First serial MOX fuel assemblies were loaded into the BN-800 core in January 2020. The first full reloading of the BN-800 with MOX fuel took place in January 2021, and over the next two reloadings, all fuel assemblies were gradually replaced with MOX fuel assemblies.

As part of the construction of a research nuclear installation based on the multi-purpose fast neutron research reactor (MBIR), work was carried out on the control assembly of a fully manufactured housing, acceptance and payment of equipment was carried out, construction work of two tanks of 2 000 m³ was completed. Work on the foundation of the turbine unit has been completed, as has the installation of fire extinguishing tanks; the thermal protection of the reactor has been installed.

The formation of the Advisory Board of the Consortium "IRC MBIR" (scientific advisory body) continues: its chairman has been appointed. The first meeting of the Advisory Board of the IRC MBIR was held on 12-13 July 2022 at the site of JSC "SSC NIIAR", thereby an international scientific platform was formed. Within the framework of the Advisory Board of the "IRC MBIR", five profile committees have been created:

- development of materials and fuel;
- safety of the use of nuclear technologies;
- validation of computer codes;
- non-energy application of nuclear technologies;
- closure of the nuclear fuel cycle.

The meeting was attended by 49 experts, 20 of whom were representatives of foreign states and international organizations.

Work is underway to create an MSR. The MSR will be a reactor with circulating fuel based on melts of metal fluoride salts. Structural materials of the MSR reactor plant with fuel salt based on the melt of lithium, sodium and potassium fluoride salts will be developed within the framework of the federal project "New Materials and Technologies" of the complex programme "Development of Equipment, Technologies and Scientific Research in the Field of Nuclear Energy Use in the Russian Federation for the Period Up to 2024" approved by the government. The detailed design of the MSR is expected to be completed by the end of 2025, and the reactor should enter into operation in 2028.

The construction of the lead-cooled fast reactor BREST-OD-300 is proceeding according to schedule. The first phase of the training and information centre (UTIC) of the experimental demonstration energy complex was put into operation. In 2024, it is planned to put a module into operation for the fabrication of mixed nitride uranium-plutonium nuclear fuel, which is part of the complex. Construction was completed of a unique stand to justify the main circulating pump unit of a fast neutron reactor with lead coolant.

A technical design of a sodium-cooled reactor with improved technical and economic characteristics (BN-1200M) was developed, including variants of core fuel elements with MNUP and MOX fuel.

Preparations have begun for the construction of power Unit No. 5 of the Beloyarsk nuclear power plant with a BN-1200M reactor. The assignment for the preparation of justifications for investments in the construction of the power unit was developed and approved, alongside with a road map for the construction of the unit. A license is expected to be obtained for construction in 2027 and complete the construction of the power unit by 2035.

Work on the repeated extension of the life until 2040 of power Unit No. 3 of the Beloyarsk nuclear power plant with the BN-600 reactor is in the final stage. It will be the only power unit in the world with a fast neutron reactor that will operate for 60 years.

South Africa

Nuclear New Build Programme

In August 2021, the National Electricity Regulator of South Africa (NERSA) issued a concurrence with the Minister of Mineral Resources and Energy under Section 34 of the Electricity Regulation Act of 2008 for 2 500 MW of nuclear energy to be procured. NERSA's agreement was accompanied by several suspensive conditions which are currently being addressed. These conditions require the Department of Mineral Resources and Energy to demonstrate the affordability, pace and scale at which the Nuclear New Build Programme would be implemented. Following the lifting of the suspensive conditions, the Department of Mineral Resources and Energy will be issuing a Request for Proposal for 2 500 MW of nuclear generation capacity.

Koeberg Long Term Operation: Eskom's implementation of the Koeberg Long Term Operation programme is continuing as planned and is subject to National Nuclear Regulatory guidelines and regulatory requirements. Koeberg Nuclear Power Station will reach its 40-year end-of-design life in 2024. To support safety case compilation, among others, in March 2022, Eskom and the IAEA completed a 50th SALTO Mission to assist in reviewing the long-term operational safety of the Koeberg nuclear power plant. In July 2022, Eskom submitted a Safety Case for Long Term Operation to the National Nuclear Regulator. The National Nuclear Regulator is currently reviewing the safety case and a final decision is expected in July 2024.

Multi-Purpose Reactor Project: In November 2022, Cabinet noted the completion of the Multi-Purpose Reactor project prefeasibility stage which is aimed at replacing the SAFARI-1 Reactor by 2030. Furthermore, the Cabinet favourably supported that the MPR Project be registered as a strategic integrated project. Benefits of registering the MPR project as a strategic integrated project include that the project regulatory approvals will be prioritized and funding will be unlocked. The project is at the feasibility stage, which is intended to be completed by March 2023. The feasibility stage will include site licensing, environmental impact assessment studies, concept design as well as communication and stakeholder engagement activities, which commenced in 2022. The IAEA conducted an Integrated Research Reactor Utilization Review for the neutron beamline facilities on the SAFARI-1 research reactor from 21 to 25 November 2022.

Centralised Interim Storage Facility Project: South Africa, through the National Radioactive Waste Disposal Institute and with oversight by the Ministerial Steering Committee, is implementing a Centralised Interim Storage Facility project for off-site storage of spent nuclear fuel. The pre-feasibility report for Centralised 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.

High-temperature research and development

The Integrated Resource Plan of 2019 advocates for SMRs and, further subject to Cabinet approval, discussions are ongoing to revive R&D of the High Temperature Reactor Programme based on the pebble-bed modular reactor technology and the work progressed by Eskom for the development of advanced high-temperature reactor technology.

Radioactive Waste Management Fund Bill: The Radioactive Waste Management Fund Bill is aimed to collect funds through the polluter-pays principle for the management of spent nuclear fuel and to provide for its governance and administration. Following Cabinet approval on 31 March 2022, the minister promulgated the publication of the Radioactive Waste Management Fund Bill for public comment for 60 days. Public comments were received and further consultations were conducted with other relevant stakeholders. The bill is currently being redrafted as part of consolidating public comments.

National Nuclear Regulator Amendment Bill: The Department of Mineral Resources and Energy engaged the National Economic Development and Labour Council, which published a report on its website once the consultations were concluded. The department is currently working on a process to submit the bill to Cabinet for tabling in parliament.

Switzerland

GIF activities

On the MSR research side, the major focus is on the safety and fuel cycle performance without particular preference for specific MSR concepts. However, the fast reactor option is considered as a reference system and chloride salt as the potentially most promising salt type for fuel cycle simplicity. In 2022, the core minimization study related to the breed-and-burn fuel cycle in molten chloride fast reactors was ongoing at PSI. The core performance was evaluated also from the perspective of several possible burnup definitions and actinides molar share evolution in the multi-fluid core layout.

The SFR research studies are also focusing on reactor safety aspects to define design improvements. A PhD study on advanced methodologies for modelling and assessing SFR safety functions was successfully completed in the framework of the EU ESFR-SMART project. The EU ESFR-SMART project was coordinated by PSI and devoted to the development and assessment of new safety measures of Generation IV European sodium fast reactors finished in August 2022, while a follow-up four-year project, ESFR-SIMPLE, coordinated by the CEA with the participation of PSI researchers and focusing in particular on metallic fuel option and small-power design development starts in October 2022.

On the material research side, the focus remains on the development of new analytical tools for high-temperature materials. A laser and IR-based equipment to measure the thermal conductivity of tubular SiC composite material samples has been constructed and is under testing. The corresponding ongoing PhD thesis deals with the micro/macro-structure analysis of the SiC composite, based on X-Ray tomography, and the connection to the measured conductivity values trough FEM models. In a synchrotron-based tomography, the resolution and contrast could be significantly increased, not only showing the pores, but also the fibres in the composite. With this, a complete model could be realized, containing the pores and fibres as simplified discrete objects. Finite element method-based conductivity calculations have been realized, and will further be refined. Another ongoing PhD thesis is investigating the basic mechanisms of irradiation-induced creep. These studies are based on diffusion coefficients in unstrained, strained and strained plus irradiated samples. The knowledge of the mechanism will lead to a better understanding of the materials behaviour under conditions relevant for Gen IV systems. A milestone has also been reached with the integration of PSI creep data into the *Materials Handbook*.

Politics and regulation

There is no relevant change in the Swiss government's strategy to reach net zero emissions in 2050. However, short term initiatives are being taken to deal with the possible energy shortage in the coming winters. This leads to more political and societal discussions on the use of nuclear power production in the future.

ENSI, the Swiss regulator, is continuing its systematic update of the guidelines.

Operation of the Swiss nuclear power plants and waste management

The dismantling of the Mühleberg/boiling water reactor is ongoing, according to plan. All reactors are in operation and operating at nominal power.

Nuclear power-related research in Switzerland

The focus of the NES Division is to make a strong contribution to the education of the next generation of nuclear experts; provide scientific support for the safe operation of light water reactors (LWRs); the delivery of the scientific basis for the assessment of the safety of deep geological repositories; and technology monitoring, including research work on Gen IV concepts.

The Swiss master course in nuclear engineering (ETHZ, EPFL and PSI) as well as participation in international research programmes (EURATOM, HORIZON Europe, NEA) is attracting an increasing number of students to the field.

United Kingdom

The British energy security strategy, published in April 2022, will see a significant acceleration of nuclear, with an ambition of up to 24 GW by 2050. This would represent up to around 25% of the United Kingdom's projected electricity demand. Subject to technology readiness from industry, SMRs will form a key part of the nuclear project pipeline.

In 2021, low-carbon sources generated 168 TWh of the United Kingdom's total 310 TWh electricity generated. Renewables accounted for 122 TWh and nuclear 46 TWh. These figures were lower than in 2020, driven by lower wind speeds, solar and hydro generation decreasing due to unfavourable weather conditions, and outages at all but one of the United Kingdom's nuclear power stations.

A new government body, Great British Nuclear, has been established to increase plans for the deployment of civil nuclear power up to 24 GW by 2050, to take two final investment decisions by 2030, and develop a resilient pipeline of new build projects. The government will back Great British Nuclear to get projects' investment ready and, as the owner of the new build programme, Great British Nuclear will have the accountability and capability to deliver long-term benefits, providing leadership to the nuclear system.

The UK government has launched the GBP 120 million Future Nuclear Enabling Fund to provide targeted support for new nuclear and make it easier for new companies to enter the market, including for both SMRs and advanced modular reactors (AMRs). The fund is the first in a series of interventions designed to achieve the government's ambition of deploying up to 24 GW of nuclear capacity by 2050.

The Future Nuclear Enabling Fund will help industry reduce project risks so they are better positioned for anticipated future investment decisions. The fund is targeted at applicants that could be able to take a final investment decision within the next parliament, subject to value for money and all relevant approvals.

The Nuclear Energy (Financing) Act received Royal Assent in April 2022. It will enable use of the regulated asset base funding model for new nuclear projects, which will unblock obstacles to developing these projects and cut the cost of financing them.

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

Since January 2021, the UK government has been in constructive negotiations on the Sizewell C project with EDF. In January 2022, GBP 100 million of funding was awarded to the Sizewell C developer to invest in the project to help bring it to maturity, attract investors and advance to the next phase of negotiations. In September 2022, the prime minister pledged GBP 700 million for the project.

The Nuclear Fuel Fund will provide up to GBP 75 million in grants to help preserve the UK front-end nuclear fuel cycle capability, in support of the British energy security strategy.

A request for information has asked industry to share views on the proposed fund design and pipeline of potential projects that might bid for funding.

The aim is to better understand:

- the barriers to investment in the UK front-end nuclear fuel cycle, supply chain and capability, which risk the government's energy policy objectives if not addressed – including, where available, evidence to support these views and international context;
- the potential projects that could be brought forward subject to these investment barriers being overcome or reduced, to preserve the capability;
- how the Nuclear Fuel Fund can best be designed and delivered to support the most promising project(s), maximize learnings and measure success.

UK GIF membership update

- The United Kingdom is continuing to present project proposals for engagement with the 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 co-ordinating this approval process and the hard work supporting the United Kingdom's wider engagement with the GIF's exciting agenda.
- The United Kingdom has joined the interim task force on non-electrical applications of nuclear heat engagement and looks forward to working with other task force members on taking this work forward across all six GIF technologies.

Regulation and Generic Design Assessment

- The Regulation and Generic Design Assessment (GDA) is the process carried out by the Office for Nuclear Regulation 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 Office for Nuclear Regulation 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.
- May 2021: the United Kingdom opened the GDA to advanced nuclear reactors.
- Rolls-Royce SMR Ltd. entered the GDA process with UK nuclear regulators on 7 March 2022. Since then, the UK nuclear regulators have announced the commencement of Step 1 of the GDA process for the Rolls-Royce SMR Ltd 470 MW SMR design. The design is expected to complete Step 1 of the GDA in April 2023 and Step 2 in August 2024.

UK Advanced Modular Reactor Research Development & Deployment Programme

- The Advanced Modular Reactor Research Development & Deployment programme aims to demonstrate high-temperature gas reactor (HTR) technology by the early 2030s, in time for any potential commercial AMRs to support net zero by 2050.
- This HTGR demonstration, which will be sited in the United Kingdom, should be shaped by end-user requirements, and should incentivize private investment in HTGRs by removing technical risk.

Phase A of the UK Advanced Modular Reactor Research Development & Deployment Programme

- The specific objectives for Phase A of the AMR RD&D programme are to understand the feasibility of technology solutions that can be developed within budget and identify the most cost-effective way to overcome market failures to enable HTGRs as an option to support net zero.
- On 2 September 2022, the UK government announced that GBP 3.3 million funding has been awarded for 6 pre-FEED studies.
- The following four organizations were awarded contracts to produce pre-front end engineering design (FEED) studies for developing advanced modular HTGR technologies as part of Phase A of the AMR RD&D Project:
  - EDF Energy Nuclear Generation Limited;
  - National Nuclear Laboratory;
  - U-Battery Developments Ltd;
  - Ultra-Safe Nuclear Corporation.
- The following two organizations were awarded contracts to produce pre-FEED studies for developing coated particle fuel for HTGR technologies as part of Phase A of the AMR RD&D Project:
  - National Nuclear Laboratory;
  - Springfields Fuels Limited;
- in addition, the Department for Business, Energy and Industrial Strategy is providing additional funding to UK regulators to develop their capability and consider innovative regulatory approaches to HTGRs.

Phase B of the UK Advanced Modular Reactor Research Development & Deployment Programme

- The Phase B competition launched on 13 December with the overarching aim of developing and demonstrating HTGR technology to enable the option for commercial HTGRs to potentially support the net zero target by 2050 and to develop UK-owned intellectual property.

- Phase B will award GBP 55 million to fund up to two different projects to develop detailed FEED+ studies, including overcoming key R&D.
- The specific objectives for Phase B are to secure a match-funded investment in HTGR technology by August 2023; develop at least one FEED in accordance with end-user requirements and a substantive market need; carry out associated enabling R&D; develop credible Phase C delivery plans; and develop skills and capability by March 2025.

Advanced Modular Reactor Research Knowledge Capture Project

- Separately, the Department for Business, Energy and Industrial Strategy is providing up to GBP 4 million funding for the Advanced Modular Reactor Knowledge Capture Project, as a complementary project to the AMR Research, Development and Demonstration Programme.
- The project seeks to facilitate knowledge capture and sharing to reduce the time, risk and cost of the AMR RD&D Programme delivery, and provide UK organizations with valuable knowledge to leverage against international funding.

United States

Nuclear energy continues to be a vital part of the energy development strategy to put the United States on a path to net zero carbon emissions by 2050.

Historic climate legislation

In 2022, President Biden signed the Inflation Reduction Act into law, which includes approximately USD 369 billion in climate provisions, making it the most significant piece of climate legislation in US history. Multiple incentives pave the way for the nuclear energy sector to help ensure energy security and reduce US emissions by 40% before the end of the decade. The Inflation Reduction Act includes several tax incentives for clean energy technologies, including advanced reactors. For more details see: www.energy.gov/ne/articles/inflation-reduction-act-keeps-momentum-building-nuclear-power.

Department of Energy efforts

The Office of Nuclear Energy and the Office of Clean Energy Demonstrations are leading the effort to move new and innovative advanced reactors from the conceptual and development stages into the commercial energy sector. The following summaries briefly highlight the Department of Energy (DOE)'s activities in 2022.

Advanced Reactor Demonstration Program

Recognising the importance of advanced reactors in meeting the United States' energy security and climate change goals, the Advanced Reactor Demonstration Program 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.

Advanced Reactor Demonstration Pathway

The goal of this pathway is to have two innovative reactor designs constructed and operated within a seven-year time frame. In FY 2022, responsibility for oversight and funding of these two major demonstration projects was moved to the Office of Clean Energy Demonstration per congressional direction provided in the Bipartisan Infrastructure Law, which included approximately USD 2.5 billion in outyear appropriations for the projects. The advanced reactor designs under this pathway are:

- *Natrium* by TerraPower, LLC: Natrium is a 345 MWe net plant, a sodium-cooled fast reactor that leverages decades of development and design undertaken by TerraPower and its partner, General Electric-Hitachi. In 2022, TerraPower progressed design maturity of all key facilities and completed site characterization. The awardee has also begun its safety analysis and held many pre-licensing engagements with the Nuclear Regulatory Commission.
- *Xe-100* by X Energy, LLC: The Xe-100 is a four-unit, 320 MWe net plant, a pebble-bed high-temperature gas-cooled reactor that leverages decades of public and private investment in the reactor design as well as the robust Tristructural-ISOtropic (TRISO) fuel form. In 2022, X-energy transitioned all reactor systems into final design, continued making excellent progress in licensing, including regular engagements with the regulator, and submitted the first ever US Category II fuel fabrication facility license application to the regulator. In its original application to DOE, X-energy had identified a preferred demonstration site on the DOE owned Hanford Reservation, but is now evaluating alternative sites.

Both projects are facing the challenge of High Assay Low Enriched Uranium (HALEU) feedstock availability that is likely to impact the already aggressive schedules established for the projects. The DOE is engaging with the domestic enrichment industry to accelerate the availability of HALEU for the needs of these demonstration reactors as well as subsequent advanced designs that are dependent on HALEU fuels.

Risk Reduction for Future Demonstration Pathway

The risk reduction projects are aiding advanced reactor developers in resolving technical, operational and regulatory challenges to enable future demonstration of a diverse set of advanced reactor designs. 2022 accomplishments for the five projects selected for award are:

- *Hermes Reduced-Scale Test Reactor* by Kairos Power, LLC: completed construction and hot argon commissioning of the engineering test unit;
- *eVinci™ Microreactor* by Westinghouse Electric Company, LLC: performed a feasibility study to identify how an eVinci reactor could be transported to a facility to be refuelled and refurbished to allow further use and enhance the economic competitiveness of the eVinci design;

- *BWXT Advanced Nuclear Reactor (BANR)* by BWXT Advanced Technologies, LLC: completed the production of natural uranium nitride kernels, including fabrication of the exact kernel design for BANR and development of the required nitriding process;
- *Holtec SMR-160 Reactor* by Holtec Government Services, LLC: completed Level 1 Probabilistic Safety Analysis, confirmation of the Nuclear Steam Supply System (NSSS) design with bounding safety analysis, and NSSS system piping and instrumentation diagrams;
- *Molten Chloride Reactor Experiment* by Southern Company Services Inc.: completed conceptual design of the Molten Chloride Reactor Experiment and the fuel salt synthesis line.

National Reactor Innovation Center

In 2022, the National Reactor Innovation Center initiated 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 builds by more than 10%. The project continues to progress well and achieved the milestone of 60% design completion in September 2022. Additionally, the National Reactor Innovation Center is establishing test beds for testing fuelled advanced reactor designs. The Demonstration and Operation of Microreactor Experiments (DOME) test bed will be capable of siting experiments to support microreactor technologies. The final design for DOME was completed in 2022 and construction activities will be initiated in 2023.

Advanced Reactor Concepts-20

The Office of Nuclear Energy is supporting three projects that enable the development of designs that could have a significant impact on the energy market in the mid-2030s or later. 2022 accomplishments include:

- *Inherently Safe Advanced SMR for American Nuclear Leadership* by Advanced Reactor Concepts Clean Energy, LLC: completed conceptual design, including all the associated system design description documentation;
- *Fast Modular Reactor Conceptual Design* by General Atomics: completed top-level design requirements;
- *Horizontal Compact High Temperature Gas Reactor* by Massachusetts Institute of Technology: completed parallel development of reactor physics models and core mechanical structures, as well as an economic model that has identified plant-level parameter requirements.

Microreactor Applications, Research, Validation, and Evaluation

The Office of Nuclear Energy is working to establish the MARVEL nuclear test bed at the Idaho National Laboratory. MARVEL will serve as a unique nuclear test platform to demonstrate microreactor operations and end-use applications and is planned for operation in 2024. Most recently, the MARVEL team completed the final design review for the project and successfully built a full-scale prototype to support the project.

Versatile Test Reactor Project

In 2022, the DOE issued the final environmental impact statement (FEIS) and record of decision (ROD) documenting the intention to build a sodium-cooled fast test reactor at Idaho National Laboratory. The FEIS and ROD were prepared in accordance with National Environmental Policy Act requirements to ensure that all environmental factors were considered. This was the first ever FEIS/ROD issued by the DOE to build a reactor.

Support for Clean Hydrogen Generation

The Office of Nuclear Energy, in collaboration with other DOE offices, has been working to further the integration of hydrogen production processes with nuclear power plants, including an industry funding opportunity for thermal system integration and end-use applications such as hydrogen trucks or synthetic fuels. In September 2022, the DOE released the Clean Regional Hydrogen Hub Funding Opportunity to provide USD 7 billion for at least four hydrogen hubs, including at least one nuclear hydrogen hub project.

A

Appendix 2. List of abbreviations and acronyms

3S	Safety, security and safeguards
AFA	Alumina forming austenitic
AFR	Advanced fast reactor
AI	Artificial intelligence
Al	Aluminum
ALLEGRO	European Gas Fast Reactor Demonstrator Project
AMR	Advanced modular reactor
ANSTO	Australian Nuclear Science and Technology Organisation
ANTSER	Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review
CEA	Alternative Energies and Atomic Energy Commission (France)
CD&BOP	Component design and balance-of-plant
CFD	Computational fluid dynamics
CFR	Chinese sodium fast reactor
CLEAR	China lead based reactor
CMVB	Computational methods validation and benchmarks
CNL	Canadian Nuclear Laboratories
CVR	Centrum výzkumu Řež (Czech Republic)
DOE	Department of Energy (United States)
dpa	Displacements per atom
ECC-SMART	Joint European Canadian Chinese – Small Modular Reactor Technology project
EDF	Électricité de France
EMAT	Electromagnetic acoustic transducer
EMWG	Economic Modelling Working Group
ENEA	National Agency for New Technologies (Italy)
ESFR	European sodium fast reactor
ESFR -SMART	European sodium fast reactor – Safety Measure Assessment and Research Tools
ESG	Environmental, social and governance
ETWG	Education and Training Working Group
EU	European Union
FFDL	Failed fuel detection and location
FFTF	Fast Flux Test Facility
FLiNaK	Salt mixture of lithium fluoride, sodium fluoride and potassium fluoride
Gen IV	Generation IV
GFR	Gas-cooled fast reactor
GIF	Generation IV International Forum
GW	Gigawatt
GWe	Gigawatt electrical
GWD/MTHM	Gigawatt days per metric tonne of heavy metal
GWt	Gigawatt thermal
HTR	High-temperature reactor
HTR-PM	High-temperature gas-cooled reactor –pebble-bed module
HTSE	High-temperature steam electrolysis
HTTR	High-temperature test reactor
IAEA	International Atomic Energy Agency
INET	Institute of Nuclear and New Energy Technology (China)

INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
ISAM	Integrated safety assessment methodology (GIF RSWG)
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre (Euratom)
JSFR	Japanese sodium-cooled fast reactor
K	Kelvin
KAERI	Korea Atomic Energy Research Institute
LFR	Lead-cooled fast reactor
LWR	Light water reactor
LBE	Lead-bismuth eutectic
M&S	Modelling and simulation
MOX	Mixed oxide (fuel)
MCFR	Molten chloride fast reactor
MSR	Molten salt reactor
MTA EK	Hungarian Academy of Sciences Center for Energy Research
MW	Megawatt
Mwe	Megawatt electrical
MWth	Megawatt thermal
MPa	Megapascal
NCBJ	National Centre for Nuclear Research (Poland)
NEA	Nuclear Energy Agency
NEaNH TF	Non-Electric Applications of Nuclear Heat Task Force
NRC	Nuclear Regulatory Commission (United States)
OECD	Organisation for Economic Co-operation and Development
PGSFR	Prototype Generation IV sodium-cooled fast reactor
PIE	Post-irradiation examination
PMB	Project Management Board
PRPPWG	Proliferation Resistance and Physical Protection Working Group
PSI	Paul Scherrer Institute (Switzerland)
pSSC	Provisional System Steering Committee
PT	Pressure tube
RANS	Reynolds-averaged Navier-Stokes (model)
R&D	Research and development
RD&D	Research, development and demonstration
RFBB	Rotational fuel-shuffling breed and burn fast reactor
RSWG	Risk and Safety Working Group
SAMOSAFER	Safety assessment for fluid-fuel energy reactors
SCWR	Supercritical water-cooled reactor
SCK CEN	Belgian Nuclear Research Centre
SDC	Safety design criteria
SDG	Safety design guidelines
SEM	Scanning electron microscopy
SFR	Sodium-cooled fast reactor
Si	Silicon
SIA	System integration and assessment
SIAP	Senior Industry Advisory Panel
SMR	Small modular reactor
SSC	System Steering Committee
STELLA-2	Large-scale Sodium Integral Effect Test Facility
TRISO	Tri-structural isotropic (nuclear fuel)
UOX	Uranium oxide
VHTR	Very high-temperature reactor
VUJE	Nuclear Power Plant Research Institute (Slovakia)
WEC	Westinghouse Electric Company

Appendix 3. Selection of GIF publications (2022)

GIF (2022), *Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review, Cost Reduction Strategy #2: Design – Modularity at Scale*, Generation IV International Forum, Paris.

Cipiti, B. et al. (2022), "Proliferation resistance and physical protection white papers of the six GIF Generation IV nuclear energy systems and cross-cutting topics: 2021-2022 update", *IAEA Symposium on International Safeguards: Reflecting on the Past and Anticipating the Future*, 31 October-4 November 2022, International Atomic Energy Agency, Vienna.

GIF PRPPWG (2022), "Bibliography compiled by the Proliferation Resistance and Physical Protection Working Group (PRPPWG)", Revision 09, GIF/PRPPWG/2022/001, Generation IV International Forum, Paris.

GIF PRPPWG and SCWR SSC (2022), "GIF supercritical water-cooled reactor proliferation resistance and physical protection white paper", GIF/PRPPWG/2022/002, Generation IV International Forum, Paris.

GIF PRPPWG and GFR SSC (2022), "GIF gas-cooled fast reactor proliferation resistance and physical protection white paper", GIF/PRPPWG/2022/003, Generation IV International Forum, Paris.

GIF PRPPWG and VHTR SSC (2022), "GIF very-high-temperature reactor proliferation resistance and physical protection white paper", GIF/PRPPWG/2022/005, Generation IV International Forum, Paris.

Ammirabile, L., T. Sofu and P. Gauthé (2022), *GIF Integrated Safety Assessment Methodology (ISAM) and Guidance for Its Implementation for Novel Advanced Reactors*, International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs, Vienna.

Futagami, S. (2022), *Development of Safety Design Criteria and Safety Design Guidelines for Generation IV Sodium-Cooled Fast Reactors*, International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs, Vienna.

GIF RSWG and GFR SSC (2022), *Gas Fast Reactor Safety Design Criteria*, GIF/GFR-SDC/2022/001, Generation IV International Forum, Paris.

GIF RSWG and GFR SSC (2022), *Gas Fast Reactor System Safety Assessment*, GIF/GFR-RSWG/2022/001, Generation IV International Forum, Paris.

Sofu, T. (2022), *GIF Framework for Risk-Informed Approach for Design and Licensing of Novel Advanced Reactors*, International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs, Vienna.

Paviet, P. (2022), "Pitch your Gen IV Research Competition – An initiative from the Generation IV International Forum", Paper No. 39512, ANS Winter Meeting, 13-17 November 2022, Phoenix, AZ, PNNL-SA-174164.

Paviet, P. and J. Sun (2022), "Generation IV International Forum Webinar Initiative", Paper No. 437, accepted for oral presentation at the International Conference on Fast Reactors and Related Fuel Cycles: Sustainable Clean Energy for the Future (FR22), Vienna, Austria, 19-22 April 2022, PNNL-SA-171792.

Workshop on Advanced Manufacturing, available at: www.gen-4.org/gif/jcms/c_115848/workshop-on-ad-vanced-manufacturing.

Alemberti, A. et al. (2022), "Status of Generation-IV lead fast reactor activities", International Conference on Fast Reactors and Related Fuel Cycles: Sustainable Clean Energy for the Future (FR22), Vienna, April.

Alemberti, A. et al. (2022), "System safety assessment of Generation-IV lead-cooled fast reactor systems", International Conference on Fast Reactors and Related Fuel Cycles: Sustainable Clean Energy for the Future (FR22), Vienna, April.

Serre, F. et al. (2022), "Reference Fuel Options for Generation-IV Sodium-cooled Fast Reactors", International Conference on Fast Reactors and Related Fuel Cycles: Sustainable Clean Energy for the Future (FR22), Vienna, April.

Yamano, H. et al. (2022), "Activities on the GIF safety and operations project of sodium-cooled fast reactor systems", International Conference on Fast Reactors and Related Fuel Cycles: Sustainable Clean Energy for the Future (FR22), Vienna, April.

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 make them available for industrial deployment by 2030. The GIF brings together 13 countries (Argentina, Australia, Brazil, Canada, the People's Republic of China, France, Japan, Korea, the Russian Federation, South Africa, Switzerland, the United Kingdom and the United States), as well as Euratom – representing the 27 European Union members and the United Kingdom − to co-ordinate research and development on these systems. The GIF has selected six reactor technologies for further research and development: 1) the gas-cooled fast reactor; 2) the lead-cooled fast reactor; 3) the molten salt reactor; 4) the sodium-cooled fast reactor; 5) the supercritical water-cooled reactor; and 6) the very high-temperature reactor.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 34 countries: Argentina, Australia, Austria, Belgium, Bulgaria, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Korea, Romania, Russia (suspended), the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Türkiye, the United Kingdom and the United States. The European Commission and the International Atomic Energy Agency also take part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes;
- to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management and decommissioning, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

The Nuclear Energy Agency serves as Technical Secretariat to GIF.

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