GEN IV International Forum

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ANNUAL REPORT 2023

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Foreword from the Chair

The Generation IV International Forum (GIF) 2023 Annual Report highlights the achievements and progress of research and development collaborations for Generation IV (Gen IV) reactor systems.

The year 2023 has been productive for the GIF, with many accomplishments in the form of publications, webinars and outreach, in addition to the technical progress made on the six GIF systems. I particularly enjoyed our in-person Policy and Experts Group meetings that provided much-needed interactions and time for side conversations. Thank you to France for hosting the April meeting in Lyon.

The development of a new framework agreement to continue the work of the GIF among mutually willing parties began in 2023 and will come into effect from February 2025. It is important that the GIF continues to provide global leadership and support through the new framework to ensure that safe, secure and sustainable nuclear energy is available for the future.

The Advanced Manufacturing and Materials Engineering (AMME) Task Force was approved to transition to a working group (AMME-WG) in October 2023, with a leadership role in the manufacturing and materials community. AMME-WG objectives are to promote the use of advanced manufacturing and materials engineering technology, thus reducing time to deployment for advanced reactor systems. Specifically, the AMME-WG aims to promote international collaboration on the qualification of advanced materials and manufacturing processes for use in Gen IV reactors. We are excited to learn about their future accomplishments in this important research area.

We continue to look for opportunities to increase our engagement with industry and vendors to accelerate the demonstration of Gen IV systems. Opportunities for direct industry contributions or data sharing are being pursued in GIF technical projects. With the increased interest in advanced nuclear energy systems and crosscutting applications to benefit from low-carbon electricity and heat to meet global carbon reduction goals, the GIF Task Force on Non-Electric Applications of Nuclear Heat (NEANH) plans to organize a second NEANH workshop in 2024 with broad industry participation. The GIF also continues to work with the GIF Senior Industry Advisory Panel to focus on the transition of research and development to demonstration and commercial deployment.

The second Pitch your Gen IV Research competition was conducted to increase junior researchers' engagement within the Gen IV community and stimulate their interest in advanced reactor systems. Forty-seven abstracts were submitted by participants from 11 different countries. Eleven selected participants produced short, informative videos, on which over 400 members of the public voted. The first, second and third place winners were invited to present a GIF webinar in 2024.

As we move forward, 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 the commercial deployment of advanced systems. I look forward to another successful year!

With best wishes,

Alice Caponiti
GIF Chair

A tribute

With deep sorrow, the Nuclear Power Plant Research Institute of Slovakia (VUJE, a.s.) conveyed the passing of Dr. Branislav Hatala, the Director of the Nuclear Safety, Research, and Development Division. Dr. Hatala was a longstanding member of the Generation IV International Forum (GIF), serving as the chair of the Gas-cooled Fast Reactor System (GFR) Steering Committee and making significant contributions to the development of the ALLEGRO GFR demonstrator.

Dr. Hatala joined VUJE, a.s., in 1995 and contributed to safety assessment of VVER-440 units operated in Slovakia. Thermo-hydraulic analyses and the theory of thermomechanical behavior of nuclear fuel were truly his passion. Notably, his early work earned him the Award of the Minister of Education of the Slovak Republic for Science and Technology, recognizing his significant achievements in safety analysis methodology. Under his leadership, his team was honored as the “Scientific and Technical Team of the Year” in 2017 for their groundbreaking work on the ALLEGRO reactor. Branislav was a highly valued contributor to GIF and the nuclear community for almost 29 years. He will truly be missed and always remembered.

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1 GIF membership and organization highlights

The Generation IV International Forum (GIF) brings together 13 countries and Euratom (representing the 27 European Union members) to coordinate research and development on advanced nuclear energy systems. The structure and organization of the GIF remained the same in 2023.

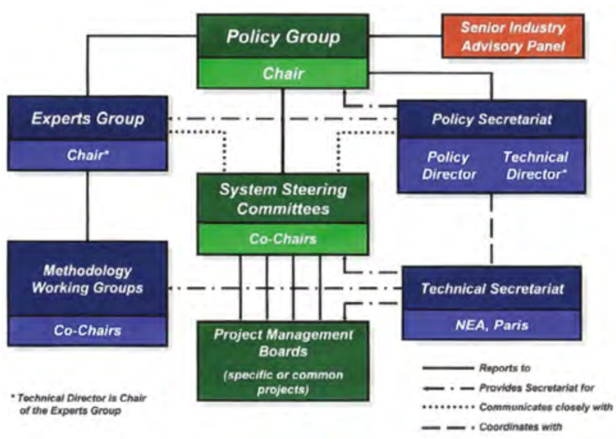
Figure 1-1: GIF membership

This chapter will focus only on major changes that occurred in 2023. More detailed information on the GIF membership and organization can be found on the GIF website (www.gen-4.org/gif/jcms/c_59452/governance-stucture.)

GIF organization

The global governance of the GIF and its main bodies are summarized in Figure 1-2.

Figure 1-2: GIF governance



To ensure continuity of operational requirements within the GIF Technical Secretariat, several staff changes occurred at the Nuclear Energy Agency (NEA) in 2023; the resulting structure as of October 2023 is shown in Figure 1-3.

Throughout 2023, GIF members made substantive progress in their work and increased engagement with industry. The NEA continued to provide support to the GIF technical bodies in charge of developing the different systems, as well as to the methodology working groups, the Senior Industry Advisory Panel and task forces. The NEA organized several GIF meetings in 2023, including the 50th Experts Group and 56th Policy Group meetings in Versailles, France, in October 2023. It also initiated efforts to revamp the GIF public website to improve performance and usability and to increase awareness and transparency with industry and the broader private sector.

GIF Framework Agreement

The existing GIF Framework Agreement expires on 28 February 2025. The GIF Policy Group agreed in 2023 to develop a new framework agreement for those parties mutually willing to continue collaborations. The Policy Group is working closely with the GIF Technical Secretariat to produce an updated draft framework agreement for consideration by GIF members in 2024.

GIF-International Atomic Energy Agency (IAEA) engagement

The GIF and the IAEA have recently enhanced their method of collaboration. This important interaction, which has been ongoing since the first annual interface meeting in 2003, allows the two organizations to exchange information, progress, status and future plans with respect to Generation IV (Gen IV) technologies. A new approach was taken at the 17th GIF-IAEA Interface Meeting in 2023 with updates on ten topic areas covering the six reactor technologies and several crosscutting or enabling topics. There are also more regular updates to the collaboration matrix document and focused discussions ("deep dives") held on specific topics. In 2023, deep dive discussions were held on safety, advanced manufacturing, digitalization and qualification.

Strategic priorities, perspectives and objectives

In 2023, the GIF Technical Secretariat pursued its efforts to improve GIF communications and outreach to industry, researchers, policy makers, government stakeholders and the general public so as to reinforce the GIF's position as a leading international collaborative organization with technical expertise focused entirely on Gen IV nuclear energy systems.

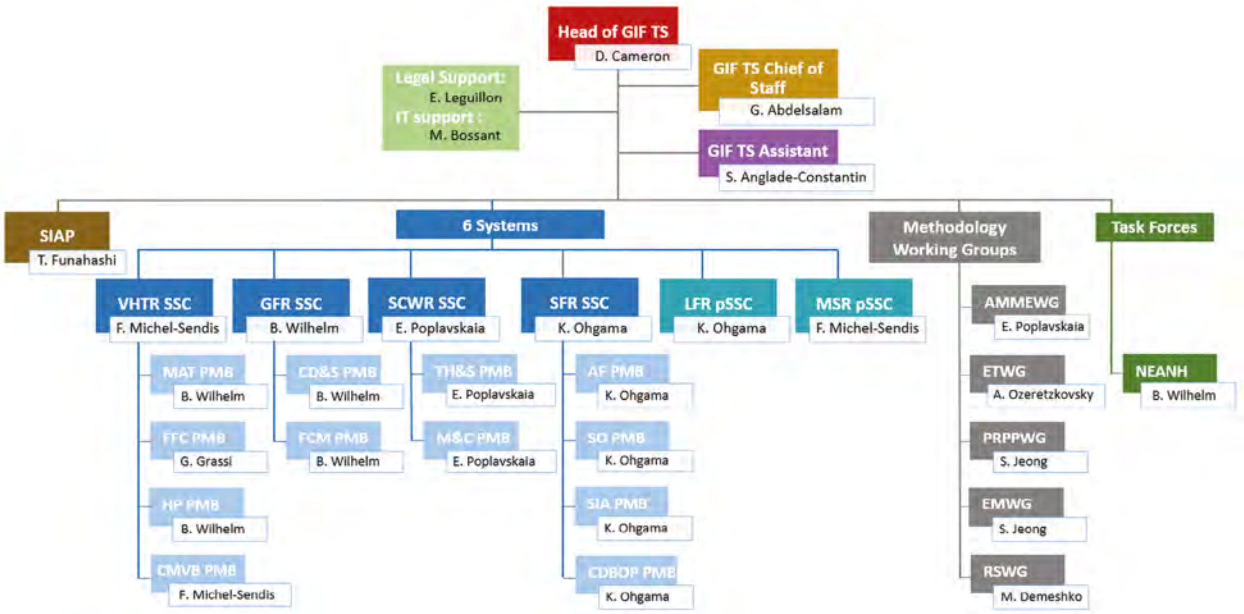


Figure 1.3. Structure of the GIF Technical Secretariat as of October 2023

In this context, and following the very successful GIF forum with industry partners held in October 2022 in Toronto, Canada, the GIF Senior Industry Advisory Panel (SIAP) launched a new initiative to organize SIAP special sessions with industry to showcase some key GIF products that could be useful to industry and Gen IV technology developers. A special small modular reactor (SMR) session was held virtually in July 2023 to provide an overview of the models and guidelines developed by the Economic Modelling Working Group (EMWG) for estimating levelized unit electricity costs and total capital investment costs for advanced reactor designs. Another SIAP session was organized with industry partners in the margins of the World Nuclear Exhibition in Paris in November 2023 to promote the use of advanced manufacturing and materials engineering technology to reduce the time to deployment of advanced reactor systems.

The 2024 perspectives and objectives are to continue to assume and improve GIF Technical Secretariat services by:

- Providing assistance in reviewing or renewing GIF contracts and agreements such as the GIF Framework Agreement, system and project arrangements, and memoranda of understanding.
- 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.
- Pursuing efforts to revamp the GIF public website to enhance its performance and usability, improve collaboration between members, and enhance brand awareness.
- Developing a special project to create a web-based searchable inventory of GIF information assets (e.g. videos, databases, expert contacts and intellectual property), and how to access them. This inventory should increase utility and impact to industry and the broader private sector, and support efforts to accelerate the demonstration and deployment of Gen IV reactor concepts.
- Maintaining coordination and cooperation with other international organizations active in the field of Gen IV nuclear reactors (e.g. NEA, IAEA, World Nuclear Association).

Diane Cameron
Head of the GIF Technical Secretariat

Fiona Rayment
GIF Policy Director

2 GIF outlook and current initiatives

The original GIF Roadmap (GIF, 2002) identified three successive phases for advanced reactor development: 1) the viability phase; 2) the performance phase; and 3 the demonstration phase. Today, several Gen IV technologies are entering the third phase, with "demonstration reactors" being designed, constructed and operated by GIF members. The demonstration goals depend on the maturity of both the underlying reactor technology and the design features utilized by a specific reactor design. A 2017 study (DOE, 2017) observed a stepwise process for reactor technology deployment; this historic four-step pattern is summarised below and shown in Figure 2-1:

1. Research and development to prove the scientific feasibility of key features.
2. Engineering demonstration at reduced scale for proof of concept.
3. Performance demonstration to confirm effective scale-up and validate integral behavior.
4. Commercial demonstration of a specific design that will be replicated for deployment.

In general, the capabilities needed for research and development (R&D) remain useful for all technology demonstration steps (e.g. fixing technical problems, resolving regulatory questions, optimizing cost and performance of key components). In the demonstration phase, technical challenges and benefits are informed by implementation and operational experience; thus, the technology development tasks to deploy a Gen IV design concept become more specific and prioritized in the latter demonstration steps. The challenge is to sustain an advanced reactor infrastructure (both facilities and experts) to effectively continue Gen IV technology development throughout this demonstration phase.

GIF collaborations target the continuation and effective utilization of international R&D infrastructure for advanced reactor technology. This chapter highlights the enabling role of GIF initiatives on knowledge management, industry engagement and innovation framework.

Knowledge management

GIF members include the original creators, technology expertise leaders, and state-of-the-art operators for each of the Gen IV technology options. Preservation of up-to-date technical knowledge is an important goal in each of the national programs, with technical approaches and lessons learned shared among GIF members. For example, a recent GIF task force documented the status and need for international advanced reactor testing capabilities (GIF, 2021). In addition, the results of key technology testing are being shared in GIF technical projects as either joint databases, benchmarks for validation and/or specific technical contributions.

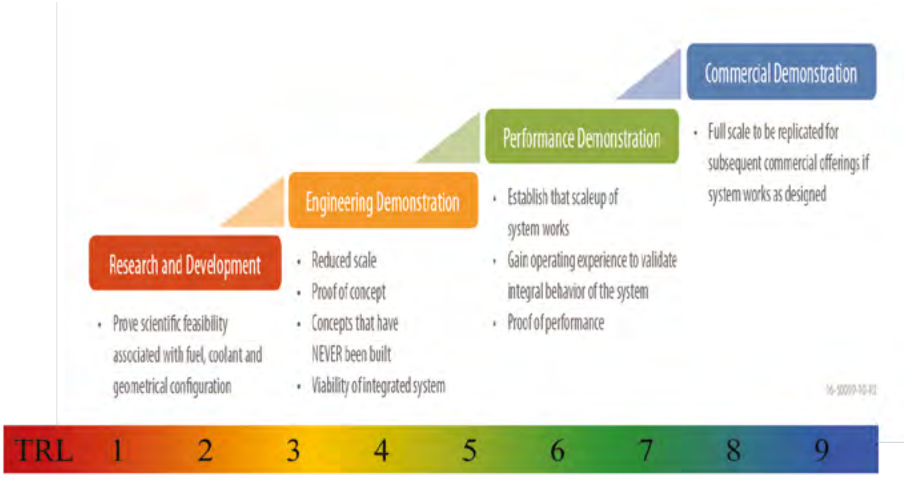
Educating both the public and nuclear community on GIF technologies is a key for knowledge transfer. The Education and Training Working Group (ETWG) has conducted a webinar series with expert presentations on all GIF technology options and other technical issues (GIF, 2023). It has also engaged with nuclear educators to promote and recognize advanced reactor R&D efforts.

Current GIF collaboration projects are a convenient pathway for transferring expertise to the broader nuclear community and the next generation of Gen IV technology innovators. GIF collaborations allow regular interactions with international technology experts through the project management boards and working groups. These R&D collaborations will continue to focus on key issues for Gen IV licensing and performance improvement, with specific challenges clarified and resolved in the demonstration phase.

Industry engagement

With emerging Gen IV demonstration reactors and technology deployment being conducted by the private sector, 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 GIF collaborations. Feedback from the Gen IV 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 the sharing of design/methods. To explore opportunities to streamline GIF mechanisms and accelerate GIF collaboration with industry, four specific collaboration opportunities were initiated in 2022:

1. Industry involvement in GIF task forces. Pursue nomination of industry experts as contributing members. Continue to actively invite private sector participants to GIF organized workshops and other mechanisms.
2. Industry input solicited for the Safety, Security and Safeguards (3S) project methodology being pursued jointly by the GIF Proliferation Resistance and Physical Protection Working Group (PRPPWG) and the Risk and Safety Working Group (RSWG).



Source: Petti et al. (2017).

Figure 2-1: Reactor technology demonstration steps based on international experience

3. Sharing of materials handbook data from the very high-temperature reactor materials project. Procedures for third-party sharing are being clarified by the project management board (PMB) and will be implemented in 2024.
4. With two private industry members, a provisional PMB negotiated a project plan for a new GIF sodium fast reactor (SFR) technical project on sodium thermal fluid dynamic validation; the new project arrangement was submitted to the GIF SFR System Steering Committee in September 2023 and is currently in the approval process.

Innovation framework

To provide opportunities for future innovation, a versatile R&D infrastructure must be sustained. In early development phases, technology is developed in anticipation of technical and performance challenges. In the demonstration phase, technology development is informed by facility construction and operating experience. However, this situation does not preclude the need for future technology innovations, which may be crucial for optimizing performance to facilitate the deployment and sustainability of Gen IV technology. The emergence of technical solutions in related fields and/or operational experience in new regimes should also be expected. For example, existing light water reactor (LWR) performance continues to be improved by power uprates, improved maintenance and new fuel testing.

GIF collaborations include focused consideration of new technology options, such as the Advanced Manufacturing and Materials Engineering Working Group (AMME-WG), and new operation paradigms, such as the Non-Electric Applications of Nuclear Heat (NEANH) Task Force. R&D on innovative technology features and extended operating regimes is also shared in system technical projects.

In summary, the R&D infrastructure (both expertise and facilities) used to develop advanced reactor technology remains vital for Gen IV demonstration and deployment. In the demonstration phase, technology development will be informed and prioritized by implementation and operations experience. Therefore, knowledge management and industry engagement are important to guide and prioritize GIF collaborations. The international R&D community also needs to retain both deep expertise and robust facilities to provide an open and active ecosystem for future technology innovations.

Robert Hill
GIF Technical Director

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3 System summaries

This chapter summarizes the 2023 accomplishments of GIF collaborations for the six reactor technologies identified in the GIF Roadmap: gas-cooled fast reactor, lead-cooled fast reactor, molten salt reactor, supercritical water-cooled reactor, sodium-cooled fast reactor and very high-temperature reactor.

Gas-cooled fast reactor

Signatories of the System Arrangement for collaboration on gas-cooled fast reactor (GFR) R&D are GIF members Euratom, France and Japan. Two technical projects have been established for GIF collaborations:

1) GFR conceptual design and safety: Members are the Joint Research Centre (JRC), French Alternative Energies and Atomic Energy Commission (CEA).
2) GFR fuel, core materials and fuel cycle: Members are the JRC, CEA and Kyoto University.

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 modeling.

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

The reference concept for the Gen IV GFR is a 2 400 MWt (megawatt thermal) plant with a break-even 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-fueled pins contained within a ceramic hextube. The current favored material for the pin clad and hextubes is silicon carbide fiber reinforced silicon carbide. 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. 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 being the use of a closed cycle gas turbine in the GFR.

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

The objective of high fuel burnup, 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 plutonium inventory in the GFR core is not much higher than 15 tons per GWe (gigawatt electric).

ALLEGRO demonstration project overview

The objectives of ALLEGRO are to demonstrate the viability of the GFR and to qualify specific GFR technologies such as fuel, fuel elements, helium-related technologies, and specific safety systems, in particular the decay heat removal (DHR) function. The project aims to 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 MWt ALLEGRO.

The original design of ALLEGRO consists of two helium primary circuits and three DHR loops integrated in a pressurized cylindrical guard vessel (Figure GFR-1). The two secondary gas circuits are connected to gas-air heat exchangers. The ALLEGRO reactor could function as a demonstration reactor hosting GFR technological experiments, and as a test pad for using the high-temperature coolant of the reactor in a heat exchanger for generating process heat for industrial applications. It could also be used as a research facility which, thanks to the fast neutron spectrum, makes it attractive for fuel and material development and testing of some special devices or other research works.

The 75 MWt reactor will be operated with two different cores. The starting core with enriched uranium or mixed oxide (MOX) fuel in stainless steel claddings (Figure GFR-2) 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 carbide fuel and will enable ALLEGRO to operate at the high target temperature.

Four nuclear research institutes and companies in the Visegrad-Four region (ÚJV Řež, a.s., Czechia; MTA EK, Hungary; NCBJ, Poland; and VUJE, a.s., Slovak Republic) have decided to start joint preparations aimed 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 and will bring its knowledge and experience to building and operating experimental reactors, in particular fast reactors.

To study safety and design issues, as well as medium and long-term governance and financial issues, the four aforementioned organizations created a legal entity in July 2013, 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 representing the project internationally.

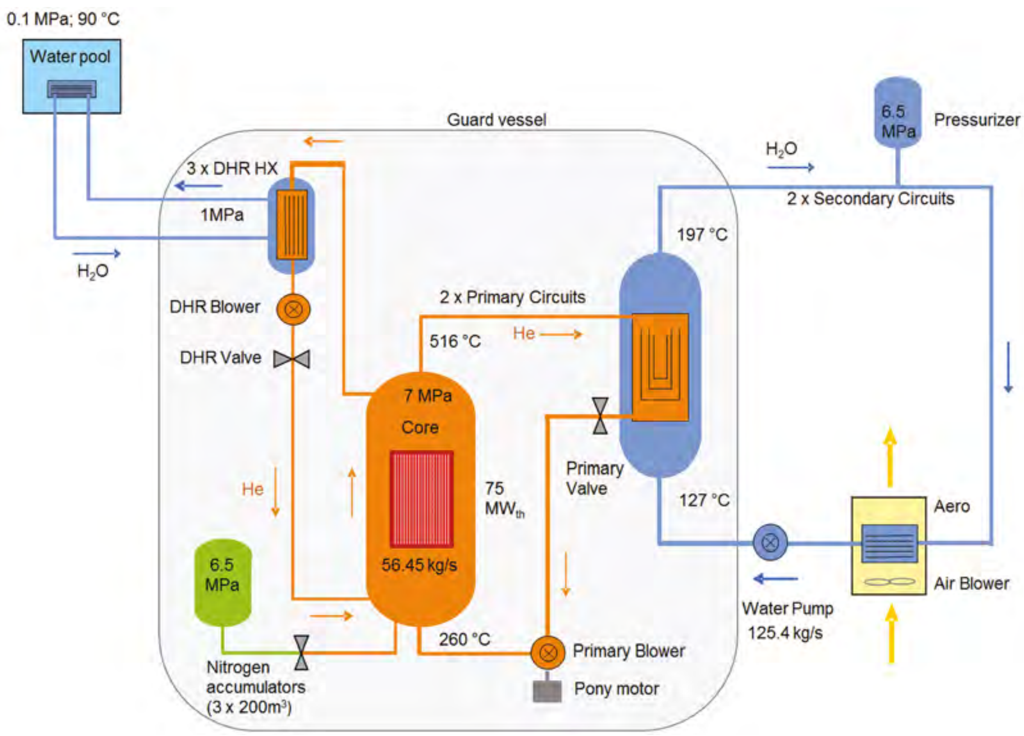
As a result of the preparatory work, it has transpired that certain safety and design issues remain unsolved, and in several aspects a new ALLEGRO design must be elaborated.

R&D objectives and technology innovations

All members of the GFR System Arrangement (Euratom, France and Japan) are active contributors to the SafeG project (www.safeg.eu). 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 SafeG project is managed by VUJE, a.s. (Slovak Republic) and involves 15 associates from eight countries. Top-rated research organizations and universities participate in the project. SafeG design is divided into six technical packages and one coordination work package.

The project design aims to answer the principal safety issues of the GFR concept and introduce the key safety systems of the ALLEGRO reactor. An

Figure GFR-1: ALLEGRO original design



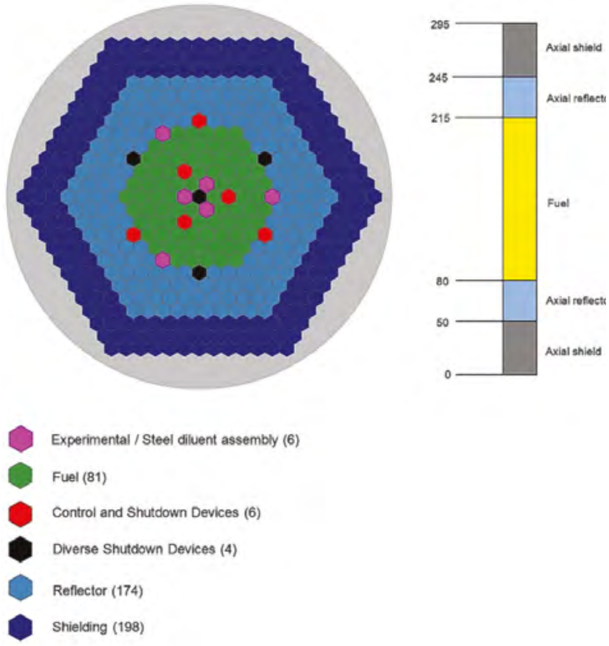
Source: Hatala et al. (2019).

important part of the design is to acquire new experimental data using recent research implements and special computational programs to carry out safety analyses drawn upon the study of relevant physical phenomena.

The design follows the long-term experience and extensive knowledge of experts involved in the research and development of nuclear energy issues, in particular those connected with designing and implementing the simulation means, licensing the nuclear equipment, and material science and reactor safety. The results achieved through the process of building the ALLEGRO demonstrator will be utilized for the SafeG design procedure, chiefly those focused on GFR advanced technology, along with the results of previous scientific programs developed within the European framework: the ALLEGRO Implementing Advanced Nuclear Fuel Cycle and the Visegard Initiative for Nuclear Cooperation.

Integration of the latest technology into the GFR concept is an ultimate ambition of the SafeG concept. These innovations will be demonstrated mainly through ALLEGRO, which is one of the four referential demonstration units of the Gen IV reactors supported by the European Sustainable Nuclear Industry Initiative (ESNII). The SafeG objectives can be grouped into four pillars:

1) Completing the ALLEGRO demonstrator safety concept:
   - Reactor core optimization on account of neutron-physics, thermo-hydraulics and thermo-mechanics for the first (starting) zone, and for further fuel loads.
   - Design of diverse ways of controlling the fission chain reaction and reactor shutdown.
   - Design of entirely passive systems for the removal of decay heat, with passive DHR tested using up-to-date equipment that is part of GFR technology.
2) Upgrading the ALLEGRO demonstrator design and concept GFR with prime materials and technologies, i.e. innovative fuel clad materials based on silicon carbide composition, and construction materials capable of withstanding extreme heat used for the primary system and safety-related systems. Experiments will result in data on the performance of these materials in diverse conditions at high temperatures.



Source: Hatala et al. (2019).

Figure GFR-2: First ALLEGRO reactor core

3) Linking national research activities and creating an integrated platform to share knowledge and achievements and to coordinate activities to spread new ideas and findings to the scientific community for their worldwide implementation.
4) Deepening cooperation between Europe and Japan in GFR research through sharing knowledge about advanced high heat resistant materials for fuel rod claddings and other primary system components.

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

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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. It also 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 goal areas of the GIF: sustainability, economics, safety and reliability, and proliferation resistance and physical protection. 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 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 (ELFR), intended for central station power generation; 2) a 300 MWe system of intermediate size – the Russian BREST-OD-300; and 3) a small, transportable system of 10-100 MWe size – the US small secure transportable autonomous reactor (SSTAR), which features a very long core life (see Figure LFR-1 and Table LFR-1). The expected secondary cycle efficiency of each LFR system is at or above 42%. GIF LFR systems thus cover the full range of power levels, from small and intermediate to large sizes. There are important synergies 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

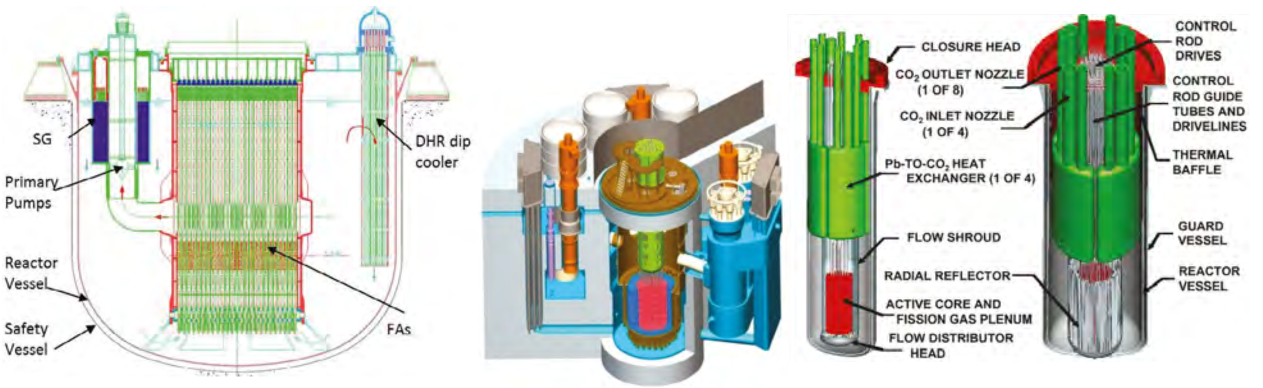
The LFR System Research Plan developed within the GIF is based on the use of molten lead as the reference coolant and lead-bismuth eutectic as the back-up. Given R&D needs for fuel, materials and corrosion-erosion control, the LFR system is expected to require two-step industrial deployment. In the first step, reactors operating at relatively modest primary coolant temperatures and power densities would be deployed by 2030, with higher performance reactors deployed by 2040 as the second step. Following the reformulation of the GIF LFR pSSC in 2012, the System Research Plan was completely revised.

Table LFR-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 temp. (°C)	400	420	420
Core outlet temp. (°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 temp. (°C)	335	340	402
Turbine inlet temp. (°C)	450	505	553

Note: MWt: megawatt thermal; MWe: megawatt electrical; T: temperature.

Figure LFR-1: GIF lead-cooled fast reactor reference systems



Note: The European Lead Fast Reactor (left), the BREST-OD-300 reactor (middle) and the SSTAR reactor (right).
Source: Alemberti, A. et al. (2018).

Technical highlights – Provisional System Steering Committee activities

The pSSC has prepared the following reports since 2020:

- LFR System Safety Assessment, issued in June 2020;
- GIF LFR: Proliferation Resistance and Physical Protection white paper, issued in October 2021;
- Safety Design Criteria (SDC) for Generation IV LFR System, issued in March 2021.

In 2022, the IAEA and the NEA Working Group on the Safety of Advanced Reactors provided comments on the SDC report, which are currently being analyzed.

Two meetings of the GIF LFR pSSC took place in 2023. The first was hosted by newcleo in Turin, Italy on 28-30 March. The second was held at the Seoul National University on 4-6 October. These meetings featured presentations on the status of member activities with descriptions of their respective national LFR programs, information sharing with participants from the industry, and discussions of the LFR SDC report.

National LFR demonstration and development highlights

In Russia, the BREST-OD-300 innovative LFR is being developed as a pilot demonstration prototype for base-type commercial reactor facilities of the future nuclear power industry with closed nuclear fuel cycle (Lemekhov, Moiseyev et al., 2023; Lemekhov, Cherepnin, et al. 2023). The complete detailed design of the BREST-OD-300 reactor has been carried out, and the construction license from Rostekhnadzor was issued in February 2021. The construction of the nuclear power plant with the BREST-OD-300 LFR began on 8 June 2021 (see Figure LFR-2) and continued in 2022-2023 in accordance with the adopted road map.

Figure LFR-2: Construction of the BREST-OD-300 power unit as part of the Pilot Demonstration Energy Complex

Source: Rosatom.

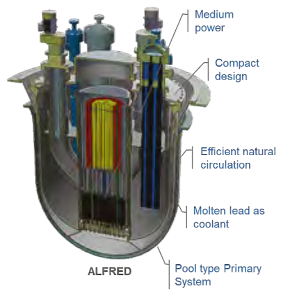
Major experiments are being completed in parallel with the construction of the BREST power unit. Full-scale modeling of the BREST-OD-300 core was finalized in 2023 on the large critical stand BFS-2, with the main core parameters experimentally justified (Borovskaya et al., 2023). BREST fuel element mock-ups were tested in the BN-600 power reactor and the BOR-60 research reactor. In total, more than 1 500 fuel elements were irradiated and the design burnup level for initial core loading was justified, with a burnup of 9% heavy atom achieved at a damaging dose of 112 displacements per atom (dpa). In the big lead acceptance test stand the preliminary testing program was completed for the full-scale prototype of the BREST main coolant pump. A prototype of a protective chamber was created to study the technology of BREST spent fuel reprocessing.

The BREST-OD-300 reactor is being created as one of the most important components of the Pilot Demonstration Energy Complex. It operates in a closed nuclear fuel cycle, 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 program at the reactor. Various studies and life tests of components and equipment with irradiation experiments in both lead coolant and a fast neutron spectrum environment will be carried out.

In Europe, activities related to LFR mainly focus on five projects:

1. The development of a small modular reactor (SMR) LFR. SCK CEN (Belgium), ANSALDO NUCLEARE & ENEA (Italy), RATEN-ICN (Romania) and Westinghouse (US) have signed an agreement to speed up the industrial deployment of SMR LFR technology.
2. R&D activities for the construction of an LFR demonstrator in Romania, the Advanced Lead Fast Reactor European Demonstrator project (ALFRED), supported by the FALCON Consortium (ANSALDO NUCLEARE, ENEA and RATEN-ICN as tier-1 members). The largest worldwide pool-type facility, ATHENA, aimed at developing LFR technologies in the frame of ALFRED has been realized and commissioned in Mioveni (Romania), supported by the FALCON Consortium (see Figures LFR-3 and LFR-4) (Caramello et al., 2023; Ciolo Puviani et al., 2023).
3. R&D activities carried out in the United Kingdom in collaboration with several EU organizations in the framework of the development of the Westinghouse LFR concept (Lee et al., 2023a).
4. R&D activities for the construction of an LFR demonstrator in Sweden (Oskarshamn), 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 (LFR-AS-30 and LFR-AS-300). newcleo is investing up to EUR 50 million to buildup large-scale research infrastructure in collaboration with the ENEA Brasimone Research Centre (Italy). newcleo is also working to set up a MOX fuel factory in Europe, fully funded by private investments, to complement its commercial LFR program (Meli, 2023).

Figure LFR-3: ALFRED cutaway view



Source: Courtesy of the FALCON Consortium.

Figure LFR-4: ATHENA facility main components

Source: Courtesy of Ansaldo Nucleare.

In parallel, several ongoing European collaborative projects (Euratom co-funded initiatives such as ANSELMUS, INNUMAT, PASCAL, PATRICIA, PUMMA, TANDEM and HARMONISE) are dedicated to heavy liquid metal technology, development and validation of numerical tools and safety assessments, and 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 (Lorusso, et al, 2023).

In Japan, reactor design studies have been performed by the Tokyo Institute of Technology regarding the feasibility of breed-and-burn fast reactors using lead coolant or lead-bismuth eutectic coolant. The breed-and-burn concepts use natural uranium or depleted uranium converting the fertile fuel in the core to achieve high burnup without reprocessing. The analyses are being performed for a lead-cooled nitride fuel rotational fuel-shuffling breed-and-burn fast reactor and a lead-bismuth eutectic cooled metallic fuel rotational fuel-shuffling breed-and-burn fast reactor. The current designs are mainly for small reactors with reactor power of about 750 MWt.

In Korea, research on LFR commenced in 1997, primarily led by universities. Following the Fukushima accident, the program shifted focus towards developing an SMR, evolving into the URANUS[1] design and later refining into MicroURANUS for maritime use. Subsequently, a start-up MicroURANUS Corporation emerged from a university-industry consortium, assuming leadership to standardize the design and pursue licensing. With the latest resolution prioritizing net zero operations in international shipping by 2050, nuclear ship propulsion became one of primary development targets. Collaboration with the university consortium continues and actively involves research groups from Ulsan National Institute of Science and Technology, Korean Advanced Institute of Science and Technology, Pusan National University, and Seoul National University, alongside industrial contributions.

MicroURANUS's conceptual design as tailored for maritime applications is refined without altering key parameters. It has a 20-30 MWe power rating, lead-bismuth eutectic coolant and a minimum 40-year service life. Notably, for maritime use it is confirmed that transportation and installation can be accomplished with solidified coolant within the MicroURANUS reactor vessel. Emphasis of safety assurance was placed on practically eliminating severe accidents. Assessments explored the risk of steam explosions, considering scenarios such as steam generator tube ruptures and nuclear ships sinking into deep, high-pressure water. LFR designs showed significantly lower steam explosion risks compared to other proposed maritime reactors. The existing MicroURANUS design adequately addresses potential steam explosion scenarios, minimizing the need for major design modifications. Continuous material development and testing maintain sufficient design margins, accounting for corrosion and void swelling.

In the United States, work on LFR concepts and technology has been carried out since 1997. Recent projects include:

- Corrosion/irradiation testing in lead and lead-bismuth eutectic, led by the Massachusetts Institute of Technology.
- Developing a versatile liquid lead testing facility, testing material corrosion behavior and developing ultrasound imaging technology in liquid lead, University of Pittsburgh.
- Addressing critical issues with the compatibility and chemical interactions of uranium nitride fuel, alumina-forming austenitic alloys, and lead coolants and sublayers, Rensselaer Polytechnic Institute.

The Department of Energy (DOE) also initiated two Technology Commercialization Fund projects to support the adaptation of the fast reactor analysis software developed at national laboratories for industry needs: safety analysis code, SAS4A/SASSYS-1, improvements and enhanced neutronic analysis software for Westinghouse LFR Design and Modeling, both with the Argonne National Laboratory and Westinghouse as an industry partner (Lee et al., 2023b).

In the industrial sector, Westinghouse is continuing development of its 450 MWe LFR. In 2023 this included both design/analysis activities and the setup of multiple test facilities to demonstrate key LFR systems, components, materials and phenomena. These facilities include a Westinghouse-funded test rig being installed at Virginia Tech University for

1. Ubiquitous Robot Accident Forgiving Nonproliferating Ultra-lasting Sustainer.

measuring the radioisotopes retention capability of liquid lead, and eight test rigs in the United Kingdom (at Westinghouse-Springfields and partner sites) as part of a project led by Westinghouse and co-funded by the UK government.

The Chinese government has provided continuous national support to develop lead-based reactor technology since 1986 through the Ministry of Science and Technology, the National Science Foundation, and China's 13th and 14th Five-Year Plan. The China Lead-based Reactor (CLEAR) was selected as the reference reactor for the accelerator-driven system (ADS) project, as well as for the technology development of the Gen IV LFR. A 10 megawatt (MW)-grade CLEAR-M10 project aimed 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 at the International Academy of Neutron Science. A series of verification experiments for T H characteristics and prototype components are being performed in CLEAR-M0, with a loss of flow transient experiment carried out in 2023 (Bai et al., 2023; Qifan et al., 2023).

For the ADS system, several concepts and related technologies are under assessment. For example, the detailed conceptual design of CLEAR for minor actinide transmutation and traveling-wave reactor for energy production has been completed. The China initiative ADS project, led by the Chinese Academy of Sciences in collaboration with the China Institute of Atomic Energy and other industrial organizations, aims to build a 10 MWt subcritical 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, and civil construction began in 2022.

In recent years, other organizations have started paying more attention to LFR development. Lanzhou University is building multifunctional T-H experimental platforms for floating small LFR verification. Xi an Jiaotong University built experimental platforms and performed experiments for safety characteristic research and code verification. In addition, the China National Nuclear Corporation and several universities, such as Shanghai Jiaotong University, Harbin Engineering University and Zhejiang University, have been carrying out LFR-related R&D.

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

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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. China, Japan, and Korea are observers. The mission of the MSR provisional System Steering Committee (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 2023, the pSSC met once online (34th meeting) and once in person (35th meeting). Based on a decision from the 52nd Policy Group meeting, the GIF MSR group will continue as a pSSC and will evaluate its collaboration status again in two years. In 2023, the GIF Proliferation Resistance and Physical Protection white paper on MSRs was published, and efforts were continuing to focus cooperation within the new system research plan for the MSR pSSC around three main areas: 1) salt behavior; 2) materials properties; and 3) system integration.

Main characteristics of the system

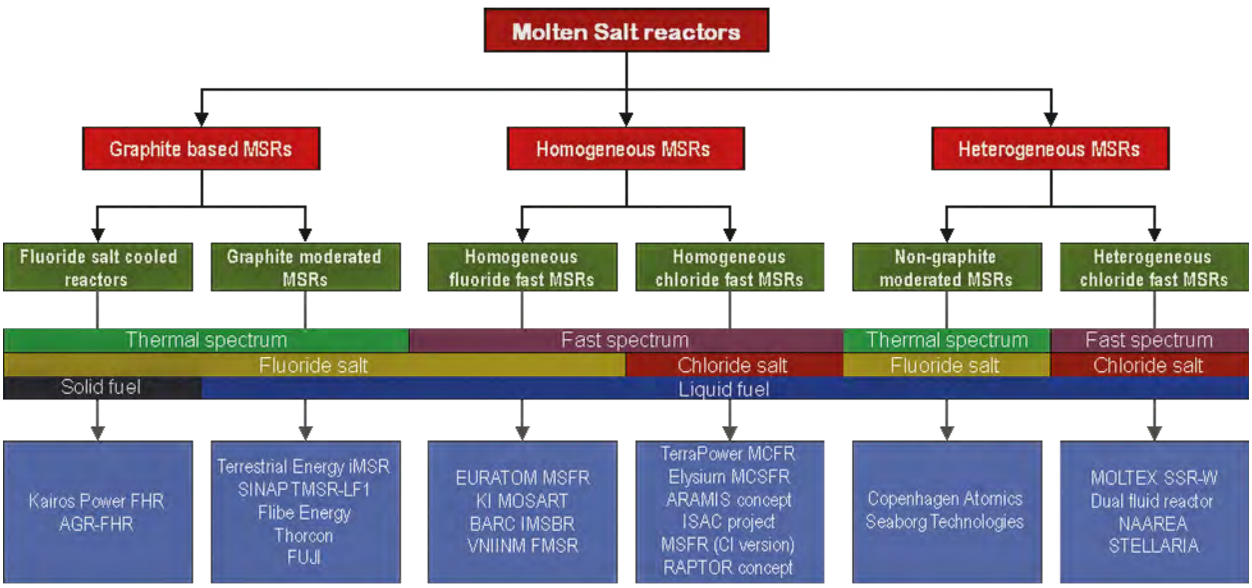
An MSR is any reactor where a molten salt has a prominent role in the reactor core (i.e. fuel, coolant and/or moderator). Liquid-fuel MSRs are a type of nuclear fission reactor in which a halide salt serves as the nuclear fuel and potentially also the coolant. In solid-fuel MSRs, the halide salt serves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

Both solid- and liquid-fueled MSRs have seen a resurgence in interest over the past two decades. Proposed designs using molten salt fluoride and chloride salt mixtures include both thermal and fast spectrum systems, as well as designs with time and spatially varying spectra. Nearly every form of fertile and fissile material is being considered for its potential 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 MSR-1; the Figure also shows concept types for prominent MSR developers.

MSRs have several advantageous characteristics, ranging from high-temperature operation (and consequent increased thermodynamic efficiency) to low-pressure operation that reduces 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.

However, the distribution of radionuclides by liquid fuels can necessitate fully remote maintenance. Furthermore, to prevent pressurization, the gaseous and volatile fission products need to be managed on site, unlike solid-fuel reactors where they are trapped in claddings. Molten salt can also become highly corrosive if exposed to oxidative impurities. Overall, MSRs have substantial technology differences from

Figure MSR-1: Classification of MSR types and key developer concepts



Note: MSR: molten salt reactor.

both existing light water reactors (LWRs) and other advanced reactor concepts, necessitating different approaches to safety assessment, safeguards and operations.

R&D objectives

The GIF's objective is to support collaboration on technology, data, and analysis methods for Gen IV nuclear systems. 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. Potentially useful collaborative projects include:

- salt fabrication and measurement of thermochemical and thermophysical 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 (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-fueled reactors.

National MSR demonstration and development highlights

The Australian Nuclear Science and Technology Organization (ANSTO), in collaboration with the Idaho National Laboratory (INL), is engaged in research on the use of graphite and carbon-carbon (C/C) composites in MSR systems. ANSTO has also formed partnerships with domestic universities to advance arc-based directed energy deposition technology for the production of nuclear engineering components. Qiu et al. (2023) has investigated the effects of heat input on the microstructure of Hastelloy C276 during the cold metal transfer, based on the directed energy deposition process.

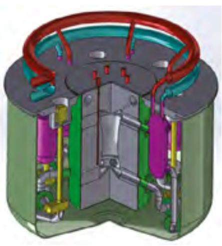
MSR research in the European Union is supported through research programs in the JRCs and through the Euratom Research and Training Programme, which is a complementary funding program to Horizon 2020 and Horizon Europe that covers nuclear research and innovation.

At the JRC Karlsruhe (Germany), R&D efforts in 2023 included the development of a method to synthesize high-purity plutonium (III) chloride ($PuCl_3$). The resulting material was used to conduct a series of experiments to measure the thermophysical properties of the most important salt compositions for the chlorine-based fast spectrum MSR being investigated in Europe. The focus was on salt mixtures of $PuCl_3$ with sodium chloride (NaCl) and magnesium chloride ($MgCl_2$). Vapor pressure, melting point, enthalpy of fusion and density were measured. The investigations of fluoride-based MSR fuels were continued, with the focus on the so-called FUNaK system, a mixture of sodium fluoride (NaF), potassium fluoride (KF) and uranium tetrafluoride (UF_4). After a comprehensive systematic investigation of the NaF-KF-UF_4 phase diagram, two eutectic compositions were experimentally identified and the thermophysical properties such as density, melting point, melting enthalpy, evaporation and boiling point were determined. Selected fluoride-based systems were also examined for the influence of the fission product on their properties.

The thermodynamic database for the main MSR fuel and coolant systems has been further developed and is one of the most important ongoing activities. It has been improved with new experimental data on some of the binary and ternary systems, and expanded to include a number of systems containing corrosion and fission products. Selected systems have been modeled to predict viscosity and density behavior. Recently, a Thermodynamics of Molten Salts collaboration was started between the JRC Karlsruhe, the Orano Group, the CEA and Delft University of Technology to jointly develop a database through both experiments and simulations. Discussions have been

Figure MSR-2: Main characteristics and schematic view of the ARAMIS-P concept

Power	300 MWt
Salt	NaCl-$MgCl_2$-(Am,Pu)Cl_3
Core volumic power	250 MW/m^3
Fuel loop number	6
Total fuel salt volume	5.2 m^3
Heat exchanger type	Shell and tube
Core volume ratio	25%



Note: MWt: megawatt thermal; NaCl: sodium chloride; $MgCl_2$: magnesium chloride; Am: americium; Pu: plutonium; MW: megawatt.
Source: Pascal et al. (2023).

initiated with the US Department of Energy (DOE) and Canadian Nuclear Laboratories (CNL) to join the project.

In Euratom, the EU Horizon 2020 project, Severe Accident Modeling and Safety Assessment for Fluid-fuel Energy Reactors (SAMOSAFER), was completed after four years. The project was dedicated to developing simulation models and tools validated with experiments and complemented with the design and demonstration of new safety barriers. The Horizon Europe project MIMOSA, which aims to conduct feasibility studies on fast spectrum chloride MSR as a concept for the incineration of actinides from spent MOX fuel, entered its second year in 2023.

Some specific findings from the SAMOSAFER project include:

- Transient analysis has been performed on the fluoride salt and chloride salt designs to investigate, among other things, the effects of compressibility of the fuel salt and the damping effect of the Doppler feedback during transients. The compressibility is a very important phenomenon to include to properly calculate the maximum fuel temperature during reactivity induced transients.
- The source term in the molten salt fast reactor (MSFR) has been evaluated and simulation tools have been benchmarked by inter-code comparisons. Subsequently, decay heat production and radiotoxicity in various compartments of the reactor system (including fuel treatment unit) have been evaluated. As foreseen, these depend heavily on the removal rates in the various processes (bubbling, fluorination, etc).
- Coupled neutronics-CFD (computational fluid dynamics) codes have been developed and compared to simulate bubble injection and transport, and associated reactor power fluctuations. Numerical safety assessment studies have been conducted to assess, and where possible reduce, the power fluctuations in the core.
- Decay heat removal (DHR) in the emergency draining system is very sensitive to the geometry and the material properties. Radiative transport plays a prominent role to reduce the maximal temperature and to arrive at realistic designs. Experimental setups and numerical studies have been used to investigate the freezing and remelting of fluids against cold walls.
- The possibilities and effects of natural circulation have been investigated numerically and experimentally in small (Rayleigh Bernard cell) and large facilities (e-Dynasty).
- For the control and monitoring of the MSFR, predictive control strategies, incident detection methods and a classification methodology have been developed.

In France, R&D programs around MSR initiated in recent years by the French National Centre for Scientific Research (CNRS), and more recently by the CEA, were continued and developed. Since 2020, the CEA, supported by Orano, has developed an MSR program around the Advanced Reactor for Actinides Management in Salt (ARAMIS) concept (Figure MSR-2). The goal of the program is to study the opportunity of fast chloride MSRs with regards to plutonium management (Pascal et al., 2023). It was decided to focus this new program on chloride MSRs where the fast neutron spectrum is well adapted to actinide transmutation processes, and where the salt chemistry is compatible with current the hydro-recycling process in France. However, the experimental background on such technology is poor and requires further R&D. The CEA focused its program on reactor design, neutronics studies, salt depletion evaluation, code development and first experiments on active salt synthesis and material corrosion studies induced by inactive chloride salt. During this period, the CNRS, supported again by Orano, also contributed to neutronic studies around actinide management solutions based on fast chloride MSR, considering both burner and breeder options.

Since 2022, the CEA and the CNRS, along with industrial partners (Orano, FRAMATOME, Électricité de France [EDF]) have been contributing to the French common project, the Innovative System for Actinides Conversion (ISAC), 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 (MSFR versions); 2) plutonium (Pu) burner (ARAMIS-P); and 3) minor actinide transmuter (ARAMIS-A in ISAC project). These programs 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 behavior, corrosion, fission product management, salt polishing, salt synthesis [$PuCl_3$], scenario studies) and the refueling and polishing strategy;
- multiphysics and chemistry modeling and simulation (neutronic/thermal-hydraulic simulations, coupling between salt depletion and thermochemical calculations, etc.).

The ISAC project is divided into five work packages: 1) sketch design; 2) scenario studies; 3) active salt chemistry; 4) material corrosion and irradiation; and 5) experimental facilities. Around 14 PhD students are funded by the ISAC project on various activities (simulation activities, experimental activities). A first batch of students will finish in 2024 (reactor control, neutron and thermal-hydraulic coupling). The first specification of the ARAMIS-A type reactor was completed in 2023, which will serve as a basis to propose a first sketch of the minor actinide incinerator (300 MWt, outlet fuel salt temperature up to 650°C). Experimental studies on salt component synthesis – $PuCl_3$, americium III chloride ($AmCl_3$) – and full mixed salt were conducted in 2023 (CEA, JRC), as were experimental screenings of material corrosion in inactive chloride salts ($NaCl-MgCl_2$) (CEA, CNRS). Chloride salt loops have been under design and construction since 2023 in the CEA and CNRS. The start of the first chloride loop is planned in 2024 with $NaCl-MgCl_2$.

The French company NAAREA (Nuclear Abundant Affordable Resourceful Energy for All) continues its development program around a micro chloride fast reactor based on a silicon carbide (SiC) matrix, with two main milestones in 2023: 1) official French state support in the framework of the France 2030 investment plan; and 2) the start of a first chloride loop at 700°C in NaCl-$MgCl_2$ built-in SiC. STELLARIA, created in 2023, continues its development program around small modular fast chloride reactors, and has completed preliminary sketches of the concept.

In Russia, Rosatom continued to support preliminary tests of 10 MWt MSR design development with homogeneous core coupled with fuel salt clean-up unit at the Mining and Chemical Combine site (Zheleznogorsk). The aim of this testing is to demonstrate the control of the reactor and fuel salt management with different long-lived actinide loadings, drain-out, shut down, etc. The 10 MWt test MSR design utilizes high nickel alloy as the containment vessel and for other metallic parts of the system. Fuel salt will leave the reactor vessel at temperatures up to 700°C and energy will be transferred to a coolant LiF-BeF eutectic salt.

The first stage of the lithium, beryllium, plutonium fluoride fuel salt physics experiment is aimed at collecting data on phenomena characteristics including: 1) handling of fuel salt (synthesis, melting, management and disposal); 2) reaching criticality of the liquid-fuel system, start-up, accurate reactivity control and shutdown; 3) control and stability of a core with liquid fuel and low effective fraction of delayed neutrons; 4) uncertainty of criticality (k-eff) due to nuclear data and thermophysical properties of the fuel salt; and 5) material behavior and components under high-temperature conditions. R&D efforts in 2023 were focused on:

- consideration of three stage construction concept for 10 MWt test MSR with processing unit;
- development and verification of multiphysics code for 10 MWt test MSR (Gasta et al., 2023);
- measurement of physical and chemical properties of fuel and coolant salts;
- development of technologies for preparing, controlling and maintaining the quality of fluoride salts (Zaikov et al., 2023);
- development of technologies for fuel salt processing in the 10 MWt test MSR;
- consideration of technologies for handling nuclear waste generated during operation of the 10 MWt test MSR and processing unit;
- justification of design components operating with fuel salt and cover gas in the 10 MWt test MSR;
- consideration of the tritium removal system design for the 10 MWt test MSR;
- development of a methodology for predicting the corrosion characteristics of Ni-Mo-Cr (nickel-molybdenum-chromium) alloys for the 10 MWt test MSR (Polovov et al., 2023);
- non-isothermal dynamic laboratory and in-reactor corrosion tests for metallic alloys with fuel/coolant salts (Table MSR-1) (Surenkov et al., 2023).

Table MSR-1: Fuel coolant salts being tested in Russia

Place	Salt	Alloy	Tmax, (°C)	Time, hours	Loop type*
MCC	Li,Be,Pu/F	SS316	650	1 000	NCL
KI	Li,Be,Ce,U/F	Ni-Mo-Cr	700	1 000	NCL
RIAR	Li,Be,Pu/F	Ni-Mo-Cr	700	1 000	RCL
IHTE	Li,Be/F	Ni-Mo-Cr	700	1 000	FCL
KI	Li,Be,Ce,U/F	Mo-base	700	1 000	NCL
MCC	Bi-Li	Mo-base	700	1 000	NCL

*NCL – natural convection loop, FCL – forced convection loop.
Note: Li: lithium; Be: beryllium; Pu: plutonium; Ce: cerium; F: fluorine; Ni: nickel; Mo: molybdenum; Cr: chromium.

In In the United States, a number of salt-cooled and salt-fueled MSR R&D activities were performed in 2023. The US DOE Office of Nuclear Energy supported the continued progress of MSR development projects. Southern Company Services and TerraPower began salt operations at their integrated effects test facility (Southern Company, 2023). Development of their Molten Chloride Reactor Experiment (MCRE) is also proceeding. The US DOE issued a finding of "no significant impact" for the environmental assessment of MCRE (DOE, 2023). Kairos Power was granted a construction permit for its Hermes non-power reactor (WNN, 2023) and applied for a construction permit for its dual-unit low-power test reactor Hermes-2 (NRC, 2023). Abilene Christian University completed the construction of the Gayle and Max Dillard Science and Engineering Research Center (Abilene, 2024), which is intended to house a research reactor and start operations as soon as 2026.

Terrestrial Energy USA continues to file multiple documents (notably principal design criteria and a methodology for calculating an off-gas source term) with the US Nuclear Regulatory Commission (NRC) in pursuit of standard design approval of its integral molten salt reactor (NRC, 2024). Metatomic was awarded a project to work with Savannah River National Lab to develop technology to convert used LWR fuel to chloride salt MSR fuel (Metatomic, 2023). Alpha Tech Research was also awarded a project to work with the Argonne National Laboratory to validate yttrium hydride as a moderator for MSRs (Alpha Tech, 2023). In response to the limited current supplies of high-assay low-enriched uranium, ThorCon released a description of its modified system design that employs low-enrichment uranium fuel.

Fuel salt is the defining element of liquid-fueled 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 R&D emphasis. Six US national laboratories are developing fundamental fuel salt property data to enable stakeholders to make informed decisions. Updated versions of two databases, Molten Salt Thermal Properties Database–Thermochemical (MSTDB-TC) and Molten Salt Thermal Properties Database–Thermophysical (MSTDB-TC), are now available for public use. The updated databases both increase the amount of information available and increase the confidence in the validity of the data.

The US DOE continues to support multiple projects focused on MSR accident progression sequences, such as salt spill testing (Thomas et al., 2023). Idaho National Laboratory performed the first ever chloride fuel salt irradiation (Hatch, 2023) and completed the final design of its fuel salt synthesis line to support the MCRE. Oak Ridge National Laboratory (ORNL) engineering scale fluoride and chloride salt pumped test loops were employed for sensor performance validation testing during 2023 (Figure MSR-3). ORNL's annual MSR workshop, sponsored by the US DOE-Gateway for Accelerated Innovation in Nuclear (GAIN) program, was held in person with a virtual option in October 2023.

CNL continues to develop expertise and capabilities in support of molten salt SMR concepts. The Canadian Nuclear Research Initiative (CNRI) program continues to foster joint-funded research projects with private stakeholders. The agreements with three MSR developers include work on electrochemical 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, CNL is developing molten salt capabilities across a wide range of areas including molten salt fuel behavior in accident conditions, salt chemistry and thermodynamic properties, multiphysics core behavior, thermal-hydraulics modeling and system behavior, and DHR. In 2023, R&D efforts included the development of encapsulation methods for measuring the melting point and thermal diffusivity of molten salts. Differential scanning calorimetry measurements were performed on selected compositions of interest for the $NaF-KF-UF_4$ pseudo-ternary system. The development and testing of electrochemical sensors, suitable for laboratory use and for online monitoring of corrosion tests in a natural circulation loop, was commissioned at the end of 2023. High-temperature molten salt corrosion test capability will be demonstrated by the long-term operation of the molten salt loop with controlled salt chemistry. Additional activities focused on static corrosion testing for candidate materials in molten salts and the development and testing of electrochemical sensors for chemical analysis in molten salts.

In Switzerland, MSR research continued in 2023 at the Paul Scherrer Institute (PSI) and focused on technology monitoring, education of new experts, and development of knowledge and simulation capabilities in fuel cycle neutronics performance, system behavior and thermodynamic areas of MSR research. In 2023, PSI and the Technical University of Munich in Germany prepared a system behavior analysis of the dual fluid reactor for the German/Canadian Company Dual Fluid Energy. As it is based on liquid fuel, the dedicated MSR capabilities of the system behavior code, such as the delay neutron precursors drift, were involved in the simulations. The fuel cycle neutronics performance tool was also further extended (Krepel et al., 2023) and applied to this reactor.

Source: DOE.

Figure MSR-3: Engineering scale chloride salt test loop

In 2023, Korea initiated a Gen IV project to develop key technologies for SMR-type MSR. Recognizing the significance of collaboration with private companies to expedite MSR demonstration, the government has established the MSR Basic Technology Development Agency. Private companies have demonstrated keen interest in MSR technology development, and the Korea Atomic Energy Research Institute (KAERI) is actively collaborating with several leading Korean companies with the aim of commercializing MSR technology in the near future.

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

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Super-critical water reactor

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

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

Main characteristics of the system

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

The main advantage of the SCWR is improved economics because of high thermodynamic efficiency, the potential for plant simplification, and decades of combined experience operating commercial water cooled reactors and supercritical fossil fired plants. 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. Table SCWR-1 provides some technical specifications of recent SCWR concepts.

Table SCWR-1: Key parameters of international supercritical water small modular reactor concepts

	Canadian SCW-SMR[1]	CRS-150[2]	ECC-SMART[3]
Parameters			
Thermal power [MW]	800	375	290
Electric power [MWe]	298	150	
Efficiency %	~37	~40%	
Pressure [MPa]	25	25.0	25
Reactor type	Pressure tube	Pressure vessel	Pressure vessel
Core inlet temp. [°C]	290	280	280
Core outlet temp.[°C]	500	520	500

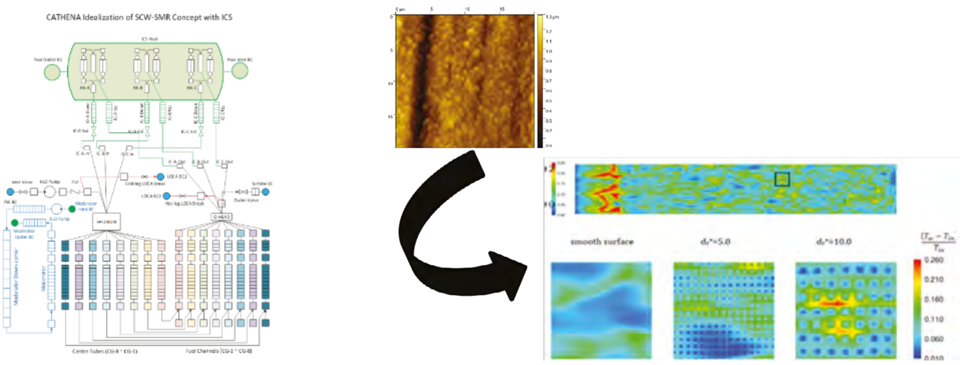
Note: MWe: megawatt electric; MPa: megapascal.

1. Nava Dominguez et al., 2024; 2. Ning et al., 2023; 3. Schulemberg and Otic, 2022.

Technical highlights – Thermal hydraulics and safety project

In Canada, Canadian Nuclear Laboratories (CNL) and Carleton University completed a direct numerical simulation (DNS) analysis on the effect of roughness on the boundary layer and heated channel flows for water near psuedocritical temperature. In previous studies, this analysis was carried out assuming smooth surfaces. The following phase aims to study the effect of an additional thermal layer to emulate corroded surfaces. Thus far, the DNS analysis showed that for a surface roughness of y+<10, the effects of roughness are negligible. Using these DNS results, new experiments will be conducted at Carleton University using the supercritical Freon loop (Mann et al., 2023). In parallel, CNL is developing a new tool to develop a heat transfer correlation/

Figure SCWR-1: CATHENA model of the Canadian SCW-SMR, and DNS analysis of a rough surface under supercritical conditions



Note: CATHENA model of the Canadian SCW-SMR (left); DNS analysis of a rough surface under supercritical conditions (right).
Source: Canadian Nuclear Laboratories (CNL).

model that takes into account the effect of surface roughness on flow. An exhaustive literature review was conducted, and models were identified.

CNL also conducted a multidisciplinary exercise to assess the feasibility of a supercritical water-cooled small modular reactor (SCW-SMR), based on the knowledge, lessons learned and experience gained through development of the Canadian SCWR (Gaudet et al., 2016). The SCW SMR operates at lower temperatures than the SCWR, but at the same pressure. The reason for this change is to reduce the core size and limit the demand on the in-core materials. A multidisciplinary exercise was completed from the beginning of the conceptualization, with emphasis also given to market conditions. An exhaustive analysis of the fuel bundle was performed in consideration of thermal hydraulics and safety. Several power profiles representing various enrichment options were analyzed. The results show that there is some margin to reduce the reactor core length (Nava Dominguez et al., 2024).

Using the thermal hydraulics system code CATHENA, three design basis accidents were modeled: 1) cold-leg loss-of-coolant accident (LOCA); 2) hot-leg LOCA; and 3) loss of feedwater accident. The reactor core was modeled by grouping channels with similar powers, resulting in eight groups. The results show that the concept is feasible, with findings in line with current water-cooled reactor analyses (Figure SCWR-1).

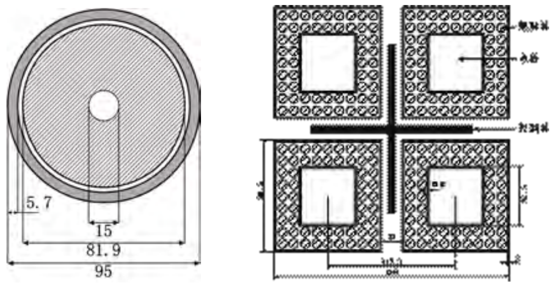
China's R&D activities are linked to two projects: 1) the Joint European Canadian Chinese – Development of Small Modular Reactor Technology (ECC-SMART) project; and 2) the IAEA coordinated research project, Advancing Thermal-Hydraulic Models and Predictive Tools for Design of SCWR Prototypes. The China Nuclear Power Institute is the main undertaker of China's SCW-SMR research and development activities. In 2023, R&D focused on the development of a small modular reactor (SMR) known as CSR150. Based on the design characteristics of CSR1000, the CSR150 concept plan was proposed by reducing the power and scaling the reactor (Ning et al., 2023), with the main design parameters shown in Table SCWR-1.

The core of CSR150 is equipped with 45 fuel assembly boxes (Lu et al., 2023): 21 with an enrichment of 5.7% in the center area and 24 with an enrichment of 7.2% in the outer area (Figure SCWR-2). A new fuel assembly with mixed moderators has been devised with simple design and improved economics and safety. In addition, the coolant flow mode was improved to meet the design criteria.

The China Academy of Sciences (CAS) is working on fundamental R&D of supercritical fluids, especially supercritical CO_2 (Zeng et al., 2023). It has proposed a novel approach using the Lattice Boltzmann Method (LBM) to solve problems related to supercritical fluid flow and heat transfer. A computational program was developed, based on the dual-distribution-function LBM model, that is suitable for simulating the natural convection of supercritical CO_2 in a two-dimensional cavity.

China is also improving the thermal hydraulics system code RELAP to improve the transition from subcritical to supercritical water conditions. The Nuclear Power Institute of China (NPIC) has completed the International Benchmark Study on SCWR Thermal Hydraulic Characteristics (Zhao, 2023), with disclosure and compilation of the results ongoing.

Figure SCWR-2: CSR150: Cross sectional views of fuel element and fuel assembly

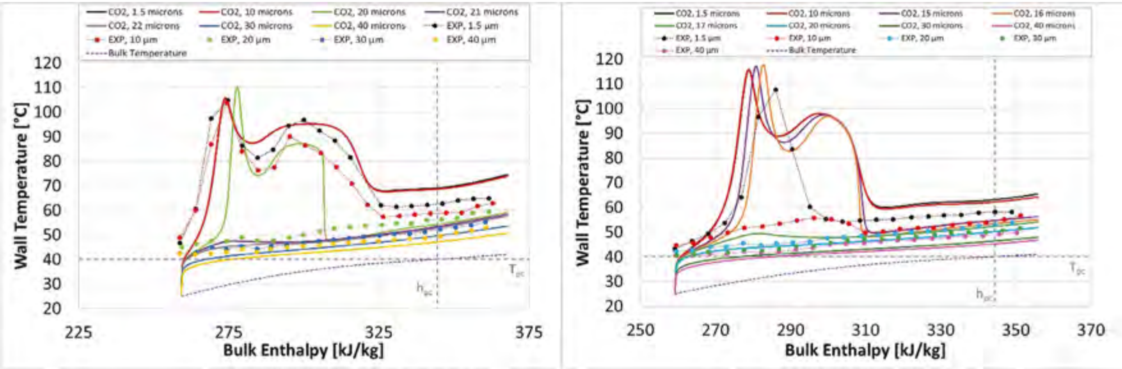


Source: Nuclear Power Institute of China (NPIC).

In Europe, the Karlsruhe Institute of Technology Model Fluid Facility is conducting an experimental investigation on the influence of corrosion on the heat transfer to supercritical fluid. This task is performed in cooperation with Czechia's Centrum výzkumu Řež (CVR). Research on SCWR focuses on experimental heat transfer studies, system design and safety, and numerical simulations using large eddy simulations (LES) and DNS to produce reliable data that complement physical experiments and support the development of practical engineering models.

Multiple European institutions are performing computational fluid dynamics (CFD) simulations. The University of Nottingham and the University of Technology and Economics in Budapest are working on activities related to CFD modeling of turbulent heat and mass transfer along corroded surfaces by investigating the influences of surface roughness induced by corrosion on heat transfer deterioration and pressure drop. The eddy effect of corrosion roughness is assessed in the turbulence flow model. In 2023, two models were mainly used: k-ε standard and k-ω shear stress transport (SST). Other models such as k-kl-omega and the transition SST model that has two or more Reynolds-averaged Navier-Stokes (RANS) equations will be used in future work. The CFD RANS model developed at the University of Pisa was also updated, including an in-house methodology that considers the impact of wall roughness effects (Figure SCWR-3) (Kassem et al. 2023). The model will be soon applied to the Karlsruhe experimental dataset to explore flow in rough tubes, thus allowing a better understanding of the capabilities of the proposed approach.

At the University of Sheffield, research on SCWRs 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 (Chinembiri at al. 2022; Wang et al.,



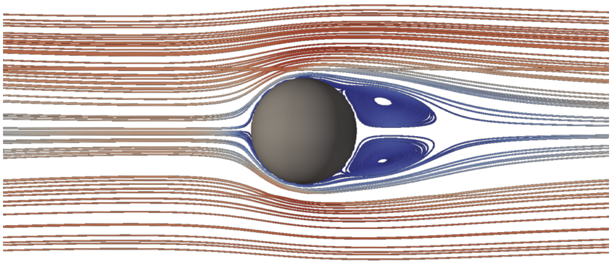
Note: Wall temperature profiles with different roughness for CO_2 at 9 MPa, inlet temperature 25°C, heat flux of 50 kW/m², and mass fluxes of 400 (left) and 450 kg/m²s (right).

Source: Kassem et al. (2023).

Figure SCWR-3: Wall temperature profiles with different levels of surface roughness

2021). This updated version of the DNS solver is being developed as a UK nuclear community code by the Collaborative Computational Project in Nuclear Thermal Hydraulics (CCP-NTH), which supports next-generation civil nuclear reactors. This technique has now been tested for a simple flow over a sphere (Figure SCWR 4). Simulations of flows over pyramid rough surfaces are also being developed and tested. The immersed boundary method will also be extended to energy modeling, thus allowing the consideration of the solid temperature distribution in complex geometries. Sub-channel and convectional CFD, both based on the CFD solver Code_Saturne, have been used in the benchmarking exercise organized by NPIC. The modeling of ribs on fuel surfaces has been developed using momentum source terms.

Figure SCWR-4: Flow over a sphere at ReD=80



Source: University of Sheffield.

The work performed at the University of Pisa consists of system thermal hydraulics and CFD applications. In the framework of the ECC SMART project, thermal hydraulics analyses of normal and accidental scenarios were performed for the ECC-SMART proposed concept (Schulenberg and Otic, 2022). Innovative Systems Software LLC worked with the University of Pisa to improve the RELAP5/SCDAP codes and the numerical stability. The results showed enhanced stability and was thus considered for the transient calculations (Pucciarelli et al., 2023).

Cooperation with the RELAP5/SCDAP developers enabled improvements relevant for the SCWR community. The code is now able to adopt a customizable heat transfer correlation, which may include some of the most common dimensionless groups considered in the heat transfer correlation for supercritical fluids proposed in the literature.

The correlation has the following general form:

$$Nu = \frac{hD}{k} = \left[1 + \frac{c}{\left(\frac{\varepsilon}{D_h}\right)^n}\right] a_1 Re_b^{a_2} Pr_b^{a_3} \left(\frac{\rho_w}{\rho_b}\right)^{a_4} \left(\frac{\mu_w}{\mu_b}\right)^{a_5} \left(\frac{k_w}{k_b}\right)^{a_6} \left(\frac{c_{p_w}}{c_{p_b}}\right)^{a_7} \left(\frac{\bar{c}_{p_w}}{c_{p_b}}\right)^{a_8}$$

Source: University of Pisa.

Technical highlights – Materials and chemistry project

In Europe, efforts in 2023 were focused on advancing corrosion tests committed in the ECC SMART project (which were detailed in the 2022 GIF Annual Report). This project studies two commercially available candidate alloys for manufacturing the cladding of supercritical water-cooled modular reactors (SCW-SMR): alloy 800H and austenitic steel 310S. Alloy 800H has been used in light water reactors, so there is a large body of knowledge regarding its corrosion behavior. Additionally, both alloy 800H and steel 310S have shown good corrosion resistance in SCW (Degmová et al. 2023; Marušáková and Šípová, 2022; Šípová et al., 2023). Therefore, under the ECC SMART and GIF materials and chemistry project management board projects, tests have continued for long-exposure times (several thousand hours). These projects are also investigating the effect of neutron radiation on these materials when subsequently exposed to supercritical water.

The phenomena of radiolysis in supercritical water are also being studied to define the chemistry of SCWRs, along with a detailed study of supercritical water electrochemical behavior. Initial radiolysis pre-tests have been conducted with promising results. This research will contribute to a prelim-

Source: Research Centre Řež (CVR)

Figure SCWR-5: Tensile specimen, oxidation coupon and transmission electron microscopy used in ECC-SMART

inary design for this type of SCW-SMR and further understanding of its chemistry. In 2023, a significant portion of oxidation tests were completed, and progress was made in slow strain rate tensile tests to evaluate crack initiation processes in materials. Oxidation tests with material pre-irradiated with neutrons began in Czechia (Figure SCWR-5).

As part of the ECC-SMART project's dissemination activities, a workshop on structural materials for fourth generation SMRs was held at the CIEMAT conference in Madrid, Spain (Figure SCWR-6). More than 50 participants attended from countries including Canada, China, Iraq, Ukraine and various EU member countries. The workshop covered topics from general introductions to more specific issues focused on waste management from fourth generation reactors, the use of artificial intelligence in material studies, testing procedures and techniques, and advanced materials, etc.

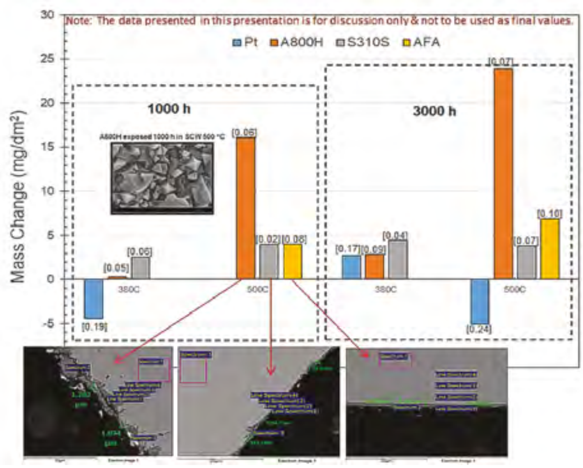
Figure SCWR-6: Workshop on structural materials for Gen IV SMR concepts flyer

Source: ECC-Smart (2023).

China has continued its study of candidate materials for cladding for supercritical water reactors, and has progressed in its participation in the ECC-SMART project (Su, Huang et al., 2023; Su, Sun et al. 2023). It has also continued to advance the study of the corrosion behavior of austenitic forming alloy (AFA) and oxide dispersion-strengthened (ODS) alloys, as well as studies of creep and fatigue, both under supercritical water conditions. There have also been advancements in the study of material behavior in supercritical CO_2. China provided a significant contribution to the ECC_SMART project in 2023 by promoting the standardization of testing procedures among participating laboratories. The goal of this standardization is to reduce the discrepancy of experimental results.

In Canada, CNL is contributing to ECC-SMART long-exposure corrosion tests and is performing tests at 23 MPa and 380°C and 500°C, which are conditions relevant to the operation of the Canadian SCW-SMR (Khumsa-Ang et al., 2023). The lower temperature was selected to investigate general corrosion close to the critical point of water, and 500°C is the nominal operating temperature at full power. At the end of 2023, results from 3 000 hours of A800H and 310S were completed (Figure SCWR-7). Specimens of AFA were manufactured by colleagues from the Shanghai Jiao Tong University and added to the test at 500°C. The last leg of testing is a 4 000 h run at 380°C to be completed in 2024.

Figure SCWR-7: Mass change of the candidate cladding materials and platinum coupons



Note: 3 000 hour results at two oxidizing temperatures 380°C (left) and 500°C (right).

Source: Canadian Nuclear Laboratories (CNL).

CNL results show that at the target operating conditions of 500°C, 800H gains about three to five times more mass than 310S and AFA. Although 310S and AFA show very similar weight gains after 1 000 h, between 1 000 and 3 000 h the AFA material did not gain significant mass, while the other two materials did. However, at 380°C, A800H gained less mass than 310S. A possible reason for this observation is a difference in activation energies between the two materials, which could suggest different oxidation mechanisms and different protective oxides. These observed trends will be confirmed after completion of the 7 000 h tests.

The CNL completed its first phase of high energy proton irradiations on two specific advanced nuclear materials (candidate materials for SCWRs applications), alloys 800H and 310S, at the TRIUMF proton accelerator facility in Canada. This study aims to assess the effect of irradiation dose (dpa, displacements per atom) on the microstructure and mechanical properties of the materials. Small punch testing was carried out on the proton irradiated samples to determine the dpa on the tensile properties of materials and estimate the yield and ultimate tensile strengths.

The samples were irradiated with high-energy protons with a maximum energy of 500 mega electron volts (MeV). The irradiation area was 10 mm × 10 mm. The achieved dose (dpa) and the respective amounts of helium and hydrogen generated by transmutation were calculated by FLUKA, which was also used to estimate the achieved dose distribution among the samples of alloy 800H and stainless steel 310S. The detailed data for the high-energy proton irradiated samples are shown in Table SCWR-2. The stack of four samples in the center of the parasitic stage achieved the highest dose, while the samples on the four sides and four corners achieved intermediate and low doses, respectively.

Table SCWR-2: Proton irradiation specifications

Material	Sample qty. irradiated	Dose level	Calculated dose (dpa)	Activity (mSv/hr)	Hydrogen (appm)	Helium (appm)
800 H	4	High	1.15	>100	381	126
800 H	8	Medium	0.5	90	150	48
800 H	8	Low	0.2	13.4	77	24
310 SS	0	0	0	0	371	123
310 SS	8	Medium	0.49	67	152	50
310 SS	8	Low	0.21	55	68	21

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

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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, with the United Kingdom welcomed to the System Arrangement in 2019. The present signatories are: the China National Nuclear Corporation, China; the Alternative Energies and Atomic Energy Commission (CEA), France; the Joint Research Centre, Euratom; the Japan Atomic Energy Agency (JAEA), Japan; the Ministry of Science and Information and Communication and Technology, Korea; Rosatom, Russia; the Department for Energy Security and Net Zero, United Kingdom; and the Department of Energy (DOE), United States. Four technical projects have been established for GIF collaborations:

1) SFR system integration and assessment, with seven members participating;
2) SFR safety and operations, with six members participating;
3) SFR advanced fuels, with seven members participating;
4) 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. In these conditions, austenitic and ferritic steel structural materials can be used and are highly compatible with sodium, while a large margin to coolant boiling at low pressure can be maintained. The reactor unit can be arranged in a pool or a compact loop layout. Table SFR-1 summarizes the typical design parameters of the SFR concepts 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.

Three general classes of SFR design concepts have been identified for Gen IV SFR research collaboration: loop configuration, pool configuration and SMRs. 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. These are the Chinese sodium-cooled fast reactor (CFR1200, China), the European SFR (ESFR, Euratom), the Japanese sodium-cooled fast reactor (JSFR, Japan), Korea's advanced liquid metal reactor (KALIMER, Korea), the BN-1200 (Russia) and the advanced fast reactor (AFR-100, United States). Gen IV SFR design tracks incorporate significant technology innovations to reduce 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 SFR-1: 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 MWt (10-2 000 MWe)
Fuel	Oxide, metal alloy and others
Cladding	Ferritic-martensitic, ODS and others
Average burnup	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 System Steering Committee (SSC) hosted a panel session for SFR demonstrations at the Generation 4 Small Reactors International Conference (G4SR-4), which ran in parallel to the GIF Industry Forum in October 2022 in Toronto, and invited international SFR companies to present and discuss potential collaboration opportunities to accelerate the demonstration of Gen IV SFRs. Several SFR companies were interested and accepted the invitation to collaborate through existing or new GIF projects. From December 2022 to April 2023, follow-up discussions took place between the GIF and some companies, with industry partners identifying "SFR thermal-hydraulic testing and validation" as the topic for a new project to be managed under the SFR SSC. This is the first time in a decade that a GIF project will involve industry members. To meet the near-term timelines needed by industry for validation it was decided that the founding members of the new project would initially be limited to expedite the establishment of project arrangements, with the full understanding that additional members will be invited to join immediately following the project's formalization. As of December 2023, the addition of two SFR industry members and the project arrangement provisions are undergoing approval by the SFR SSC before the project can proceed to the next step.

In the near term, SFR demonstrations continue to be developed in GIF countries. In China, in addition to the operation of the China Experimental Fast Reactor with 20 MWe, two CFR-600 units, which are demonstration SFR plants generating 600 MWe each, are under construction in Xiapu County, Fujian Province. The construction of units 1 and 2 started in 2017 and early 2021, respectively (WNN, 2020). A commercial-scale unit with a capacity of 1 000-1 200 MWe is also planned (WNN, 2021).

Within the complementary Euratom Research and Training Program and Horizon Europe, the ESFR-SIMPLE project aims to improve the safety of the European SFR through innovative monitoring, power level flexibility and experimental research (ESFR-SMPLE, 2024). In addition, the Sustainable Nuclear Energy Technology Platform, through the European Sustainable Nuclear Industrial Initiative (ESNII), supports SFR R&D coordination activities.

In France, a wide-ranging investment plan, France 2030, was set up in October 2021 to transform key sectors of the French economy through innovation, industrialization and research. Support from France 2030 includes developments focusing on disruptive nuclear reactor concepts and the fostering of new, emerging players. A first call for proposals was launched in 2022 aimed at startups involved with the development of innovative modular nuclear (fission or fusion). Currently, eight projects have been selected, with two led by new companies, HEXANA and Otrera Nuclear Energy, that are looking to develop new reactors based on SFR technology.

In Japan, the Strategic Roadmap for Fast Reactor Development was revised and approved at a ministerial meeting in December 2022. In July 2023 the sodium-cooled tank fast reactor was selected as the conceptual design target. The conceptual design will be conducted in the period 2024 to 2028, with Mitsubishi Heavy Industry Co., Ltd chosen as the core company responsible for the design, manufacturing and construction (WNN, 2023).

In Korea, the engineering design of Prototype Gen IV Sodium-cooled Fast Reactor (PGSFR) was completed in 2020, and a new SFR program is underway aimed at developing an SFR-based SMR (SALUS). SALUS is a pool-type SFR with an electric output of 100 MWe. It operates on a 20-year refueling cycle utilizing metal fuel. The conceptual design of SALUS keeps most design features of the PGSFR, except for fuel and core design. Ongoing R&D programs working on the SFR area consist of three groups of common and necessary technologies: 1) experimental design validation and computational design codes verification and validation; 2) innovations in metal fuel technology; and 3) demonstration of safety performance (Eoh et al., 2024).

In Russia, commercial SFRs such as the BN-600 and BN-800 are currently operating. The BN-600 first began operating in 1981, and its operating license was extended by a further five years in 2020. Several important upgrades of the BN-600 are being conducted to seek a further license extension to 2040 (WNN, 2024a). The BN-800, which produces about 820 MWe, entered commercial operation in 2016 with uranium oxide (UO_2) fuel. In 2022, the reactor was fully loaded with uranium-plutonium mixed oxide (MOX) fuel and continued operation (WNN, 2024b). In 2023, an engineering survey was initiated to specify geodetic, hydro-meteorological and environmental characteristics of the site for the BN-1200M (Rosatom, 2023).

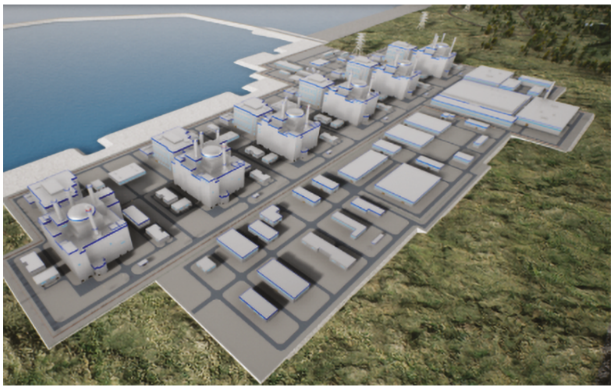
In the United Kingdom, the Department for Energy Security and Net Zero is providing funding for the Advanced Modular Reactor (AMR) Knowledge Capture Project to facilitate knowledge capture and sharing with the aim of reducing the time, risk and cost of AMR deployment. The project includes the collection and transfer of all legacy knowledge gained in developing a range or reactor types, including SFR (Bangor University, 2023).

In the United States, the first Natrium demonstration SFR plant by TerraPower is planned for construction in Kemmerer, Wyoming, with anticipated submission of the plant's construction permit application to the US NRC in 2024. Natrium is a 840 MWt pool-type SFR that contains a compact safety footprint and a molten salt energy storage system which enables the plant to vary its supply of energy to the grid, up to 500 MWe net, while maintaining constant power. The construction is funded by a public-private partnership through the US DOE Advanced Reactor Demonstration Program, which authorizes a 50/50 cost share (TerraPower, 2024a). Most recently, TerraPower has awarded supplier contracts for the design and fabrication of the Natrium demonstration reactor equipment (TerraPower, 2024b). Oklo intends to submit an application for the Aurora Powerhouse SFR design with a 15 MW capacity. The design is based on metal alloy fuel, sodium coolant and recycling technologies. Oklo has received a site use permit from the US DOE, with plans to use recovered fuel from the Experimental Breeder Reactor II in its first reactor (INL, 2020).

Technical highlights – System integration and assessment project

China has initiated a new SFR project, the integrated fast reactor nuclear energy system (CiFR1000), as the future commercial fast reactor. The CiFR1000 layout will be six GWe-class SFR units in one plant site, with a fuel regeneration sub-component, a waste treatment sub-component and other supporting sub-components (Figure SFR-1). Six reactors and the supporting fuel reprocessing facility together constitute a complete closed fuel cycle system. The most important feature of the CiFR1000 is the rapid circulation and multiple cycles of nuclear fuel within the plant, with the spent fuel being directly regenerated after leaving the reactor, processed into fresh fuel, and then reintroduced into the reactor. The electric power of CiFR1000 is 1 200 MW and the fuel type is U-TRU-Zr (uranium-transuranics-zirconium) metal alloy fuel. A four-loop design is used in the reactor, with each loop configured with two steam generators arranged side by side around the reactor vessel to reduce the size of the nuclear island. There is an innovative DHR system design to enhance the natural circulation capability under accident conditions.

Figure SFR-1: Layout of China's integrated fast reactor nuclear energy system, CiFR1000



Source: China Institute of Atomic Energy (CIAE).

In France the CEA has continued considering the application of a selected set of GIF Safety Design Criteria and Safey Desing Guidelines (SDC-SDG) to the ASTRID-600 plant design with the aim of providing recommendations and highlighting potential gaps. A 2023 contribution reviewed the application of the SDC-SDG to DHR systems; in particular, the issue of sodium natural circulation was considered.

The ESFR concept is being further developed within the current Euratom project, ESFR SIMPLE, which focuses on innovative monitoring, power level flexibility and experimental research. The Euratom contribution is a review of sodium technologies and a dedicated roadmap for R&D studies, and an exploration of the synergies and commonalities with other liquid metal systems. The study was performed with the ESFR SMART project and reviewed sodium technologies that are commonly used or under development in sodium experimental facilities for safety enhancement in SFR. The study identified promising technologies for further development and assessed their technology readiness levels (TRLs), with the priorities and actions to raise the TRLs of some selected technologies outlined. The considered technologies include permanent magnet electromagnetic pumps (EMPs), flow regime identification from magnetic field mapping in EMPs, thermoelectric systems, and inductive level measurements. The sodium permanent magnet EMPs can be regarded as reaching proof of concept (TRL-3) for conditions close to those in fast reactors. The technology is tested at laboratory scale to identify/screen potential viability in wider applications. Two actions to raise the TRL are ensuring permanent magnet pump parameter stability and compliance with existing electromagnetic emission standards. The review showed the feasibility of flow regime identification from magnetic field mapping in EMPs when operating with a sufficiently high magnetic Reynolds number. Using thermoelectric systems to create secondary emergency cooling solutions in SFRs and inductive level measurements of the position and shape of the free surface of sodium in the upper plenum were found to be promising.

The JAEA are applying the GIF SDC-SDG to a pool-type JSFR to draw recommendations on how to apply the GIF SDC-SDG to a specific SFR design and to highlight any potential gaps. As a first study, the DHR systems of the pool-type JSFR were analyzed according to the GIF SDC-SDG.

The Korea Atomic Energy Research Institute contributed to the evaluation of uncertainty on physics parameters for a TRU metal fuel core. The purpose of their study is to quantify uncertainty and rationalize methodology to reduce uncertainty given the lack of suitable experimental data for validation of TRU metal fuel core design. The SANDY/NJOY/McCARD system has been developed to assess the statistical uncertainty induced by covariance data from evaluated nuclear data files. The criticality uncertainty for the reference TRU core model was found to be about 1 285 percent mille (pcm). Several fast reactor benchmark problems were also reviewed and selected to evaluate the similarity coefficients against the reference TRU model. Among the 63 benchmark problems, the similarity coefficients of 32 benchmark problems were found to be higher than 0.9.

The US DOE supported an assessment and application of the MOOSE software framework for modeling SFR core bowing. Work was performed at the Argonne National Laboratory to assess the capability of the new MOOSE-based tools, which can tightly couple individual high-fidelity physics tools to predict the integrated behavior of core bowing due to multiphysics. SFR core bowing is a complex behavior that can be tailored to provide beneficial reactivity feedback through the design of the core restraint system. There is therefore motivation to better predict its behavior using state-of-the-art multiphysics tools. Work in 2023 involved utilizing established IAEA benchmark test cases to verify calculations using various structural mechanics modules within MOOSE, such as tensor mechanics and contact. The preliminary results showed that the MOOSE-based approach was promising, with additional verification efforts with increasingly complex models planned to continue.

Technical highlights – Safety and operations project

The China Institute of Atomic Energy (CIAE) presented a study regarding the design of a two-tank thermal heat storage system based on molten salt to integrate with an SFR that has a thermal power of 280 MW. The study proposed a salt-sodium heat exchanger based on the thermal physical properties of the sodium and salt, and storage tanks for both cold and hot salt based on the thermal heat storage system of a real solar thermal power plant. Modeling and dynamic simulation of the proposed storage system showed that the salt-sodium heat exchanger functioned as anticipated, storing heat for 18 hours when demand was low and providing 500 MW thermal power to the steam generator for 6 hours when demand increased. The variables of the storage heat exchanger sodium side remained unchanged during the whole process, meaning that the SFR can work at full power during the heat storing and releasing process.

As part of the CEA SFR technology development, innovative options are being explored to improve technological safety responses in advanced fuel degradation conditions or even severe accident conditions. One of these options consists of a mitigation device in the form of a transfer tube designed to discharge a large fraction of the molten fuel into a core catcher as early as possible in the case of a severe accident sequence. In this context, the SAIGA program (Severe Accident In-pile experiments for Generation IV reactors and the ASTRID prototype) is currently underway (Payot et al. 2023). The program consists of in-pile experimental tests that consider the degradation of one or more fuel pins during a loss-of-coolant sequence, implemented at the Impulse Graphite Reactor operated by the National Nuclear Center of the Republic of Kazakhstan. The experimental device copes with three azimuthal sectors including two sub-assemblies of 16 fuel pins, and a third sector filled with sodium simulating a corium transfer tube.

The CEA is also continuing work on upgrades and enlargements of its out-of-pile experimental capabilities for the understanding and modeling of corium behavior during severe accidents. The conceptual design of the future experimental platform SAFETY, which will replace and extend PLINIUS capabilities, was carried out in 2023 (Journeau et al., 2023). The basic design should be launched in 2024.

As part of the ESFR-SMART project, Euratom performed an assessment of different accidental conditions. The unprotected loss of flow (ULOF) scenario is the most severe accident evaluated within the research project and provides insight into what happens to the reactor in a beyond design basis accident where the integrity of the reactor cannot be guaranteed by specifically designed safety features. However, a state-of-the-art low void effect core is implemented as it has been previously demonstrated that such a design can result in a stabilized sodium boiling process, and subsequent power excursion can be evaded. The behavior of the ESFR-SMART core has been investigated with four codes: TRACE, SIM-SFR, SAS_SFR and SIMMER. In the analysis, special attention has been given to the sodium boiling reactivity effect and the coolant flow pattern evolution throughout the accident. Different aspects of the ULOF progression were studied and the main findings summarized for every code. Different phenomena that can affect the boiling phase have been highlighted thanks to the different modeling used, and the main issues for future R&D are proposed. To reinforce the stabilization process, a refined core assembly design has been successfully implemented and tested.

In an additional study devoted to the dynamic reactivity effect of pressure waves, Euratom developed general methodologies to simulate time-dependent neutronic behavior in the case of non-uniform core deformations, with a particular interest in evaluating dynamic reactivity effects. Two methodologies were created: 1) the PSI methodology applicable for the use of point kinetics approximation for neutronics; and 2) the HZDR (Helmholtz-Zentrum Dresden-Rossendorf laboratory) methodology for users with 3D diffusion solvers. Both approaches agree remarkably in terms of neutronic behavior during a scenario of pressure wave propagation. Strengths and challenges were identified for each approach.

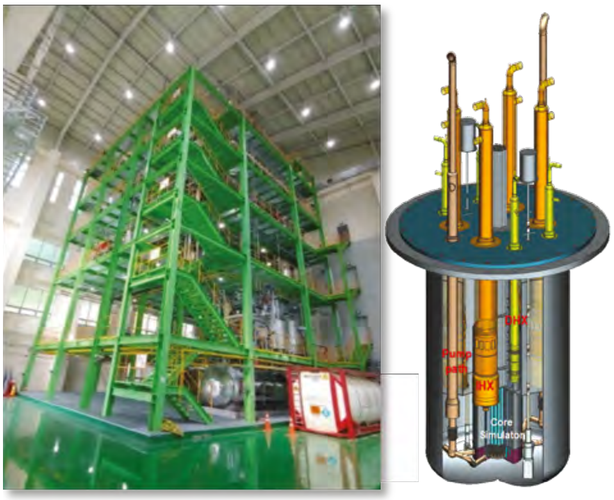
The JAEA has conducted a detailed structural analysis of a reactor vessel in a loop-type SFR using a commercial finite element analysis code, FINAS/STAR. The aim of this analysis was to understand the deformation behavior of the reactor vessel under extremely high-temperature conditions and to identify the areas that should be focused on to mitigate the impacts of failure. In a proposed scenario of loss of the heat removal system that might lead to a severe accident, the reactor vessel was heated from normal operational conditions to the sodium boiling temperature in the upper sodium plenum for 20 hours (the study assumed depressurization). The analysis revealed less significant stress and strain than expected, sufficiently lower than failure criteria. The upper body of the reactor vessel was identified as the important area in terms of mitigation of structural failure. The reactor vessel was eventually deformed downward by about 160 mm, with no failure. This analysis implies that maintaining the reactor vessel sodium level in the long term can enhance its structural resilience.

The JAEA has developed a Lagrangian particle method code to improve the conservativeness of the LEAP-III code, which can evaluate the tube failure propagation initiated by a small water leak jet in a steam generator. The jet behavior and chemical reaction are simulated in this code using Newton's equation of motion. Engineering approximations replace explicit solving of multiphase thermal-hydraulic equations for sodium-water reactions. A test analysis was performed to compare the results calculated by this particle method with those calculated by the SERAPHIM code, which has been validated by existing experiments and can predict temperature distribution in detail. Through this test analysis, the basic capability of LEAP-III code, coupled with this particle method, is confirmed to predict a realistic temperature distribution with low computational cost, and to predict tube failure occurrence.

KAERI has been actively working on thermal-fluidic transient experiments using a large-scale test facility, STELLA-2 (Figure SFR-2), since 2020. In 2023 it conducted a substantial set of experiments for asymmetrical operations with single pump failure. The flow distribution inside the pool exhibited distinguishable temperature behavior with various combinations of the multiple DHR system conditions. Backflow to the cold pool through the failed pump line was observed, but it did not influence the overall heat transfer. The code calculation comparison was also conducted and showed reasonable agreement; however, further analysis on the heat loss effect needs to be undertaken.

KAERI has conducted modeling and analysis of the loss of heat sink transients as an EBR-II benchmark problem using the GAMMA+ code (General Analyzer for Multi-component and Multi-Dimensional Transient Application). In this analysis, the transient core power was assumed as a boundary condition to conduct validation of the thermal-hydraulic model. A sensitivity analysis of the cold pool model was conducted to simulate temperature transients of the stagnant and mixing region in the cold pool. In 2024, analysis incorporating a reactivity feedback model will be conducted based on the established thermal-hydraulic model, and the results will be shared.

Figure SFR-2: STELLA-2 facility layout



Source: Lee et al. (2018); KAERI.

Technical highlights – Advanced fuels project

The CIAE has completed the process test of simulant Cu-Ce-Zr (copper-cerium-zirconium) injection casting. The U Ce Zr alloy test melting scheme was determined as U+20Ce+10Zr integrated melting. The melting test of U-Ce-Zr alloy at different temperatures and holding times has been carried out in an injection casting furnace. The completion of the U-Ce-Zr ternary alloy melting test and the subsequent casting test will lay the technical foundations for the injection casting of U-Pu-Zr rods in 2024.

The CIAE also completed research on the vacuum induction melting process of U-10Zr. The technology consisted of two steps: non-consumable arc melting and induction melting. The well-alloyed ingot with Zr content between 9.5% and 10.9% was obtained by secondary melting of intermediate alloy. Research on injection molding technology of U-10Zr was also completed. Using uranium as the raw material, two kinds of injection tests were carried out, showing good surface quality for 120 mm and 190 mm long rods with an outer diameter of 6 mm.

In 2023, the CEA in France continued improving the simulation and optimization of SFR fuel and a fuel assembly design. Regarding oxide fuel performance evaluation, new measurements of thermal properties (thermal diffusivity and heat capacity) of MOX fuel were carried out as part of the ESNII experimental program on fresh and irradiated fuels under fast reactor conditions. These fuels were characterized and their thermal diffusivity and heat capacity measured using the laser flash technique. Their thermal conductivity was calculated using hydrostatic density measurements when available, or calculated density. A clear discrepancy between experimental results and the most commonly used thermal conductivity model (Lucuta et al., 1996) was observed. Lucuta's model was then modified and a new recommendation for fast reactor MOX fuels has been provided. This new thermal conductivity model describes different effects induced by irradiation on the degradation of thermal conductivity: soluble fission products, precipitated fission products, radiation damage and porosity. The chemical composition, microstructure and temperature of the irradiated fuel were estimated using thermomechanical calculations (PLEIADES-GERMINAL V2) and thermochemical calculations (Thermo-Calc V4.1 with the TAF-ID V11 database). This work demonstrates that by introducing the neodymium content as the parameter driving the thermal conductivity of the matrix, a better agreement between the model and the experimental data can be achieved.

The CEA is also continuing its work on ODS as candidate material for the fuel cladding of future fast neutron reactors. This material presents a very good swelling resistance to radiation damage, but the effect of radiation on the fracture behavior must be addressed. A pin loading tension test was developed to evaluate the fracture behavior of thin-walled tubes. The fracture resistance curve was determined using the single specimen technique based on the elastic unloading method following the procedure described in the ASTM standard[1]. In this work, the geometric functions needed to evaluate the crack length from the unloading compliance, the elastic and plastic parts of the -integral, were determined using finite element analysis. Special care was taken to consider the effects of contact and friction between the test specimen and the loading device. Several tests were carried out and a good reproducibility of the fracture resistance curve was observed. The low scatter is mainly attributed to the variation of the gap between the internal radius of the pin loading tension (PLT) specimen and the radius of the mandrels. It is recommended to minimize the gap to limit progressive contact. A direct method was used to determine the crack extension at the beginning of the test to provide a better measure of the fracture tightness for the low ductility material. In the case of more ductile materials, the developed method is likely to be directly applicable without relying on optical measurements of crack extension.

The JAEA continues to measure and evaluate the physical properties of MOX fuels. Specific heat is one of the basic physical properties used to evaluate thermal conductivity and fuel temperature in transient conditions. The JAEA measured the enthalpy of MOX (Pu/metal=18 atom%) up to 1 950°C using a drop calorimeter to evaluate the specific heat.

1. ASTM E1820-22e1: Standard Test Method for Measurement of Fracture Toughness.

Low-decontaminated fuel that contains significant fission products has been investigated as a fuel for the advanced fast reactor cycle. However, the number of studies on the thermal conductivity of MOX containing fission products are limited. The JAEA evaluated the thermal conductivity of MOX, including Nd_2O_3 and Sm_2O_3, which are expected to be the main fission products to remain as solid solutions in low-decontaminated fuel. The JAEA measured the sintering behavior of dry-recycled MOX powder with various particle size distribution, depending on grinding conditions.

Metal fuel has been developed by KAERI for the PGSFR, where fuel assembly is being designed to satisfy requirements for core performance and safety. A structural analysis of the fuel assembly was carried out to consider accident conditions such as protected transient overpower and protected loss of flow. The maximum stress values in those cases were found to be far below the stress limit design criteria. New material casting parts capable of reducing reaction loss have been developed. A new material, $NdYO_3$, was developed as a candidate coating material to inhibit crucible reactions for metal fuel casting. Green compacts were made by cold isostatic pressing, with molar ratio of Nd_2O_3 (neodymium(III) oxide) and Y_2O_3 (yttrium oxide) powder. The out-pile performance test (Sessile drop test) showed good results. Further work will be done by optimizing the process of coating and liner fabrication using $NdYO_3$.

KAERI is developing barrier cladding tube technology to suppress fuel-cladding chemical interactions (FCCI), particularly for minor actinide-bearing metal fuel. Electroplating is considered one of the best methods on account of its applicability to the cladding inner surface. Chromium (Cr) has been selected for the electroplating because of its higher resistance to fuel-cladding chemical interaction and its higher technical maturity. The introduction of a Cr and Cr nitride multi-coating barrier between the nuclear metallic fuel and the cladding is being studied as a potential candidate to mitigate FCCI. The influence of direct current and pulse-reverse current Cr coating electrodeposition on multi-coating barrier performance is being investigated.

The US DOE has recently been working to revitalize its R&D efforts for U-Pu-Zr metallic fuels in the United States. Recent progress has been primarily in the areas of transient testing of metallic fuels and performance characterization for accelerated burnup testing methods. The first transient irradiations were performed in a new temperature heat sink overpower response device in the Transient Reactor Test Facility. These experiments are evaluating the off-normal performance of metallic fuels, including for model development and validation purposes. A report was completed in 2023 to provide pre-test predictions of the experiments using the Bison code developed for metallic fuels. This will enable comparisons with later as-run evaluations. Continued experiments are planned on fuel pins irradiated in EBR-II.

The fission accelerated steady-state test experiments at INL are continuing irradiation exposure in the advanced test reactor. Several specimens have achieved burnup targets (low-mid burnups) to evaluate the application of this testing method relative to historical prototypic irradiations performed in EBR-II. Initial post-irradiation examinations have been performed, and progress is being made towards more advanced characterization (Figure SFR-3). Initial findings show interesting results that are consistent with the prototypic results, but that raise some questions that are under continued evaluation.

Figure SFR-3: Example of first irradiated FAST rodlets

Source: INL

Note: rodlet (lower left) and metallography (lower right).

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

To carry out mechanical testing under hot sodium, the CEA has designed and built a new testing facility called MEChAnical Tests facility in sodium (NA), MECANA, that has the flexibility of use of a glove box and the sodium testing capabilities of a test loop. This testing device consists of a classic sodium loop base with storage and a sodium purification unit with test pots. The upper part is integrated into an argon inert glove box, which allows the test configuration to be rapidly changed as the test section can be easily opened without air ingress into the circuit. MECANA handles sodium with a high chemical quality obtained through an active purification system. The classical subsystems of such a sodium facility are present: storage vessel, cold trap, plugging-meter. The atmosphere above the sodium surface is composed of argon, and pressure is regulated at a slight overpressure to prevent air ingress. The sodium circuit has a test section for the implementation of experimental devices. Qualification tests of ultrasonic instrumentation and of building elements robotics are already performed at the facility.

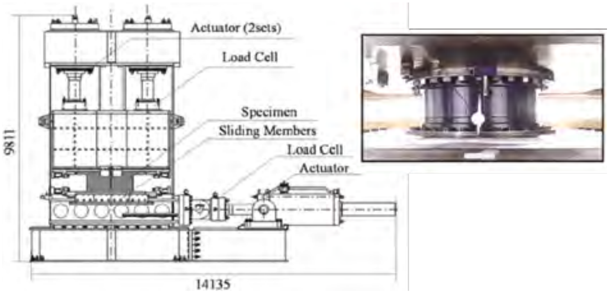
Regarding SFR decommissioning, the CEA developed the NOAH process during the 1980s. This process is dedicated to the treatment of drainable radioactive sodium, which corresponds to approximately 99% of the sodium inventory when dismantling an SFR. The name was chosen according to sodium hydroxide, 'NOAH' being an anagram of NaOH. The process consists of a continuous and controlled reaction between liquid sodium and water, producing aqueous sodium hydroxide and hydrogen according to the following reaction: Na liq + H_2O liq → NaOH aq + ½ H_2 (hydrogen) gas. The process employed is different depending on the amount of sodium that has to be treated. Residual sodium in components can be treated by hydrolysis in cleaning pits or by using the carbonation process, whereas drainable sodium is treated using the NOAH process. This process has been employed at an industrial scale using a CEA patent (FR8606266/EP0245148B1). Drainable sodium from the reactors RAPSODIE (37 t treated in DESORA facility, France), PFR (1 500 t treated in SDP facility, United Kingdom), SUPER-PHÉNIX (5 500 t treated in TNA Facility, France) and KNK2 (Germany, drainable sodium treated in SDP facility by the UK Atomic Energy Authority) have been treated using the NOAH process. In the future, drainable sodium from PHÉNIX will be treated with this same process (1 500 t to be treated in NOAH facility, France).

The JAEA has developed horizontal seismic isolation technology as a key component of SFR development. In Japan, several types of seismic isolation system have been developed; however, as the vertical natural frequency is designed from 20 Hz (hertz) to 16 Hz, such a system did not meet design requirements in the seismic response. Therefore, to fully satisfy seismic design requirements such as horizontal and vertical natural periods, the JAEA developed an advanced natural rubber bearing with rubber sheets of 31 mm thickness. Experiments confirmed that the developed natural rubber bearing satisfies the specific requirements such as shear strain and compression stress, and its use as a horizontal seismic isolation technology is feasible. The experimental apparatus is shown in Figure SFR-4.

In 2023, KAERI continued the development of the plate-type ultrasonic waveguide sensor array for under-sodium viewing. Based on preliminary tests conducted in 2022, the signal analysis software was upgraded by adopting the cross-correlation technique for more accurate calculation of the time delay value for each receiving waveguide sensor in the sensor array. The characteristics of the prototype waveguide sensor array were also analyzed through several under-water viewing tests conducted for a specimen with two adjacent slit defects.

KAERI also continued a study to develop advanced sodium instrumentation techniques based on a high-fidelity distributed temperature sensor. Several pre-tests of a prototype reference component for a high-fidelity distributed temperature sensing and monitoring system were conducted, and the signal characteristics were analyzed through ice-bath tests to improve the reliability of temperature sensing. New test equipment to demonstrate the performance of high-fidelity distributed temperature sensing and monitoring system were then designed and fabricated.

Figure SFR-4: Experimental apparatus of seismic isolation



Source: Fukasawa et al. (2015).

In 2023, the US DOE tested the primary sodium operations of a new experimental test device, the Thermal Hydraulic Experimental Test Article (THETA), an electrically heated pool configuration used for validating thermal-hydraulic codes. THETA consists of a primary and secondary system. Due to analysis conducted during primary sodium testing, the THETA test article required the addition of insulation around the core and intermediate heat exchanger. Details of the THETA removal, cleaning, reconfiguration and installation were provided, along with information and progress on installing the THETA intermediate heat transport system. THETA has installed multiple conventional temperature sensors and specialized fiber-distributed temperature sensors that provide high resolution data for thermal-hydraulic code validation. The US DOE has also shared information on recent operations and maintenance of the intermediate scale sodium facility.

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

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Very-high-temperature reactor

The System Arrangement for Gen IV international R&D collaboration on the very high-temperature reactor (VHTR) was 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:

1) **VHTR fuel and fuel cycle**: China, Euratom, France, Japan, Korea, and the United States as members. Canada and the United Kingdom are currently observers.

2) **VHTR materials**: China, Euratom, France, Japan, Korea, Switzerland, and the United States as members. Canada and the United Kingdom became new members with the approval of the third project arrangement on 3 January 2024.

3) **VHTR hydrogen production**: Canada, China, Euratom, France, Japan, Korea, and the United States as members.

4) **VHTR computational methods validation and benchmarks**: China, Euratom, Japan, Korea, and the United States as members. Canada and the United Kingdom are currently observers.

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.

In terms of GIF VHTR SSC collaboration activities, the GIF Risk and Safety Working Group (RSWG) published important regulatory information on VHTR safety design criteria in June 2023, and a new joint study with the RSWG and Proliferation Resistance and Physical Protection Working Group (PRPPWG) was launched to assess the intersection of Safety, Security and Safeguards (3S) by Design. This joint study utilizes a generic pebble-bed high-temperature gas-cooled reactors (HTGR) model developed by the US DOE.

Main characteristics of the system

HTGRs are helium-cooled graphite-moderated nuclear fission reactors that use fully ceramic tri-structural isotropic (TRISO)-coated particle-based fuels. HTGRs are characterized by inherent safety features, excellent fission product retention in the fuel, and high-temperature operation suitable for high efficiency generation of power and industrial process heat, particularly 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 HTGR, targeting even greater efficiency and more versatile use by further increasing the helium outlet temperature to 1 000°C or higher, which will require new structural materials, especially for the intermediate heat exchanger.

The operational envelope of HTGRs can be adapted to specific end-user needs, and a significant near-term market exists for process steam of approximately 400-550°C, which is achievable with lower temperature HTGR designs. There was a significant development in the United States in 2023, with X-energy agreeing to deploy their first four units at a major chemical producer site in 2029 to deliver electricity and high-quality steam for process heat (X-energy, 2023). Inherent safety in accident conditions is assured by the low power densities and high thermal inertia of typical HTGR designs. The potential for high fuel burnup (150-200 gigawatt days per metric ton of heavy metal [GWd/tHM]), high efficiency and modular construction, all constitute advantages favoring commercial HTGR deployment.

The HTGR standard fuel form is based on UO_2 TRISO-coated particles (UO_2 kernel, buffer/inner pyrocarbon/silicon carbonide/outer pyrocarbon coatings) embedded in a graphite matrix, which is then formed either into 6 cm diameter pebbles or compacts of various geometries embedded in hexagonal fuel blocks. This fuel form exhibits 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 using a uranium oxycarbide (UCO) fuel kernel, a zirconium carbide coating layer instead of silicon carbide (SiC), or the replacement of the graphite matrix material with SiC.

The current HTGR fuel cycle is a once through, very high burnup, low-enriched uranium fuel cycle, with solutions to adequately manage the back-end of the fuel cycle or synergetic fuel cycles under investigation. Significant research is being performed internationally on TRISO and graphite waste processing, in particular to reduce waste volumes. Power conversion options include indirect Rankine cycles or direct or indirect Brayton cycles.

High-temperature reactor (HTR) demonstration projects

Several demonstration projects are currently being pursued to meet the needs of industries interested in electrical and process heat applications. Both prismatic and pebble-bed HTGR designs are being developed, ranging from small transportable units with typical power outputs of less than 10 MWt to multi-unit plants generating more than 1 000 MWt.

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. After demonstrating a 168-hour continuous running period in late 2023, the HTR-PM (Figure VHTR-1) entered commercial operation (WNN, 2023a). In addition, a loss of off-site power test was carried out on reactor module No. 1 on 13 August 2023, followed by an emergency shutdown test on reactor module No. 2 on 1 September 2023. The tests confirmed that the reactors could be naturally cooled down by utilizing only inherent physics feedback, and without the need to rely on any emergency core cooling systems. Significant progress was also made on the detailed design of the HTR-PM600, a 600 MWe commercial plant with six modules, with the preliminary safety analyses report review completed in October 2023 by the Chinese regulatory body.

Figure VHTR-1: The nuclear island of the High-Temperature Gas-Cooled Reactor – Pebble-bed Module (HTR-PM), China

Source: Tsinghua University.

The High Temperature Engineering Test Reactor (HTTR) in Japan was restarted in 2021 after a decade of shutdown following the Great East Japan Earthquake in 2011. Following a successful simulated loss-of-coolant test in 2022, in which all cooling systems were shut down with the reactor at 30% power and no control rod actuation, a third loss of forced coolant test is planned for 2024 with the reactor at 100% power to further demonstrate the inherent safety shutdown mechanism in this test reactor. There are also plans to perform various tests concerning safety, core physics, thermal-fluid characteristics and fuel performance, including a heat application test that will couple the HTTR and a hydrogen production plant.

The HTGR landscape has seen significant new activities in the United Kingdom, with three 50% cost-share awards made in April 2023 for two-year projects:

1) Ultra Safe Nuclear Corporation (USNC) UK to develop a modified version of its Micro Modular Reactor with an increased power output (≥ 60 MWt) and outlet temperature (750°C), employing the USNC's Fully Ceramic Micro-encapsulated fuel (WNN, 2023b).

2) National Nuclear Laboratory (NNL), in collaboration with JAEA, to develop a 250 MWt HTGR based on JAEA's experience with the HTTR and existing JAEA-Mitsubishi Heavy Industries' concepts for commercial systems.

3) NNL and JAEA to implement a pilot-scale TRISO fuel manufacturing capability.

As part of the French government's larger commitment to support nuclear projects and startups developing small nuclear reactor technology, the France 2030 investment plan has awarded funding to eight projects, including two based on HTGR technology (O'Brian, 2024):

1) Jimmy Energy: A thermal spectrum prismatic HTGR microreactor that operates at high temperatures and aims to reduce radioactive waste.

2) Blue Capsule and CEA: A compact and safe HTGR type core using TRISO fuel and sodium as heat carrier in natural convection, providing industrial heat at 700°C.

The National Centre for Nuclear Research (NCBJ), in cooperation with JAEA, has released the conceptual design of a new Polish high-temperature research reactor (HTGR POLA) (WNN, 2023c). This 30 MW prismatic-type research reactor could be used for co-generation, with a maximum gross electric power of 10 MWe and high-temperature steam with a temperature of 540°C. It could also be used for industrial processes or municipal heating.

The Lappeenranta University of Technology plans to deploy a micro-HTGR as a research and test reactor in southern Finland. The reactor will be operated as a training, research and demonstration facility, with the possibility to connect it with the district heating network of the local municipal utility (WNN, 2022).

In Canada, Global First Power's demonstration project with USNC is underway. This project proposes constructing and operating a 5-15 MWe Micro Modular Reactor plant at the CNL Chalk River campus that would serve as a model for future SMR deployments to support remote and industrial applications (CNL, 2023).

In the United States, the US DOE is supporting several HTGR-related projects. Some notable 2023 developments are summarized below:

- X-energy announced a deal with Dow in 2023 to deploy the Xe-100 at a site in Seadrift Texas (X-energy, 2023). X-energy also signed a joint development agreement with Energy Northwest for up to 12 Xe-100 reactors in central Washington State (WNN, 2023d). X-energy was awarded a contract with the US Department of Defense to develop the transportable microreactor design (ANS, 2023).
- Kairos Power announced plans to fabricate fuel for the Hermes reactor in collaboration with Los Alamos National Laboratory in New Mexico (NEI, 2023). The US NRC completed a safety evaluation of the Hermes reactor and issued a construction permit (Kairos Power, 2023). The reactor is to be built in Oak Ridge, Tennessee.
- BWXT is developing both commercial and defense-related microreactors. Efforts include plans to perform an irradiation test of new TRISO fuel designs in the INL advanced test reactor (NRIC, 2023). It has also announced a contract with the Wyoming Energy Authority to assess deployment of small reactors (BWXT, 2023).

- USNC continues to develop its Micro Modular Reactor and unique TRISO/FCM fuel (USNC, 2023a). It has announced an agreement with Framatome to establish a joint venture to fabricate fuel for advanced reactors, including Micro Modular Reactors (USNC, 2023b). Earlier in 2023, USNC provided advanced TRISO particles (uranium nitride kernel and zirconium carbide [ZrC] coating) to NASA for nuclear propulsion applications (USNC, 2023c).
- Westinghouse is developing its eVinci microreactor, powered by TRISO fuel. This was one of three designs selected by DOE for experimental design activities to support potential testing in the Demonstration of Microreactor Experiments test bed at the Idaho National Laboratory (DOE, 2023). Westinghouse also announced that it has begun construction of an eVinci manufacturing plant in Pennsylvania (Neutron Bytes, 2023), and that the first eVinci customer is to be the Saskatchewan Research Council in Canada (Patel, 2023).

Technical highlights – Fuel and fuel cycle project

The Fuel and Fuel Cycle Project Management Board (PMB) annual meeting was held remotely on 19-20 July 2023. The PMB again welcomed representatives from Canada and the United Kingdom as observers. Both countries are actively pursuing full membership in the PMB.

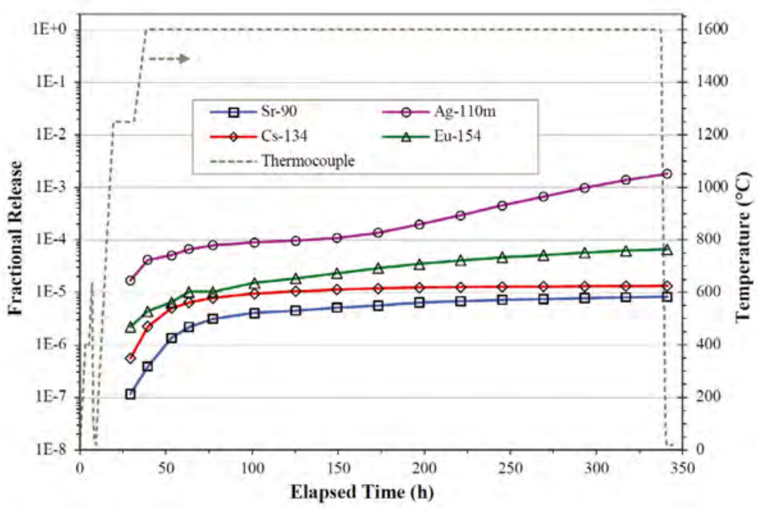
The fuel and fuel cycle project is intended to provide demonstrated solutions for VHTR fuel (design, fabrication and qualification) and its back-end management, including novel fuel cycle options. TRISO-coated particles, which are the basic fuel concept for the VHTR, need to be qualified for relevant service conditions. Its standard design – a UO_2 kernel surrounded by successive layers of porous pyrocarbon, dense inner pyrocarbon, SiC and an outer pyrocarbon – could evolve along with the improvement of its performance through the use of a UCO 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, safety testing, fission product release evaluation, and the measurement of chemical and thermomechanical material properties in representative conditions will feed a fuel material database. Further development of physical models will enable the assessment of in-pile fuel behavior 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 or symbiotic cycle.

The US TRISO fuel development and qualification program continues to progress toward completion after 20 years of dedicated effort. The post-irradiation examination and high-temperature safety testing for the final UCO TRISO fuel qualification experiment (AGR-5/6/7) are currently in progress. Several safety tests were performed in 2023 to assess fission product release at 1 600°C (Figure VHTR-2) and 1 800°C. In addition, destructive examination of fuel compacts and particles has been performed, including deconsolidation leach-burn-leach analysis, individual particle gamma counting to assess fission product release, and particle microanalysis. The program also reached a milestone with the completion of all experimental work in 2023 related to the AGR-3/4 fission

Figure VHTR-2: Preliminary results from the safety test of AGR-5/6/7 Compact 5-5-3 at 1600°C



Note: Release of fission products as a function of time is shown as a fraction of total calculated fuel inventory.
Source: Demkowicz (2023).

product transport irradiation experiment. The major task remaining for AGR-3/4 is extensive data analysis to refine the existing understanding of fission product transport behavior.

In China, post-irradiation examination on discharged HTR-10 fuel and fuel spheres irradiated previously in Petten is in progress to assess fuel performance. This includes deconsolidation and leaching followed by burnup measurements.

Efforts by several PMB members are continuing to develop post-irradiation fuel safety testing capabilities. This includes progress at the Institute of Nuclear and Nuclear Technology (INET) in China to deploy the KÜFA furnace system for high-temperature testing of irradiated fuel spheres in pure helium. In the United States, work is continuing to deploy the Air and Moisture Ingress Experiment system that will be used to assess the fission product retention of irradiated fuel specimens in oxidizing atmospheres. Testing of the system has been performed in the laboratory with inert specimens, and the system is expected to be operational in the hot cell in 2024. In separate work, KAERI is developing a computational model of air and moisture ingress into an HTGR core to calculate graphite and fuel oxidation rates.

The leach-burn-leach round robin collaborative experiment performed within the fuel and fuel cycle project is drawing to a close. The objective of this study is to compare the results of this fundamental TRISO fuel quality control measurement method as obtained by several participants using their in-house procedures. All participants (China, Korea, United States) have completed the experimental work associated with the round robin. The United States has completed a final report on its results, while China and Korea are currently preparing similar reports to submit to the GIF database.

The development of advanced TRISO fuel particles is being pursued by several PMB members. KAERI plans to develop capabilities to fabricate coated particles with alternate (i.e. non-UO_2) kernels, including UCO (to achieve higher burnup compared to UO_2), and UC and UN (for nuclear thermal propulsion applications). Initial trials have been carried out using CeO_2 (cerium oxide) and ZrO_2 (zirconium oxide) as surrogates. KAERI is also performing compacting studies to improve the properties of cylindrical fuel forms. In China, INET is developing zirconium carbide coating technology as a potential substitute for the conventional SiC coating layer.

Technical highlights – Hydrogen production project

The VHTR hydrogen production project is focused on researching, developing and optimizing high-temperature thermochemical and electrolysis water splitting processes. It is working to define and validate technologies and processes for coupling Gen IV systems to these processes, including through developing process flows, identifying suitable materials and catalysts, and conducting experiments to validate process parameters. The VHTR Hydrogen Production PMB has increased connections with the GIF Non-Electric Applications of Nuclear Heat (NEANH) Task Force to inform opportunities for Gen IV reactor systems to support hydrogen production. Individual countries in the hydrogen production project have been engaging with industry.

In 2023, KAERI in Korea established the Alliance for Nuclear Heat Utilization to develop technologies and create a network of stakeholders related to the development and use of nuclear heat for industrial processes. The alliance includes chemical companies, a steel company and other potential industrial end users of hydrogen.

Throughout 2023, the hydrogen production project has continued to advance research in this area, with a growing emphasis on accelerating and upscaling hydrogen production through high-temperature steam electrolysis (HTSE) technologies, which produces more hydrogen per energy input than low temperature water electrolysis (LTE) systems. In Korea, an integral test facility with an experimental helium loop has been developed to explore coupling HTSE with an HTGR reactor. In 2023, high-temperature steam was supplied at over 800°C to complete tests on a 3 kilowatt-electric (kWe) solid oxide electrolyzer cell (SOEC) stack, with progress towards a 6 kWe stack in the near future. The helium loop system can scale to 2 MW applications of SOEC hydrogen production to serve as a test bed for multiple partners. KAERI is also advancing research in support of design and licensing for hydrogen production through LTE and HTSE using existing light water reactors.

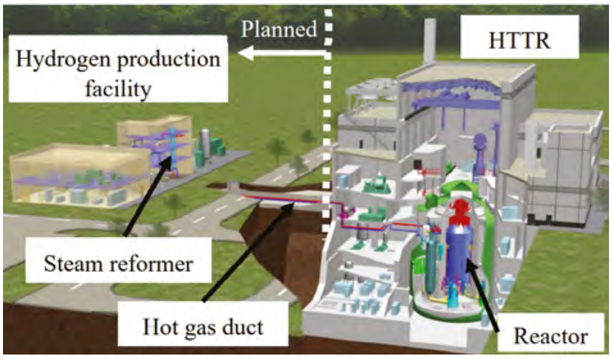
In France, progress on HTSE development includes techno-economic modelling, and testing and upscaling lab-scale electrolyzer cells, including HTSE modules that would ultimately support multi-stack, scalable hydrogen production assemblies. The CEA successfully developed and validated cells in the 0.8 to 0.9 Amp/cm^2 range at 1.3 Volt and 700°C to produce hydrogen at a daily rate of 10 kg. After completing a research campaign on the copper-chlorine hydrogen production process, R&D activities in Canada related to hydrogen production are now focused on HTSE technologies. CNL R&D in 2023 included: infrastructure and techno-economic assessments; broad research regarding safety and storage of hydrogen; and process integrations related to the coupling of hydrogen production processes with the production of hydrogen-derived products such as synthetic diesel and other synfuels.

In the United States, materials research and long-term stability testing is being performed for HTSE systems, including the completion of pressurized solid oxide electrolysis cell button testing. Testing of commercial systems is also underway, and a 100 kW HTSE system is currently being tested as a precursor to a larger facility to test HTSE at the megawatt scale. There are also multiple projects moving towards MW-scale production of hydrogen in partnership with industry through electrolysis and the existing nuclear reactor fleet, including the Nine Mile Point nuclear power plant that began producing hydrogen through a 1.25 MWe LTE system in February 2023.

Through the EU's HySelect research project, the electrochemistry of the hybrid sulfur cycle process was advanced in 2023, with contributions from partners in Austria, Finland, Germany, Greece and Italy. Additional research activities in the EU have focused on scaling up a range of technologies related to clean hydrogen production, including biomass gasification for distributed hydrogen production, thermochemical water splitting technologies, photo-electrochemical and photocatalytic water splitting, and other biological and bioelectrochemical hydrogen production processes.

The JAEA in Japan is working on coupling its operating sulfur-iodine process hydrogen production test facility to the HTTR, which has successfully operated at temperatures of 950°C. In 2023, a rotative process was developed to improve separation efficiency and effectiveness, and new membranes with silicon dioxide (SiO_2) as a middle layer were developed. The JAEA is also exploring the potential to use high-temperature heat from the HTTR to produce hydrogen through a variety of processes, including steam methane reformatting and methane pyrolysis (Figure VHTR-3). The HTTR heat application test plant is expected to produce hydrogen of more than 800 Normal meter cubed per hour (Nm_3/h).

Figure VHTR-3: Layout image of HTTR heat application test plant



Source: Nomoto et al. (2023).

Technical highlights – Materials project

The materials project arrangement continued under its second amendment until the end of 2023. The third project arrangement amendment became effective on 3 January 2024 and incorporated a new project plan for technical activities and planned contributions for the period 2023 to 2027. It also added the United Kingdom and Canada as new and rejoined signatories, respectively.

The development and qualification of materials are crucial for the advancement of the VHTR system. Key challenges include irradiation-induced and/or time-dependent failure, as well as microstructural instability in operative environments. Existing materials can suffice for core coolant outlet temperatures below 950°C, while novel materials must be developed and qualified for higher temperatures and harsher conditions, including corrosive process fluids and off-normal operations. Multiscale modeling is indispensable for optimizing design methodologies. Materials are categorized into graphite for core structures and fuel matrixes, very/medium-high-temperature metals, and ceramics and composites. Emerging advanced manufacturing techniques such as additive manufacturing offer new possibilities for innovative material classes, potentially addressing the abovementioned challenges. Qualifying these methods and their resulting materials has become an integral part of the project's R&D objectives, and is fundamental to the development of codes and standards required to design and license HTGR components.

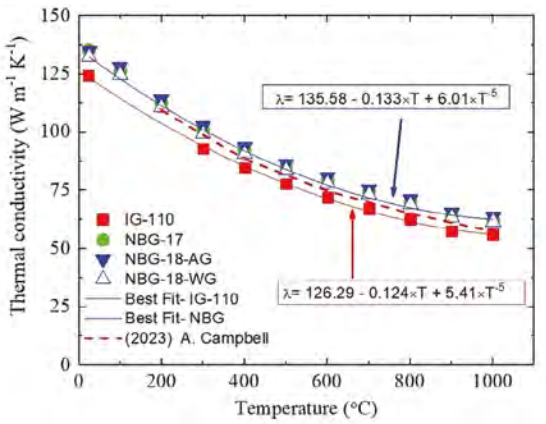
A comprehensive materials database (also known as the materials handbook) has been compiled and serves as a platform for efficient data management, international R&D coordination, and prediction of damage and lifespan through modeling. After showcasing the materials project to a broader audience in 2022 and 2023, the handbook generated significant industry interest. As a result, various options for access to the handbook were assessed, with specific focus on the development of a plan to address third-party access. By the end of 2023, a total of 519 technical reports had been uploaded into the handbook, as well as over 40 000 supporting data records.

In 2023, the project's research focus remained on near- and medium-term projects, with emphasis on developing graphite and high-temperature metallic alloys. Several signatories conducted extensive research into cutting-edge manufacturing techniques for nuclear components, such as laser fusion, metal powder consolidation and direct deposition. In addition, novel high-temperature structural materials were developed using innovative synthesis methods, and a dedicated task on advanced manufacturing methods has been incorporated into the third amendment of the VHTR materials project arrangement to further explore and develop these technologies.

Multiple members performed comprehensive characterization and analysis of baseline data for various grades of graphite, including mechanical, physical and fracture properties. Graphite samples were subjected to irradiation and post-irradiation examinations, providing valuable data on property changes, while oxidation studies investigated the effects of air and steam exposure on graphite. The effectiveness of boron coatings in mitigating oxidation damage to graphite core components was also explored. Finally, large-scale experiments were conducted to examine and validate the multi-axial loading response of graphite, specifically addressing dimensional changes and seismic events. Data to support graphite model development were generated in the areas of microstructural evolution, irradiation damage mechanisms, creep and thermal properties. Support was provided to the American Society for Testing and Materials and the American Society of Mechanical Engineers for the development of codes and standards required for the use of nuclear graphite, which continue to be updated and improved.

The CNL has conducted studies on the thermal properties of pristine nuclear graphite grades. As shown in Figure VHTR-4, its investigation includes a comparative thermal conductivity analysis of various grades, featuring best-fit functions for both IG-110 and NBG. This research provides very useful information on the thermal performance of these materials, which is crucial for optimizing their in-reactor use. The examination of high-temperature alloys has provided valuable insights into their suitability for use in heat exchanger and steam generator applications. This evaluation involved extending the existing database through a range of tests, including aging, creep, creep fatigue, and creep crack growth rate testing up to 950°C for alloys 800H and 617. In addition, welding studies were conducted on 617, 800H, and dissimilar welds of T22 to 800H. The goal of these efforts was to qualify new metallic materials such as alloy 709, high entropy alloys and oxide dispersion-strengthened (ODS) alloys for use in the construction of high-temperature nuclear components. Furthermore, advancements have been made in enhanced diffusion bonding techniques for the fabrication of compact heat exchangers, which have shown promising results. Extensive modeling and testing are currently underway to pave the way for their qualification in VHTRs.

Figure VHTR-4: Thermal conductivity of different nuclear graphite grades as a function of temperature



Source: Saoudi et al. (2023).

At KAERI, experiments on diffusion-welded alloy 617 were conducted to study its behavior under low cycle fatigue conditions. The tests were carried out at various temperatures, ranging from 25°C to 950°C, with a total strain range of 0.6-1.2%. The results showed that the primary crack initiation occurred through an intergranular mode at the surface, while the crack propagated away from the interface. Crack branches formed during propagation, resulting in a mixed fracture mode near the crack tip.

The Australian Nuclear Science and Technology Organisation (ANSTO) and INL developed data-driven models employing regularized linear regression (RLR) and recurrent neural network (RNN) architectures. These models are designed to predict the fatigue life of alloy 617. According to a study published in the International Journal of Pressure Vessels and Piping (Molina et al., 2023), these data-driven models have demonstrated potential accuracy superior to some established empirical models; however, it is important to note that their predictive accuracy diminishes significantly when applied to scenarios outside their training datasets.

Technical highlights – Computational methods validation and benchmarks project

Validation of new computational methods and codes in the areas of HTGR 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. The data are generated by the current HTTR, HTR-10 and HTR-PM operational 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.

At the INET in China, the development of a state-of-the-art HTGR design software package continued in 2023. New models and capabilities were added to the reactor physics code PANGU, including improved resonance and double-heterogeneity treatment, kinetics parameters, random fuel shuffling, and diffusion coefficient calculations. A new 3D system analysis code DAYU3D was also developed. These codes are used as design verification tools for the larger future HTR-PM600 and HTR-PM1000 projects, and are expected to replace the legacy German codes after licensing from the National Nuclear Safety Administration.

In Japan, the JAEA's R&D code and calculation methodology developments are expected to contribute to computational methods, validation and benchmarking activities, such as a reactor physics benchmark activity using US Advanced Test Reactor (ATR) TRISO irradiation data. As part of this benchmark activity, JAEA has been developing a model of the ATR using the JAEA-developed Monte Carlo simulation code MVP (Nagaya et al., 2017).

The VHTR R&D program in Korea aims to improve the design code development and assessment of very high-temperature system key technologies. Activities in 2023 included a coupled calculation (CAPP/GAMMA+) to simulate pressurized conduction cooldown, depressurized conduction cooldown, water-ingress, load following operation and air-ingress transients. An effective homogenized cross section method with explicit TRISO model was also developed to treat the spatial self-shielding effect for a double heterogeneous region, and the method was implemented in the two-dimensional transport lattice code DeCART2D_HTR. A TRISO particle failure model was also investigated for air-ingress and water-ingress conditions, and the model was implemented in the fuel performance analysis code COPA.

In the European Union, the Horizon Europe Framework Program project, GEMINI 4.0, (GEMINI For Zero Emission) has completed the first 18 months of activities related to HTGRs (scheduled until May 2025). The focus areas are optimization of safety and competitiveness, decarbonization of industry, fuel and fuel cycle, licensing readiness, and socio-economic impact. An assessment of the licensing readiness of the GEMINI HTR design was completed, which identified design and safety aspects to be studied and improved. Other studies include the assessment, suitability and completeness of codes and standards in Europe relevant to HTGR deployment, and investigations into fuel cycles and back-end solutions.

The US DOE is currently funding several activities related to the production of experimental validation data and software for the analysis of HTGRs. The Natural Convection Shutdown Heat Removal Test Facility experiment is currently producing valuable data on off-normal conditions for a water-based reactor cavity cooling system (Qiuping, et al., 2023). The "Thermal hydraulic code validation benchmark for high temperature gas-cooled reactors using High Temperature Test Facility data" is now an official NEA benchmark, based on a subset of high temperature test facility data that will be used for system and CFD codes validation (Kile, 2023). A Serpent Monte Carlo-based running-in methodology was integrated into the Pronghorn thermal-hydraulic core analysis to enhance user capabilities with multiphysics functionalities (Stewart et al., 2023), and two fluoride- and gas-cooled pebble-bed reactor models were developed to assess TRISO fuel performance during a control rod withdrawal accident (Labouré et al., 2023a). A comprehensive HTTR gas-cooled pebble-bed reactor model was also developed for code-to-experiment comparisons (Labouré et al., 2023b).

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

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4 Working group reports

Advanced Manufacturing and Materials Engineering Working Group

Working group goals and industry engagement

The Advanced Manufacturing and Materials Engineering Working Group (AMME-WG) promotes the use of advanced manufacturing and materials engineering to reduce the time to deployment of Gen IV reactors. Specifically, it aims to promote international collaboration on the qualification of advanced materials and manufacturing processes for use in Gen IV reactors. The working group's work spans all GIF reactor systems, with a focus on crosscutting technologies that could impact multiple reactor technologies.

The group develops projects in four areas, each with separate but integrated project teams (Figure AMME-1):

1) Survey: Conduct a regularly updated survey on recent applications to maintain a direct industry connection for gauging current sentiment towards advanced materials and manufacturing technologies. The results of this survey help to guide the working group on the most relevant manufacturing techniques and materials.

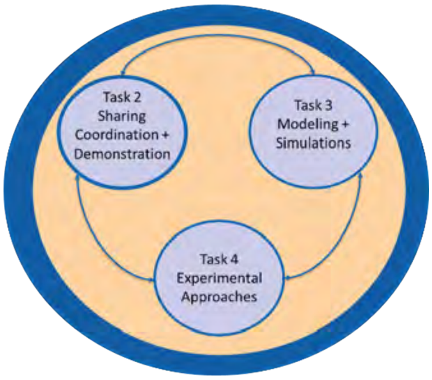
2) Sharing Forum: Manage a forum for sharing, coordination, and demonstration of work on the qualification of advanced materials and manufacturing techniques. This supports technology readiness and therefore minimizes duplication in activities to accelerate the deployment of advanced manufacturing technologies beneficial to the Gen IV community. It also provides another venue for receiving industry feedback.

3) Modeling and simulations: Develop activities and projects to promote the qualification of advanced manufactured systems, components and structures through the innovative use of modeling and simulation.

4) Experimental approaches: Develop activities and projects to promote the qualification of advanced manufacturing systems, components and structures through innovative testing and monitoring techniques.

The GIF activities in these topic areas will help to maintain close connections with industry, with specific technical projects also being performed in collaboration with the relevant system steering committees and project management boards. The working group's goal is to maintain or increase the frequency of international stakeholder interactions in the future.

Figure AMME-1: Advanced Manufacturing and Materials Engineering Working Group project areas



2023 accomplishments

Since forming as a task force in 2018, the AMME crosscut collaboration on advanced materials transitioned to a permanent GIF working group in 2023. The Experts Group and Policy Group approved the transition at the October 2023 meeting. The working group continues to build on the accomplishments of the task force (see Figure AMME 2).

As part of the transition to a permanent working group, the task force prepared a formal terms of reference document, currently available on the GIF website. This document includes a detailed work plan for how the task force will develop new technical projects and engage with industry, in addition to formal operating rules (GIF, 2023).

Project teams for the four topic areas developed an initial breakdown of the activities and timelines, while building the teams. The first two project teams build on the fundamentals of task force collaborations started in 2023 with an industry survey and collaboration with other stakeholders and technical working groups.

The AMME-WG has prepared and started distributing an updated industry survey on the applications of advanced materials in Gen IV reactors. This is the third iteration of the survey, with the previous two versions released in 2019 and 2021. The survey is distributed to a large group representative of industry and the R&D community across the GIF member countries. The survey tracks changes in sentiment towards particular manufacturing technologies, materials and applications over time. Previous surveys found a narrowing of industry focus towards specific technologies and a broad support for collab-

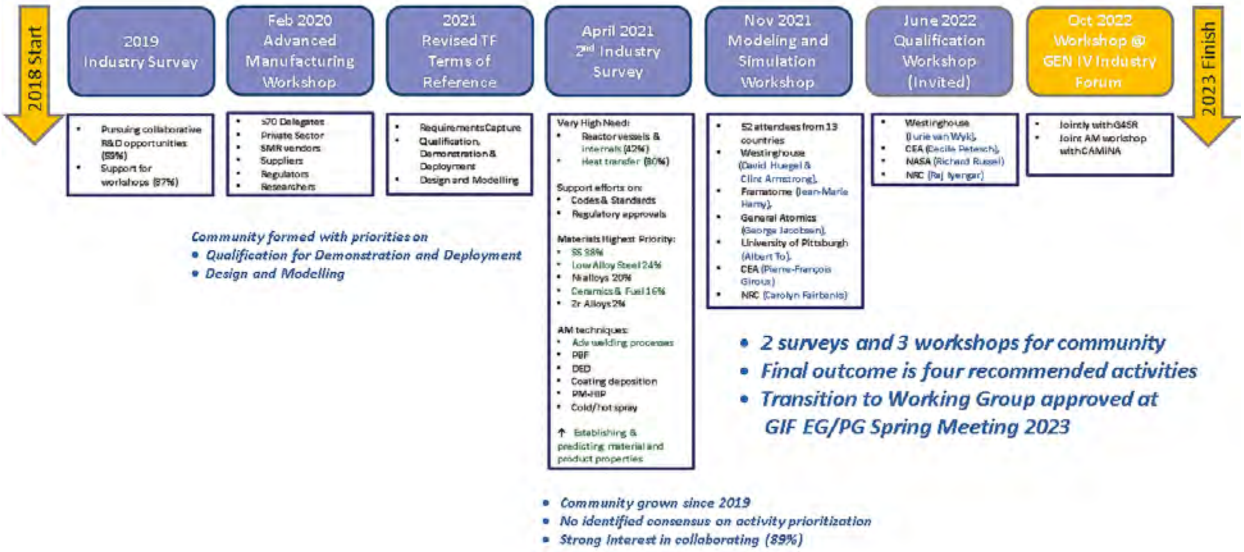


Figure AMME-2: History of the Advanced Manufacturing and Materials Engineering Working Group

orative R&D activities to mature and qualify materials and manufacturing processes. The AMME-WG will update the survey results in early 2024.

The AMME-WG also developed a calendar of technical networking opportunities (conferences, meetings etc.) to facilitate participation and the subsequent sharing of information obtained from these interactions.

The task force participated in the 17th GIF-IAEA Interface Meeting in Vienna in July 2023. The main outcome of this meeting was the addition of an IAEA observer to the task force membership, with the goal of fostering future collaboration. This interaction continues as part of the AMME-WG meetings, and for individual project team meetings.

Information regarding the AMME-WG was presented at the INNUMAT Workshop on Qualification, 16-17 November 2023 in Madrid Spain; the Workshop on Materials for Generation IV Reactors, 27-29 November 2023 in Madrid Spain; and the Working Party on Materials Science Issues in Nuclear Fuels and Structural Materials (WPFM), NEA, 19 December 2023 in France (virtual).

A dedicated session on Advanced Manufacturing and Material Engineering was hosted by the CEA and the GIF SIAP at the World Nuclear Exhibition 2023 on 30 November 2023 in Paris, France. After the presentation on the newly formed AMME-WG, members consulted with industry participants and distributed the survey details.

Lyndon Edwards
Co-Chair of the AMME-WG, with contributions from AMME members

Mark Messner
Co-Chair of the AMME-WG, with contributions from AMME members

Isabella van Rooyen
Co-Chair of the AMME-WG, with contributions from AMME members

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Economic Modelling Working Group (EMWG)

The EMWG was established in 2003 to provide a methodology for the assessment of Gen IV systems against two economic-related goals:

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

The EMWG has published *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* (GIF, 2007) and released the Excel-based software package, G4ECONS v3.0, that provides the means to calculate the levelized cost of energy and total investment cost, and therefore evaluate Gen IV systems against GIF economic goals. These resources are made available to the public through the GIF Technical Secretariat. Subsequent publications have demonstrated the EMWG methodology for the economic assessments of Gen III and Gen IV systems, and enabled benchmarking against economic models developed by the IAEA, including the Nuclear Economics Support Tool and the Hydrogen Economic Evaluation Program.

In 2018, the terms of reference for the EMWG were amended to incorporate the expanded mandate 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 launched a survey to collect user feedback on G4ECONS and identify potential model improvements to be considered for version 4.0. 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. Many users also expressed interest in additional training sessions on the G4ECONS tool.

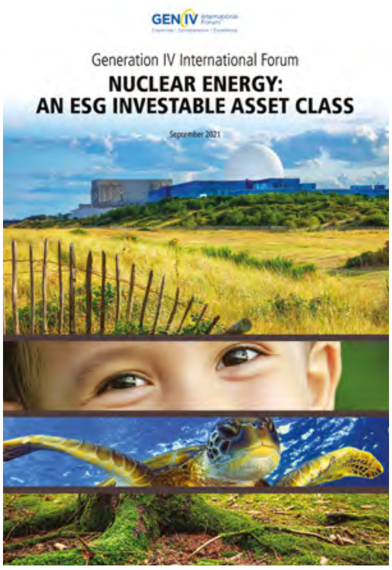
In 2021, the EMWG worked with the finance industry to produce the report, *Nuclear Energy: An Environmental, Social, and Governance Investable Asset Class* (GIF, 2021). This report considered the nuclear industry's ability to report against a broad range of environmental, social and governance data collection and accounting metrics.

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" barriers to simplify radionuclide retention. In 2022, a second ANTSER cost reduction strategy was developed for "modularity at scale" (GIF, 2022).

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 how the impact of the way in which projects are established can considerably influence their financing. In 2022, the EMWG agreed to undertake an update to the Cost Estimating Guidelines for Generation IV Nuclear Energy Systems document and the G4ECONS tool. A small team of EMWG members was formed and the table of contents was drafted to be shared with others within the EMWG (SIAP and NEANH) to receive early feedback.

EMWG activities in 2023

In 2023, the EMWG drafted the third instalment of the ANTSER framework focusing on technology-inclusive cost reduction, which is a strategy that aims to identify the interrelated components of a nuclear system so that system-wide cost impacts



can be measured following a change in one part of the system. The strategy begins with a description of the requirements for different reactor technologies, safety features and engineering approaches that can affect the cost of nuclear energy. To illustrate the approach, the report uses a case study that describes how structural layouts impact costs, particularly when reactor technology allows for the construction of less demanding structural designs with smaller footprints. Looking at the design of containment walls, the case study shows that even if designing containment buildings increases the total cost, there are trade-offs that can result in cost savings in other parts of the system that may offset the incremental increase in costs from wall thickness.

The EMWG also engaged with the PRPPWG to assess the impact of safeguards and security costs on Gen IV reactor economics, and with the NEANH task force regarding techno-economic analysis of non-electric or hybrid Gen IV systems.

In September 2023, the EMWG financing task force was remobilized and expanded to include more people from the global finance industry. The task force will be looking at several issues, including updating the environmental, social and governance report and considering the financing requirements of a first Gen IV reactor demonstration project.

Also in 2023, the EMWG progressed the update to the cost estimating guidelines by completing a review of current state-of-the-art approaches documented in the literature. It produced a first draft of the updated guidelines, which includes expanded discussions on scale and modularity to better address SMRs, and further explorations of how to best provide cost estimates for direct steam use and the co-generation of heat and electricity. This work was presented to the SIAP in July 2023.

Work will continue in 2024, with a shift to review the cost estimating guidelines and the development of G4ECONS v4.0. 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 existing and new users of these EMWG tools.

Megan Moore
Co-Chair of the EMWG

Fiona Reilly
Co-Chair of the EMWG

Jason Hansen
Co-Chair of the EMWG

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Education and Training Working Group (ETWG)

The GIF ETWG started in November 2015 as a task force, since then it has organized, offered, recorded and archived 84 webinars, with 12 new webinars presented in 2023 (Webinars can be found on the GIF portal: www.gen-4.org/gif/jcms/c_84279/webinars). An international online panel discussion entitled "International Knowledge Management and Preservation on SFR" took place in June 2023 and discussed lessons learned 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 to companies planning on building SFRs ranging in power from 300 to 1 200 MWe. The ETWG launched a virtual "Pitch your Gen IV Research (PyGR)" competition in December 2022, and the three winners are invited to present a GIF webinar in 2024. The ETWG has also started an initiative on Knowledge Management and Knowledge Preservation (KMKP).

2023 Pitch your Gen IV research competition

In December 2022, the ETWG launched the 2023 Pitch your Gen IV Research competition to increase junior researchers' engagement in the Gen IV community and stimulate their interest in advanced reactor systems. This competition also informed the public about Gen IV reactors and related topics, and gave them an opportunity to participate by voting for a preferred video/candidate/topic. The production of short informative videos available on YouTube (https://tinyurl.com/53ky2ep8) and Bilibili (https://tinyurl.com/dy48v8tm) can increase a global awareness of the nuclear energy sector. Of the 47 abstracts submitted from 11 countries, 11 were selected (Figure ETWG-1). The candidates benefited from a large and informed audience for pitching their research (Figure ETWG-2). The three winners were invited to present GIF webinars in 2024.

2023 GIF webinar series

Twelve webinars were presented live in 2023 and are archived on the GIF website (Table ETWG-1). A joint GIF/IAEA webinar, "Thorium based advanced reactor design concepts", was also organized by Tatjana Jevremovic from the IAEA and Patricia Paviet from the GIF, and presented on 5 May 2023. The webinar featured four panelists: Aslak Stubsgaard, Copenhagen Atomics, Denmark; Ritsuo Yoshioka, International Thorium Molten Salt Reactor Forum, Japan; Brittney Saenz, Alpha Tech Research Corporation, United States, and Armando Nava-Dominguez, CNL, Canada. These panelists highlighted the potential advantages of the thorium fuel cycle over the uranium fuel cycle, and the unique features in the design of advanced thorium-based reactors.

Table ETWG-1: GIF Webinars offered in 2023

Presentation title	Presenter
Molten salt reactors taxonomy and fuel cycle performance	Jiri Krepel, Paul Scherrer Institute, Switzerland
Safe final disposal of spent nuclear fuel in Finland	Mika Pohjonen and Sanna Mustonen, Posiva Solution, Finland
Advanced reactors safeguards and material accountancy challenges	Ben Cipiti, Sandia National Laboratories, United States
Overview of nuclear graphite R&D in support of advanced reactors	William Windes, Idaho National Laboratory, United States
Graphite-molten salt interactions	Nidia Gallego, Oak Ridge National Laboratory, United States
International knowledge management and preservation of sodium fast reactor	Panel session: Cal Doucette, Arc Energy, Canada; Joel Guidez, France; Hiroki Hayafune, JAERI, Japan; Patrick Alexander, TerraPower, United States; Ron Omberg, PNNL, United States
Off-gas Xenon management in support of MSR	Hunter Andrews, Oak Ridge National Laboratory and Praveen Thallapally, Pacific Northwest National Laboratory, United States
Corrosion and cracking of SCWR materials	Lefu Zhang, Shanghai Jiao Tong University, China
EPRI virtual reality training	Robert Eller, Electric Power Research Institute, United States
The nuclear workforce of the future – Opportunities and needs for the international nuclear sector	Callum Thomas, Thomas THOR, United Kingdom
MOOK: The knowledge management method applied to a Gen IV project. The continuation of a successful story	Gilles Rodriguez, CEA, France
Characterization of U-233 for thorium fuel cycle safeguards	Madeline Lockhart, North Carolina State University, United States

Figure ETWG-1: Pitch your Gen IV Research competition: Abstracts submitted

Total number of summaries	47
Students	37
Postdoc	5
Researcher/engineer	5

Country of PhD Institution	Summaries
China	11
USA	10
France	7
UK	7
South Korea	5
Japan	2
South Africa	1
Hungary	1
Spain	1
Nigeria	1
Belgium	1

Systems	Summaries
SFR	1
LFR	2
MSR	11
SCWR	1
SFR	6
VHTR	10
FR	12

Topic	Summaries
Modeling/Simulation	9
Chemistry	2
Material	12
Fuel	9
Design	11
Micro-reactor	2
Other	2

Figure ETWG-2: Pitch your Gen IV Research competition: Videos accessible on the GIF portal

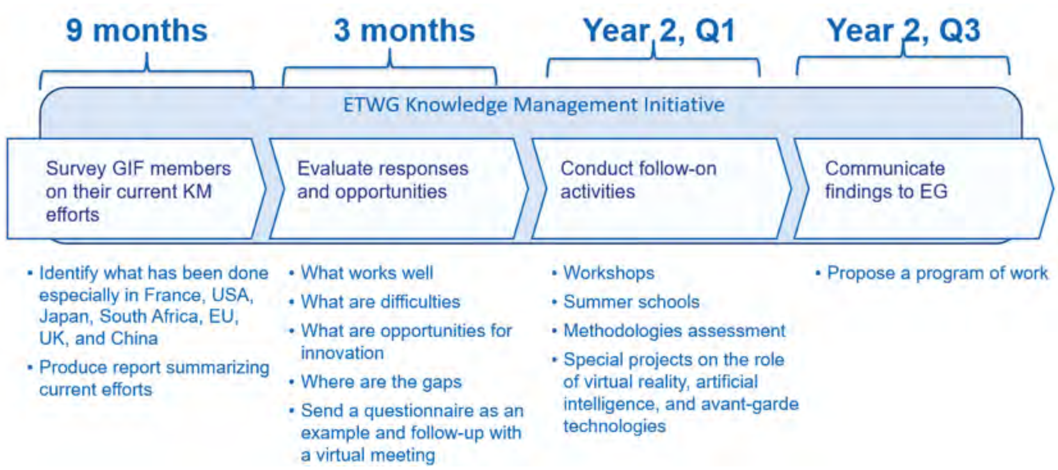
As of December 2023, attendance during the live webcasts totaled 7 827, with the number of viewings of recorded webinars reaching 7 459. In seven years, the webinars have been viewed live or recorded 15 286 times, and have reached scientists and engineers across 81 countries.

Knowledge management and knowledge preservation (KMKP)

Knowledge management has always been important in the nuclear energy field, but with the influx of the younger generation and the imminent deployment of Gen IV technologies, it is at the forefront again. The GIF community recognizes recent challenges in the preservation of advanced reactor knowledge and expertise, with a near loss of Fast Flux Test Facility data, available French SFR experience, the condensation of the international R&D community, and the retiring of facilities and knowledgeable experts. Many start-up companies are in dire need of prior experience with Gen IV technology, but the traditional methods of knowledge management (e.g. books, coursework, on-the-job training) may not be the most effective for the younger generation. Modern communication and software tools may also allow improved KMKP and archiving procedures. Several GIF members already have active KMKP programs, meaning that the GIF could play a role in communicating, supporting and augmenting national efforts in KMKP. The ETWG has proposed to undertake a multi-year initiative (Figure ETWG-3) to explore opportunities for GIF engagement in KMKP. The first step involves identifying what has been done in several GIF member countries. A report summarizing the current efforts will be issued in 2024.

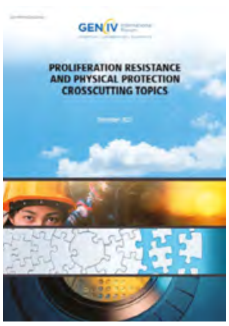
Patricia Paviet
Chair of the ETWG,
with contributions from ETWG members

Figure ETWG-3: GIF knowledge management and knowledge preservation outlook



Proliferation Resistance and Physical Protection Working Group (PRPPWG)

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 (GIF, n.d.). 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&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, 2011), with translations in Japanese and Korean produced for national use.



Since 2018, the group has been updating white papers on PR&PP for the six GIF reactor technologies, in collaboration with the six GIF SSCs and provisional SSCs. These updates reflect changes in reactor design since the issue of the original white paper (GIF, 2013). Individual white papers, after endorsement by both the PRPPWG and the responsible SSC, have been submitted to the Experts Group for approval and published as GIF documents. PR&PP aspects that transcend all six GIF systems have also been investigated. Crosscutting topics include common themes such as 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 (GIF, 2021a and, 2021b) and the GFR, SCWR and VHTR white papers published in 2022 (GIF, 2022a, 2022b, 2022c), two white papers were finalized in 2023:

- The Proliferation Resistance and Physical Protection Crosscutting topics (GIF, 2023a), available for download since March 2023 (see picture).
- The Molten Salt Reactor Proliferation Resistance and Physical Protection white paper (GIF, 2023b).

The MSR PR&PP white paper represents the last output of a major multi-year collaboration effort that saw the involvement of all six GIF SSCs, showing the excellent level of collaboration between the various GIF working groups.

The PRPPWG also maintains an annually updated bibliography of official publications referring to the PR&PP methodology, and of relevant issues. The latest edition, revision 10, was published in November 2023 (GIF, 2023c).

The PRPPWG maintains regular exchanges with the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the IAEA Department of Safeguards and the IAEA Department of Safety. Several PRPPWG members support INPRO efforts, in particular through updating the INPRO Manual on Proliferation Resistance (Scherer et al., 2023). An IAEA representative participates regularly in the PRPPWG activities. The PRPPWG made a presentation at the 17th GIF-IAEA Interface Meeting on 11-12 July 2023, highlighting the ongoing PRPPWG-IAEA collaborations on PR and safety, security and safeguards (3S) by design.

At the ESARDA/INMM Joint Annual Meeting in May 2023, the PRPPWG and the IAEA held a special panel session entitled "Generation IV Proliferation Resistance and Physical Protection: Transitioning from R&D to Deployment", where they presented recent accomplishments of the group along with future activities aligned with transitioning from research and development to deployment. A summary of the discussion is available in the proceedings (Cipiti et al., 2023).

Collaboration with the Risk and Safety Working Group (RSWG) continued through exchanges at group meetings: PRPPWG representatives attended the 2023 RSWG semi-annual meetings, and RSWG representatives participated in the 33rd PRPPWG annual meeting.

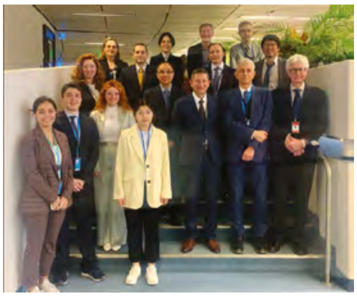


Photo from 33rd PRPP Annual Meeting held at IAEA headquarters in Vienna, 25-27 January 2023.

An important activity in 2023 was collaboration with the RSWG and the Very High-Temperature Reactor SSC on a case study to identify 3S interfaces on a generic VHTR pebble-bed small modular reactor design. The goal is to draw generalized conclusions on how these interfaces can be analyzed in a technology-neutral, bottom-up approach. The insights gained from this study will help in implementing 3S by design for reactor implementations in the near future. The current status of the activity has been presented in several workshops and meetings throughout 2023,[1] and received positive feedback and expressions of interest in the expected outcome.

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 the economic analysis of Gen IV

1. For example, van der Ende, B., "Joint RSWG/PRPPWG/VHTR-SSC effort for establishing safeguards, safety, and security interfaces for VHTR system", *IAEA Interregional Workshop on Safety, Security and Safeguards by Design in Small Modular Reactors (SMRs) (general issues)*, hosted by the government of the United States of America through the Idaho National Laboratory, 11-15 September 2023.

reactor systems. In April 2023, a representative of the PRPPWG attended the EMWG meeting, presenting the content of the PR&PP crosscutting topics report and discussing its relevance for potential collaborations.

As a follow-up on the suggestions given to the PRPPWG at the G4SR-4 and GIF Industry Forum in Toronto, Ontario, 3-7 October 2022, the group started work on a report examining PR&PP implications of siting options for SMRs and microreactors. The report will have a similar format to the PR&PP white papers and will cover several different siting options (i.e. remote locations, near industrial complexes and population centers, floating options), together with some relevant crosscutting considerations. The activity will be one of the priorities for the 2024 workplan, together with the 3S interfaces case study.

Over the last two years, a review of the current PR&PP evaluation methodology highlighted how the PP part might benefit from an update of the considered measures and metrics. The update activity was decided at the 33rd PRPPWG meeting, and was followed by a dedicated internal workshop where a new framing for the PP measures and metrics was proposed and discussed. The group will continue the revision during 2024, and will report the outcome to the Experts Group and Policy Group.

The PRPPWG holds monthly teleconferences to report on the progress of group and member activities, with the summary records filed in the PRPPWG archive for documentation and retrieval. The 33rd Annual meeting was held in Vienna on 25-27 January 2023, and hosted by the IAEA (see picture). It provided an excellent opportunity to exchange information with the IAEA on topics of mutual interest and potential areas for strengthening collaboration and synergies between PRPPWG and IAEA activities. The group will hold its 34th annual meeting in Paris in March 2024, hosted by the NEA.

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Risk and Safety Working Group (RSWG)

The RSWG was formed in 2005 to promote a consistent approach to safety, risk and regulatory issues among Gen IV systems. The RSWG also advises the GIF Experts Group and Policy Group on matters such as:

- Gen IV safety goals and evaluation methodologies to be considered in design and R&D programs.
- Interactions with the nuclear safety regulatory community, the IAEA and relevant stakeholders.

In 2023, 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 Department of Nuclear Safety and Security also participated as an observer.

The Integrated Safety Assessment Methodology (ISAM) was developed by the RSWG as a technology-neutral toolkit that supports design, safety and risk evaluation. The RSWG has subsequently coordinated with the SSCs to apply ISAM to select Gen IV designs, as documented in a series of white papers and system safety assessments. These documents are available on the GIF RSWG web page (GIF, 2023a).

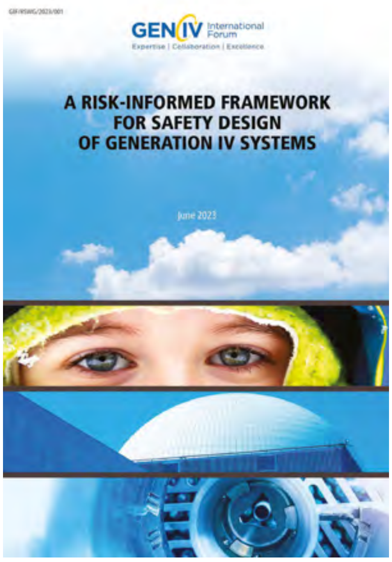
The RSWG and SSCs have also collaborated on the development of safety design criteria (SDC) and safety design guidelines (SDG) for specific Gen IV systems. These SDC and SDG are intended to bridge the gap between the high-level GIF safety goals and country-specific codes and standards by establishing minimum requirements for design, fabrication, construction, inspection, testing and operation of the first Gen IV reactor demonstrations. Although initiated by a dedicated task force to produce SFR SDC, the effort has subsequently been included within RSWG's activities and extended to other Gen IV systems.

The SDC and SDG are extensively reviewed. In 2023, significant effort was made to address GIF Experts Group comments on the second SFR SDG on systems, structures and components, as well as to address comments on the LFR SDC received from the IAEA and the NEA Working Group on New Technology (WGNT). Resolutions to these comments are being evaluated by the RSWG prior to publication of the revised documents.

The first VHTR SDC report was also published in 2023 (GIF, 2023b). The VHTR SDC were developed by an informal subgroup of RSWG and VHTR SSC members using the IAEA safety standard SSR-2/1 Rev. 1 (IAEA, 2016) as a starting point. Compared to IAEA SSR-2/1, the VHTR SDC clarifies requirements for design extension conditions, the confinement function and decay heat removal (DHR) systems, while also introducing new requirements for coated particle fuel, non-electric applications and multi-module plant designs. The WGNT have agreed to review the current published version of the VHTR SDC, and the RSWG is awaiting its feedback.

A risk-informed framework for the safety design of Gen IV systems was also published as a position paper (GIF, 2023c). The approach integrates deterministic inputs and risk insights to help select design basis accidents and design extension conditions, as well as classify plant equipment according to its risk significance and role in plant safety. The frequency-consequence structure used is consistent with regulatory requirements. In 2023, the RSWG addressed comments on the position paper to the satisfaction of the GIF Experts Group and Senior Industry Advisory Panel prior to publication.

Coordination of activities with the NEA Committee on Nuclear Regulatory Activities continued in 2023. RSWG and VHTR SSC members presented a



summary of the VHTR SDC at a meeting of the WGNT (a successor to the Working Group on the Safety of Advanced Reactors). The WGNT has subsequently agreed to review the current published versions of the VHTR SDC, as well as the risk-informed framework position paper.

The RSWG also maintained a direct technical interface with the IAEA. This relationship was reaffirmed during a "deep dive" on safety-related collaboration opportunities at the annual GIF-IAEA Interface Meeting held on 11-12 July 2023 in Vienna, Austria. RSWG members delivered several presentations at the joint IAEA-GIF Workshop on the Safety of Non-Water Cooled Reactors (30 May to 2 June 2023, Vienna) on topics including the development of the SFR SDC and SDG, a trial implementation of the risk-informed framework to SFR, and the newly completed VHTR SDC. The RSWG also represented GIF in the IAEA's Nuclear Harmonization and Standardization Initiative Industry Track Topic 3 on experiments and safety analysis code validation. The RSWG co-chair joined the Topic 3 Industry Group as the designated representative of GIF and participated in several consultancy meetings to help steer the direction of the initiative.

The RSWG continued to collaborate with the PRPPWG and the VHTR SSC on the 3S interface investigation. The objective of this activity is to generate new insights on the interaction between the 3S using the VHTR system as a basis for a case study. In 2023, RSWG members reviewed the safety attributes of a design concept provided by the VHTR SSC against the VHTR SDC, as well as an example application of the risk-informed approach to a high-temperature gas-cooled reactor, to identify possible initiating events and to confirm the completeness of the generic design description for the 3S interface assessment.

Looking ahead to 2024, the RSWG is beginning two new investigations: 1) mechanistic source term assessment methodologies; and 2) identification of practically eliminated situations. These topics were identified as priority areas for investigation by RSWG members, and in 2023 the GIF Experts Group and Policy Group endorsed both to proceed. Terms of reference for the new activities were prepared and subsequently reviewed by RSWG members. The first step will see the distribution of questionnaires to experts (including, but not limited to, RSWG members) to establish the international state of the art for each topic. The feedback received from the completed questionnaires will be used to direct the assessments, with each investigation targeted towards the production of a summary publication within two years. These topics were also introduced to the IAEA and WGNT, and both expressed interest in early coordination with the RSWG.

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5 Task force reports

Non-Electric Applications of Nuclear Heat Task Force (NEANH TF)

Task force goals

Reducing CO_2 emissions solely through the decarbonization of electricity generation is not adequate to achieve the ambitious CO_2 reduction goals set by many nations and commercial entities around the world. There is an opportunity for further emissions reduction in the industrial and transportation sectors by directly utilizing heat generated from nuclear sources, as well as through the production of process intermediates that can be created using heat and electricity generated by nuclear fission.

Non-electric applications of nuclear heat include the solutions and processes that make optimal use of the energy produced by a nuclear fission reactor – all or part of the heat it produces over the extent of operational temperature and power levels – to provide alternatives to using fossil fuels as a source of thermal energy, as well as energy services that are complementary to the electric grid. This approach is designed to optimize energy utilization efficiency, system economics and decarbonization.

There is historic precedent for the use of nuclear technologies for non-electric applications: 43 reactors have been used for district heating, totaling approximately 500 reactor years of experience; 17 reactors have been used for desalination, totaling approximately 250 reactor years of experience; and seven reactors have been used for industrial process heat.

While past experience using nuclear energy for non-electric applications mostly used water-cooled reactors, there is an opportunity to expand the role of nuclear energy in supporting non-grid applications with the emerging development of Gen IV reactor technologies in areas such as chemicals production, mining applications, reducing emissions in the oil and gas sector, and for the production of low-carbon intensity hydrogen for a range of applications.

The NEANH TF was formed in 2021 to address this opportunity. It also considers the role of Gen IV systems to support hybrid applications operating alongside other clean energy generators to flexibly provide reliable power to the grid, while also contributing clean energy to a range of industrial applications.

In particular, the NEANH TF aims to improve the general level of knowledge of non-electric or hybrid energy systems among GIF members and the general public. The task force provides an expert view of relevant solutions and shares expertise beyond the nuclear field to highlight the role of nuclear energy systems as a key contributor to a clean energy future. A current focus is to support the techno-economic analysis of these coupled systems using available information to understand the potential for Gen IV systems to support industrial heat demands and other non-grid applications.

Demonstration and industry engagement opportunities

The NEANH TF is working with partner organizations and initiatives to enhance the knowledge of potential non-electric or hybrid energy applications to support near-term system demonstrations. Engagement with industry is a key focus as it ensures that energy end users understand the opportunity to use Gen IV systems to support the decarbonization of their sectors.

The efforts of the task force are complemented by the demonstration projects of NEANH systems around the world, including:

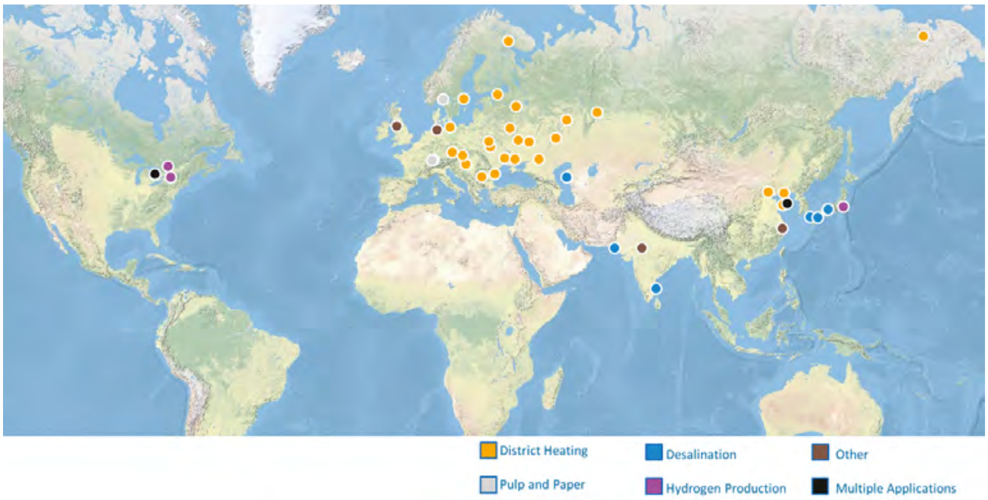
- The Tianwan plant in China, which is aiming for steam integration into petrochemical manufacturing by 2023.
- Hydrogen production demonstrations at the Nine Mile Point, Davis-Besse and Prairie Island nuclear power stations in the United States by 2024.
- Various international projects across Argentina, Canada, Japan, Korea and the United Kingdom to showcase the application of Gen III and Gen IV technologies in desalination, hydrogen production and chemical manufacturing processes.

These projects exemplify the task force's goal to accelerate the deployment of advanced nuclear technologies in non-electric applications to sustainably meet global energy needs.

NEANH TF highlights in 2023

Throughout 2023, the task force expanded engagement within the GIF to align efforts related to non-electric and hybrid applications of nuclear energy with other international or regional initiatives working to explore this area. The task force represents international expertise from GIF member countries, as well as international initiatives such as: the International Energy Agency's (IEA) Hydrogen from Nuclear Energy Task Group under the aegis of the Hydrogen Technology Collaboration Program; the IAEA's non-electric activities through the Nuclear Power Technology Development section; and the Nuclear Energy Agency's Small Modular Reactor Industrial Applications advisory groups.

The NEANH TF is also working to develop a network to connect the GIF to the high-temperature commu-



Note: The data are compiled from publicly available resources including IAEA Power Reactor Information System publications, EPRI, 2022, Decarbonizing Industry with Nuclear Energy: A Review of Nuclear Industrial Applications, and other news articles or reports.

Figure NEANH-1: Use of nuclear heat for non-electric applications, geographical distribution

nity outside the nuclear field, including through regional engagement in GIF member countries and other events. In 2023, NEANH members participated in multiple engagement opportunities, including the Sustainable Nuclear Energy Technology Program from 15-17 May in Gothenburg, Sweden; the ARPA-E (Advanced Research Projects Agency – Energy) Nuclear Heat Workshop from 30 May to 1 June in Houston, United States; and the IAEA Technical Meeting on Advances in High Temperature Processes for Hydrogen Production with Nuclear Energy from 5-8 September in Vienna, Austria.

In 2023, the task force worked to finalize an initial GIF NEANH database, which will become publicly available in 2024. Version 1.0 of the database comprises an initial inventory of NEANH activities populated by task force members. The NEANH database is an evolving resource that will be regularly maintained. New versions will be released periodically on the NEANH website; the versions may differ in format, content and style. The task force envisions adding website functionality that will allow both GIF and non-GIF members to submit additional studies, demonstrations and deployment cases of non-electric applications of nuclear energy for review prior to inclusion within the public database.

The database includes an inventory of demonstrations or commercial systems where nuclear heat has been applied to non-electric applications. As shown in Figure NEANH-1, more than 70 reactors have been used for non-electric applications, and the database is also tracking a range of planned demonstrations or commercial systems that will use nuclear energy for purposes such as the production of hydrogen, to support chemical production and for desalination.

The NEANH database also includes a range of reference information, with more than 60 research studies and publications identified related to non-electric applications of nuclear heat or analysis of heat markets, or the use of nuclear energy for non-electric applications identified for further study. The database also tracks 25 relevant international or regional collaborative initiatives in this domain.

Finally, 13 modelling and analysis tools have been identified that can be used to inform opportunities for nuclear heat, the integration of nuclear with renewables, and other technologies. Some of these tools are available as open source resources.

In 2024, the NEANH TF will continue to interact and collaborate with industrial end users and will advance towards the following milestones:

1) **Expand collaboration**: Invite new members to the NEANH Task Force who participate in related groups external to the GIF. NEANH will also work to strengthen collaborations with other GIF working groups/task forces, SIAP and international efforts exploring non-electric applications of nuclear heat.
2) **Create a digital NEANH database**: This digital database will centralize information on NEANH activities pursued globally, promoting international knowledge sharing and collaboration. It will be available externally to the GIF and will allow submission of additional studies and demonstrations for review by task force members prior to inclusion in the database.
3) **Host the Non-Electric and Hybrid Energy Applications Workshop** at the 39th Korea Atomic Power Annual Conference on 26 April 2024 in Busan, Korea to strengthen engagement with industrial end users and engage new audiences.
4) **Conduct system analysis on NEANH**: The task force will inform the energy community of the opportunity for Gen IV systems to support NEANH applications by evaluating generic system scenarios using performance indicators.

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with contributions from NEANH TF members

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

Industry perspective on Gen IV demonstration and deployment

In 2023, the global community reaffirmed nuclear energy's role in combating climate change and advancing decarbonization. At the G7 summit, the importance of nuclear power was recognized, with particular emphasis on the need for research and development in small modular reactors (SMRs), which include many Gen IV reactors (The White House, 2023). Additionally, at COP28 a coalition of countries and organizations acknowledged the necessity to triple nuclear capacity by 2050 to achieve decarbonization goals, highlighting nuclear energy as a pivotal element in the global strategy to mitigate global warming (NEA, 2023).

To achieve the ambitious goal of tripling nuclear capacity by 2050, the advancement and integration of Gen IV nuclear reactors is crucial. These reactors represent a significant evolution in nuclear technology, offering enhanced safety features, greater fuel efficiency and the potential to minimize nuclear waste.

In 2023, the private sector made significant progress in developing Gen IV reactors.

In the field of lead-cooled fast reactors (LFRs), a consortium of organizations including SCK CEN (Belgium), ENEA (Italy), Ansaldo Nucleare (Italy), RATEN (Romania) and Westinghouse Electric Company (United States) agreed to accelerate the development and commercial deployment of lead-cooled SMRs. The consortium's collaboration aims to realize a small-sized reactor in Belgium as an initial step towards global commercialization (WNN, 2023a).

newcleo and Tosto Group have partnered to develop the newcleo small modular LFR, with plans to commission a 30 MWe reactor demonstrator in France by 2030, followed by a 200 MWe commercial unit (WNN, 2023b). Ansaldo Nucleare and Westinghouse have successfully completed a testing campaign at the Passive Heat Removal Facility in the United Kingdom, marking a crucial step in the development of LFR technology. This effort is part of the Advanced Modular Reactor program, indicating the maturation of LFR projects and their potential for future energy production.

Implementation continues of the Russian strategic project area "Breakthrough", designed to open the way to the basic "green" energy of the future based on fast reactors and the closed nuclear fuel cycle. In accordance with the schedule, a unique pilot and demonstration energy complex is being constructed on the territory of the Siberian Chemical Combine in Seversk. The complex will include three interconnected pilot facilities: an innovative fast neutron reactor with lead coolant BREST-OD-300 with a capacity of 300 MWe, a module for production of mixed uranium-plutonium nuclear fuel, and a module for the reprocessing of spent fuel. The design of an industrial power complex is being developed, consisting of two power units with high-capacity lead-cooled BR-1200 reactors, industrial robotic fabrication and mixed fuel reprocessing facilities (WNN, 2024).

Regarding high-temperature gas reactors, X-energy completed the combined phases one and two of the pre-licensing vendor design review (VDR) for the XE-100 reactor with the Canadian Nuclear Safety Commission (Government of Canada, 2024). The regulator concluded that X-energy correctly understands and interprets the regulatory requirements for the design of nuclear power plants in Canada and reported no fundamental impediments to licensing. This agreement underlines a significant step towards commercializing the XE-100 reactor in the Canadian market.

Kaleidos, developed by Radiant Industries, has been selected by the National Reactor Innovation Center, a program led by the Idaho National Laboratory (INL) under the US Department of Energy (DOE), for demonstration at one of INL's Demonstration and Operation of Microreactor Experiments facility sites in United States (Office of Nuclear Energy, 2023). Testing could begin as early as 2026.

China's National Energy Administration announced the commercial operation of the world's first modular high-temperature gas-cooled reactor nuclear power plant, HTR-PM, in Shidao Bay, Shandong Province, showcasing Gen IV nuclear energy technology and a move towards achieving China's "dual carbon" goals (WNN, 2023c).

GIF Senior Industry Advisory Panel (SIAP) activity in 2023

The GIF SIAP was established in 2005 to provide strategic guidance on the commercialization potential of Gen IV nuclear energy systems, leveraging industry insights and expertise.

In 2023, SIAP conducted elections for the Chair and Vice Chair due to term completions. At the meeting in April 2023, Nawal Prinja (United Kingdom, Jacobs) was elected as the new Chair, succeeding Eric Loewen (United States, GE Hitachi Nuclear Energy). Subsequently, through an online voting process, Philippe Monette (Euratom, Tractebel) was elected as the new Vice Chair.

In 2023, SIAP convened three meetings to consider specific topics identified by the Policy Group to address the following charges:

- Assessing effective industry engagement opportunities with the GIF, exploring ways for improvement and acceleration, and identifying which end users can benefit from such engagement.
- Determining the best mechanisms for the regular communication of GIF achievements to industry, and enhancing the visibility and awareness of GIF products to end users.

Based on the feedback and discussions at the SIAP meetings, the SIAP provided advice to the GIF PG on these issues:

Effective industry engagement opportunities with the GIF:

- The most effective industry engagement opportunity highlighted was the Industry Forum held in Canada, which was considered very successful. It is recommended to be repeated and possibly integrated into larger events such as the World Nuclear Association and the World Nuclear Exhibition to foster broader industry engagement.
- Engagement opportunities should not be limited to vendors but expanded to include all end users, such as consultants, new startups and government entities involved in the promotion of new technologies. This inclusive approach ensures that the benefits of engagement are widely distributed, and that innovative and emerging sectors within the industry are also involved.
- To improve and accelerate engagement it is suggested that the success of the Industry Forum in Canada be used as a model for future events. Repetition at major events can help gather a large and diverse group of industry stakeholders, thus enhancing engagement and collaboration opportunities. This will help establish good industry practice for the application of new technologies to evolve into needed industrial codes and standards for new technology deployment.

Mechanisms for regular communication of GIF achievements:

- Regular communication of GIF achievements to the industry should be conducted through SIAP membership, online platforms, educational webinars and large meetings such as the Industry Forum. These channels have been effective in reaching a broad audience, as evidenced by the attendance of over 100 participants in some webinars.
- Enhancing the visibility and awareness of GIF products requires making these products visible to end users and providing clear guidance on how to access them. Access to different GIF products varies depending on factors such as country partnerships, which highlights the need for streamlined access to materials databases, economic modelling tools and other resources.
- SIAP special sessions, as identified below, have also proved fruitful for more detailed sharing on specific GIF collaborations and products.
- The visibility of GIF products to end users is considered adequate, but there is a need to improve how these products are shared with the industry and the awareness of end users in terms of how to access them. This involves providing clear instructions on accessing various GIF products, some of which may be freely available while others require specific partnerships or qualifications.

Digital technologies for knowledge management:

- Over the years, the GIF has collected many research reports and documents that are in an unstructured format and stored in different places. The GIF should consider using cognitive search engines powered by artificial intelligence (there was a successful demonstration with MSR research reports in October 2023 using Goldfire) to extract and provide the correct information to the right people.

SIAP special sessions on selected topics

In 2023, SIAP initiated special sessions as an innovative platform to engage existing members and a broader spectrum of stakeholders, particularly from the industrial sector, in discussions about GIF activities. This move came following the success of a special session held in Canada in 2022 that underscored significant interest and potential for collaboration in the field of advanced nuclear technologies. The SIAP session was a strategic response to both the evolving landscape of nuclear energy development, which increasingly involves private sector participation, and the specific directives outlined in the SIAP charge, which focuses on assessing effective industry engagement opportunities and enhancing the visibility of GIF achievements.

In 2023, the SIAP conducted two special sessions in July and November.

The July session featured Megan Moore, Co-Chair of the Economic Modelling Working Group, and included discussions on the "Overview of the Economic Modelling Working Group", "Cost Estimating Guidelines", and the G4ECONS tool. The virtual event attracted over 100 participants, fostering discussions on how to collaborate with industry to refresh information and update projects effectively.

In November, leveraging the occasion of the World Nuclear Exposition in Paris, a special session on "Advanced Manufacturing and Materials Engineering" was co-hosted with the CEA. Isabella van Rooyen, Co-chair of the Advanced Manufacturing and Materials Engineering Working Group, presented the task force's achievements and the working group's activity plans. This session also initiated a survey to determine the direction of SIAP activities, capitalizing on the gathering of a wide industry audience.

Looking forward to 2024, SIAP plans to host multiple special sessions to continue fostering exchanges between the GIF and industry. SIAP's goal remains to steer Gen IV technology towards commercial success, ensuring a collaborative and progressive pathway for nuclear innovation.

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Appendix 1. Country reports 2023

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 recognize 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 life-cycle emissions. The Australian 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 materials and technology.

Australia continues to develop the AUKUS trilateral security partnership with the United Kingdom and the United States. AUKUS will enable Australia to leverage nuclear-powered submarine expertise from the United Kingdom and the United States, building on decades of experience in their respective programs. Announcements by the Australian Government in early 2023 indicated continuing support for the program, including the formation of a dedicated agency to manage Australia's efforts. The Australian Submarine Agency was formed on 1 July 2023 and will be responsible for the management and oversight of Australia's nuclear-powered submarine program. The Australian Nuclear Science and Technology Organization (ANSTO) continues to play a role in supporting the AUKUS program, including advice to the Australian Government.

On 26 September 2023 the Australian Government announced plans for a new nuclear medicine facility at ANSTO. The new facility, which represents a significant investment from the Federal Government to further safeguard the production of life-saving nuclear medicines in Australia, will replace the existing aging Nuclear Medicine Processing and Distribution Facility, which was constructed in 1959 as a research laboratory. It will also support ongoing maintenance of the existing facility until the new nuclear medicine facility is commissioned and operational, expected in the mid-2030s.

ANSTO continues to support Australia's progress in the development of the Australian Radioactive Waste Agency (ARWA). ANSTO provides a broad range of engineering design, safety, operations and security services to ARWA. This assists with the design development and regulatory basis for the future Australian low-level waste repository and intermediate level waste storage facility.

ANSTO also continues to support the development of an Australian National Radioactive Waste Management Facility (NRWMF). In July, the Australian Federal Court set aside the 2021 declaration of the preferred site for the NRWMF by the Australian government. The government has now elected not to proceed with any of the shortlisted sites for the NRWMF and the siting process will recommence. The need for the NRWMF and mission of ARWA remain unchanged. ANSTO continues to provide a broad range of engineering design, safety, operations and security services to ARWA to support the future NRWMF.

As part of its overall nuclear medicine production facilities, ANSTO is building a first-of-a-kind waste treatment facility. The plant will use ANSTO's patented Synroc technology to treat and immobilize intermediate level waste that arises from the extraction of molybdenum-99. The project has transitioned into the cold commissioning phase, with key process technology and process integration being verified and validated. This commissioning process will continue over the next 18 to 24 months.

ANSTO'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, that any disruption to the domestic supply of nuclear medicine to Australians is minimized.

A focus for Australia in its role as a member of the Generation IV International Forum continues to be the mutual benefits reaped from international cooperation in programs that underpin the next generation of nuclear technology. Australia is pleased to provide the majority of its contributions in support of high-temperature reactor projects, along with contributing research in support of molten salt reactor projects. For instance, ANSTO is undertaking fundamental studies into the development of Synroc-type wasteform options for fluoride-rich salt waste projected from molten salt reactors.

Canada

Canadian domestic updates

1. Federal and provincial policy and investments announced in support of nuclear energy

Small modular reactors (SMRs)

In support of Canada's SMR Action Plan, the government of Canada (Natural Resources Canada, NRCan) announced CAD 29.6 million (Canadian dollars) over four years "Enabling SMR Program" to fund R&D to address waste generated from SMRs and to develop Canadian supply chains for SMR manufacturing and SMR fuel supply. Additionally, NRCan partnered with the Natural Sciences and Engineering Research Council (NSERC) of Canada to fund research on SMRs at Canadian universities. Through this partnership, NRCan will be providing CAD 1 million per year over four years to co-fund research projects through NSERC's Alliance grants program.

The government of Canada approved up to CAD 74 million in federal funding for SMR development in Saskatchewan, led by SaskPower. Additionally, the Saskatchewan Research Council (SRC) and Westinghouse Electric Canada have signed a Memorandum of Understanding to advance very small modular reactors, also known as micro-reactors, in Saskatchewan. Westinghouse and SRC will jointly develop a project to locate an eVinci micro-reactor in Saskatchewan for the development and testing of industrial, research and energy use applications. SRC is Canada's second largest research and technology organization.

The Alberta provincial government announced CAD 7 million through Emissions Reductions Alberta to support a study by Cenovus Energy on how small modular reactors could be used for oilsands operations in the future.

The government of Canada will provide CAD 7 million to support predevelopment activities for ARC Clean Technology Canada's small modular reactor at Point Lepreau in New Brunswick.

Large-scale nuclear reactors

The government of Ontario and Bruce Power formally announced the potential to build up to 4 800 megawatts of new electrical generation capacity at the Bruce site. In addition, it was announced that Ontario Power Generation would begin the planning and licensing for three additional SMRs, for a total of four SMRs at the Darlington nuclear site. Once deployed, these four units would produce a total of 1 200 megawatts of electricity.

Financing and tax incentives

The government of Canada has announced updates to the Green Bond Framework, aligning with the announcement in August 2023 associated with the Clean Technology Investment Tax Credit. The Clean Technology Investment Tax Credit was expanded, and draft legislation was published by the Department of Finance to include small modular reactors in the definition of clean technology for these credits.

2. Radioactive waste and liability insurance

In response to the recommendation in Canada's Small Modular Reactor Action Plan to review liability regulations under the Nuclear Liability and Compensation Act, to ensure that nuclear liability limits for SMRs are aligned with the risks they pose, the government of Canada is engaging with key federal partners and stakeholders to understand the risks associated with SMRs.

Canadian international relations:

Canadian PM – US POTUS Leader Statement (March 2023)/Natural Resources Canada (NRCan) – US Department of Energy (DOE) Joint Statement (March 2023)

The March 2023 Joint Statement affirmed the intent to enhance nuclear energy, technology, and fuel security collaboration. A simultaneous NRCan-DOE Statement on nuclear energy and technology committed the two countries to working together and with allies to accelerate advanced nuclear energy generation and a reliable nuclear fuel supply. Canada announced its intention to join the Foundational Infrastructure for Responsible Use of Small Modular Reactor Technology (FIRST) program, providing funding and in-kind support.

The NRCan-DOE statement also includes language on working together to develop a secure and reliable nuclear fuel supply of low enriched uranium fuel for existing reactors and high assay low enriched uranium (HALEU) fuel for advanced reactors, and to explore enabling frameworks with like-minded allies and partners.

Sapporo 5 Statement (April 2023)

Canada, France, Japan, the United Kingdom, and the United States (the Sapporo 5) signed and released a joint statement on new nuclear fuel cooperation at the Nuclear Energy Forum at the G7 in Sapporo, with the objective of reducing reliance on Russia for nuclear fuel.

Romania Export Financing (September 2023)

Canadian Federal Minister of Energy and Natural Resources, Jonathan Wilkinson, and Romanian Minister of Energy, Sebastian Burduja, announced Canada's decision in making available up to CAD 3 billion in export financing to Romania's Nuclearelectrica for the CANDU reactors 3 and 4 Project.

Sapporo 5 II Statement (December 2023)

On the margins of COP 28, the "Sapporo 5", announced their collective intent to support increased deployment of zero-carbon, peaceful nuclear energy by expanding nuclear fuel production capacity across trusted, high-quality suppliers free from manipulation and influence. The Sapporo 5 will work to mobilize at least USD 4.2 billion in government-led and private investment in the five nations' collective enrichment and conversion capacity over the next three years, with a view to further increase private sector finance and invite all like-minded nations to join in securing the global uranium supply chain.

28th Conference of Parties (COP28) Statement (December 2023)

At COP 28, Canada, along with 24 other nations, endorsed the Multilateral Declaration on Tripling Nuclear Energy. The initiative involves a commitment to work together to advance a global goal of tripling nuclear energy capacity by 2050, taking domestic actions to ensure responsible operation of nuclear power plants, mobilizing investments in nuclear power, and encouraging the inclusion of nuclear energy in international financial institutions' energy lending policies.

China

Nuclear energy development and application

As of 30 September 2023, there are 55 nuclear power units in operation with the installed capacity of 57 GW, and 24 units under construction with capacity of 27.8 GW in the Chinese Mainland. As of August 2023, the Chinese Government has approved the construction of three nuclear power projects in Shidaowan, Ningde and Xudapu, totaling six units, and is expected to maintain a construction pace of six to eight units per year in the future.

The Interregional Workshop on Technology Development and Application of Small Modular Reactor (SMR) was convened on 4 September in Hainan, China. This event was co-hosted by the International Atomic Energy Agency and the China Atomic Energy Authority, with China National Nuclear Power as the organizer. Over 200 government officials and experts from more than 50 countries and regions, including Brazil, China and France, gathered to engage in in-depth discussions on SMR technology, its development, user requirements, and safety oversight.

On 10 August, the core module of the Linglong One reactor, a multi-purpose modular small reactor in Hainan, was lifted and moved towards the nuclear island. The pressure vessel, evaporator and other key equipment were installed in one step, marking the peak phase of Linglong One's installation. This is the first milestone of the installation of Linglong One nuclear island. It marks the debut of modular manufacturing and installation of nuclear reactor modules, representing a historic step of global nuclear miniaturization.

The groundbreaking ceremony for Unit 5 (C-5) of the Chashma Nuclear Power Plant in Pakistan was held on 14 July. This is the third one-million-kilowatt-class unit of Hualong One, China's self-developed third-generation nuclear power technology, exported from China to Pakistan. It also marks China's seventh nuclear power unit exported to Pakistan, another milestone in nuclear energy cooperation between the two countries.

Progress of GIF projects

- **SFR**: The instrumentation and controls system of the China Experimental Fast Reactor were upgraded. The reactor was restarted on 14 July and connected to the grid on 3 August. In 2023 it operated at 35% power for 15 days and a scheduled shut down took place on 25 August. Work is ongoing to change the primary pump motor to improve the reliability in residual heat removal state. It will reach full power operation by the end of 2023. The China Demonstration Fast Reactor, CFR600, Unit 1 is being commissioned, and Unit 2 is under construction.
- **VHTR**: The HTR-PM finished 100 hours of steady operation on 30 September 2023. The basic design for new HTR-PM600 with six nuclear steam supply system modules and one steam turbine was finished, with the capability to provide high-temperature steam in co-generation mode. The R&D in GIF Projects on fuels, materials and computational modelling are going as planned, with plans to join the VHTR hydrogen production project.
- **SCWR**: Two of China's national R&D project contributions under the framework of GIF Projects have been completed in 2023. These projects solved the key materials, flow and heat transfer simulation, and safety analysis procedures for the development of SCWR, and pointed out the R&D direction. For the CSR150, a new fuel assembly with mixed moderators is devised, with the advantages of not only simple design, but also high capability of economy and safety. In addition, the coolant flow mode has changed to meet the design criteria.
- **LFR**: The State Power Investment Corporation, China General Nuclear Power Group and Xi'an Jiaotong University have recently started paying more attention to LFR development. Series conceptual designs such as the BLESS-D and CLFR-300 have been proposed. Some transient safety analysis tools are developed. In order to explore advanced UN powder preparation for fuel fabrication, the simulation experiment and theoretical calculation are carried out. To improve corrosion resistance of 15-15Ti steel for the cladding tube, grain boundary engineering technology is being used.

Euratom

The Russian military aggression against Ukraine continues after more than one year and a half. This act of war is a violation of international law, undermining European and global security and stability. Russia must bear full responsibility in front of the international community for its unlawful and reckless actions. The European Union reaffirms its support to Ukraine and its people.

Ensuring autonomy and independence of energy-related supplies remains at the forefront among the main challenges and priorities in the present geopolitical context. Several Member States are expressing or re-affirming interest in nuclear energy technologies. The decisions on the national energy mix is a competence and responsibility of each Member State.

Relevant policy developments in 2023

Small modular reactors

On 4 April 223, Commissioner on Innovation, Research, Culture, Education and Youth, Mrs. Gabriel, signed with Nucleareurope, the Sustainable Nuclear Energy Technology Platform, the European Nuclear Society and the European Nuclear Education Network an ambitious Declaration: EU Small Modular Reactors (SMRs) 2030: Research & Innovation, Education & Training.[1] Following this Declaration, the EU will continue to lead research, innovation, education and training for the safety of European SMRs. SMRs are defined to include not only LWR but also innovative technologies such as those of Gen IV systems.

The European pre-partnership on SMRs[2] finished its work in 2023 with the publication of a report for each of the five work streams: market analysis, licensing, financing, supply chain, and research and innovation. Following a stakeholder forum held by the European Commission on 26 October 2023 it was proposed to set up an SMR "Industrial Alliance" at the EU level to support the development and deployment of SMRs in EU Member States. The Alliance will become a forum for drawing up a strategic research agenda for SMRs to be implemented in a coordinated way by industrial actors and interested EU Member States.

On 12 December 2023, the European Parliament adopted an "own initiative" Report on Small Modular Rectors in support of these technologies and recognizing its potential benefits in terms of helping to decarbonize Europe's energy mix, ensure security of supply and support hard-to-abate sectors such as industry.

Several Member States and the private sector have been assigning significant funds for research, development and innovation on SMRs notably in Bulgaria, Estonia, France (for the design of NUWARD), Poland and Romania.

Net-Zero Industry Act

In March 2023 the EU proposed electricity market reforms under the Net-Zero Industry Act, which is intended to bolster Europe's manufacturing output in technologies needed for decarbonization. Small Modular Reactors (SMRs) were included in the initial proposal by the commission. The European Parliament and the European Council have both adopted their own positions, where nuclear power technologies are included. The objective is to finalize this legislation before the conclusion of the current legislative cycle.

Nuclear Alliance

Eleven European countries committed on 28 February 2023 to "cooperate more closely" across the entire nuclear supply chain, and promote "common industrial projects" in new generation capacity as well as new technologies like small reactors. The Alliance meets regularly and issues declarations on nuclear topics of interest.

Nuclear hydrogen

In March 2023, a provisional agreement was reached to reinforce the EU's Renewable Energy Directive. This new text now recognizes "the specific role of nuclear power, which is neither green nor fossil". And EU came to an agreement on labelling nuclear hydrogen as "low carbon".

New research reactor for Petten site

In September 2023, the Dutch government confirmed it will fully finance the cost of a new nuclear research reactor at Petten in the Netherlands, the Pallas reactor, which is being built to replace the ageing high-flux reactor (HFR).

European Nuclear Energy Forum

European nuclear sector stakeholders met with policy makers in the annual European Nuclear Energy Forum in Bratislava on 6 and 7 November. The event confirmed an increasing interest in nuclear technologies in some EU countries, and their potential role in meeting Europe's decarbonization targets and security of energy supply objectives.

Euratom contributions to GIF

JRC and Euratom member state representatives proactively contribute to the six systems in steering committees, Projects, working groups, and task forces. The contributions to GIF consist of:

a) Indirect Actions, The Call HORIZON-EURATOM-2023-NRT-01, with 11 topics: opened 4 April and closed 8 November, 2023. It has 11 topics with one dedicated to "Safety of light water small modular reactors (LW-SMRs)" and other budget line for "Safety of advanced and innovative nuclear designs".

b) Direct Actions, which correspond to parts of the JRC research program; JRC mainly contributes along three dimensions: 1) coordination and management (including interfacing at various levels); 2) system design integration; and 3) supply of data from experiments performed in JRC research facilities on inactive structural materials and active/irradiated fuels and compounds.

c) EU Member States contributions from their national research programs.

1. https://research-and-innovation.ec.europa.eu/system/files/2023-04/ec_rtd_eu-smr-declaration-2030.pdf.
2. https://energy.ec.europa.eu/topics/research-and-technology/small-modular-reactors_en.

France

French national context

1. Nuclear energy policy

Following the multi-annual energy plan performed in the previous years, affirming that nuclear energy is a long-term option in the frame of a more balanced electricity mix (nuclear production should reach 50% of the electricity production mix by 2035). In November 2021, President Emmanuel Macron announced the launch of new nuclear power plants – (European Power Reactor type PWRs) in order to progressively replace currently operating nuclear power plants. During its speech in February 2022, the reaffirmed this policy which includes:

- The future construction of six EPR2 plants (to be build 2*2 starting in 2028).
- The launch of engineering studies for eight more EPR2.
- A call for projects for SMRs and for innovative reactors allowing fuel cycle closure for nuclear waste decrease (through the France 2030 investment plan).
- On the other hand, the reprocessing strategy and recycling of nuclear fuel was confirmed and will be maintained until the 2040s horizon.

Nuclear Policy Council

The President of the French Republic convened two Nuclear Policy Councils in 2023 (a third one is planned beginning of 2024), whose role is to define and implement the major orientations of the French nuclear policy. These meetings confirmed that nuclear energy policy is being steered at the highest level of government, in line with the President's Belfort speech.

EPR2 reactor program in France

The Flamanville EPR is now on the launch pad. Now that all the weld remediations have been completed, the future reactor has embarked on the key phase of overall requalification testing in October 2023. For a period of 10 weeks, the reactor will be tested in details before receiving its fuel in Spring 2024.

In addition, Electricité de France (EDF) is engaged in the authorization procedures required for the launch of the construction of the first pair of EPR2 reactors at Penly, Normandy, as well as the administrative procedures for its completion and its link-up to the electricity grid. EDF's target is to start preparatory work by mid-2024. On 28 June 2023, the EDF Board of Directors decided to proceed with the planned construction of the first pair of EPR2 reactors at Penly.

2. Innovation program

Innovative Nuclear Program Agency

Besides decisions on PWRs, the Nuclear Policy Council supports the acceleration of two nuclear innovation items. the NUWARD small reactor project detailed design and innovative and advanced reactors with the objective to get, at least, one FOAK by the 2030s. The central role of the CEA to guide these developments has been reinforced through the creation in August 2023 of the Innovation Nuclear Program Agency to support this new start-up ecosystem.

In November 2023, the government has announced six new winners of the "innovative reactors" call for projects under the France 2030 investment plan: Jimmy Energy (HTR), Renaissance fusion (Fusion), Calogéna (SMR), Héxana (SFR), Otrera Nuclear Energy (SFR) and Blue Capsule (HTR) will receive financial support worth EUR 77.2 million, coupled with EUR 14 million in technical support from the CEA.

3. R&D programs

During a speech in December 2023, President Macron announced his intention to transform major national research organizations into genuine program agencies. Each agency will have to be increasingly strategic in its own field and participate in the creation of priority research themes, interact with international counterparts and oversee the development of research infrastructures. They will thus have a real mandate and specializations, and will be encouraged to take risks and prepare the major research programs of tomorrow. Therefore, the CEA is foreseen to become the agency for low-carbon energies, components of digital systems and infrastructures.

Status of the French fast reactor program

The CEA is implementing its new fast reactors -related activities focused on a strong R&D program dedicated to further progress on the fast-reactor technology and the associated fuel cycle. The priority is still given to the sodium-cooled fast-reactor technology, considered as the most mature to comply with the mid-term target (second half the century). The program also includes other fast reactor concepts assessment.

Other concepts of fast reactors, in particular fast spectrum molten salt reactors are being studied in order to identify the key feasibility issues, as well as their specific features and potential performances. To that end, CEA, Orano, EDF, Framatome and the CNRS have launched a four year project supported by the France 2030 investment plan. Design studies, as well as material and corrosion investigations are to be carried out in parallel with an important R&D program dedicated to salt manufacturing and treatment processes. International collaboration is being discussed in order to accelerate industrial development of this technology.

Material test reactor: Jules Horowitz Reactor project

Since March 2020, a new project organization required by the French government has been implemented. It has several advantages compared to the previous arrangement, with a stronger involvement of the French nuclear industry (Framatome, EDF and Technicatome) and an integrated technical platform gathering all key subcontractors (the engineering companies and the CEA). This new organization is helping to secure the completion of the project and putting it on a more robust track. One of the main outcomes of the new organization is to reassess some detailed design studies (ventilation, circuit and electricity) to freeze the 3D mock-up before starting the implementation of the components.

The Nuclear Policy Council reaffirmed the wish to continue and complete the project in order to start the reactor as soon as possible.

Japan

Green Transformation

Prime Minister Kishida expressed that making up for the delay in energy policies is a pressing issue in implementing Green Transformation (24 Aug. 2022, Green Transformation Implementation Council). Items requiring political decisions to expand the introduction of renewable energy were presented, including

- drastically accelerating the development of power systems;
- speeding up the installation of fixed storage batteries;
- promoting offshore wind power and other electricity sources.

At the same time, items requiring future political decisions concerning nuclear power were also put forward:

- Combined efforts by the concerned parties to resume operations.
- Maximum utilization of existing nuclear power plants, including the extension of their operation period, while ensuring safety.
- Development and construction of next-generation advanced reactors with built-in new safety mechanisms.

On 10 February 2023 the cabinet approved a basic policy aimed at implementing a Green Transformation (GX). The Japanese government's Green Transformation (GX) Implementation Council (chaired by prime Minister KISHIDA Fumio) and councils at relevant ministries have been deliberating, since last summer, ways to achieve decarbonization, stable energy supplies and economic growth simultaneously.

The basic policy includes the following delineated points toward securing stable supplies of energy:

- Promoting thorough conservation of energy.
- Making renewable energies the main power source.
- Utilization of nuclear power: On the basic premise of gaining local understanding, materializing plans for building next-generation advanced reactors within the sites of existing nuclear power plants that have determined to be decommissioned.
- Development based on the revised strategic roadmap for fast reactor decided by the Inter-Ministerial Council for Nuclear Power on 23 December 2022, is ongoing.

Advanced reactors

2023 Summer: Selection of a demonstration reactor concept and a core company.

- Mitsubishi Heavy Industries was selected by the Japanese government in July 2023 to lead the conceptual design of a demonstration reactor, of a pool type, sodium-cooled fast reactor planned to enter operation in the 2040s.
- 2024-2028: Conceptual design and related R&D
- 2028: Decision to move to basic design and licensing application for the demonstration reactor.

On 26 July 2023, the experimental fast reactor Joyo was granted a reactor modification license under the new regulatory standards, and preparations for modification and reinforcement of facilities and buildings are ongoing. Joyo is scheduled to be restarted in the middle of FY 2026 and will be used for irradiation tests for fuels relating to the demonstration reactor, materials, etc., and demonstration of medical radioisotope production.

The High Temperature Engineering Test Reactor (HTTR) was restarted in 2021. A safety demonstration test from the full power condition with all helium gas cooling systems stopped will be conducted as NEA international cooperation program (the Loss of Forced Coolant, LOFC) in 2024. The HTGR demonstration reactor development project has also been initiated based on the plan, and Mitsubishi Heavy Industries has been selected as the developer. A demonstration project for hydrogen production by the existing HTTR is currently underway.

Restart of existing nuclear power plants

As of December 2023, 12 light water reactor units have been restarted, 4 have been certified, and 9 are under examination based on new nuclear regulations.

Korea

Nuclear power in Korea

As of July 2023, there were 25 nuclear power plants (22 PWRs and 3 CANDUs) in operation in South Korea. The combined nuclear capacity of these 25 power plants accounted for 17.2%, equivalent to 24 650 MWe of the total capacity. In 2022, these nuclear power plants generated 176 054 GWh of electricity, representing 29.6% of the total domestic production. Additionally, South Korea is currently constructing two PWRs, Shin-Kori units 5 & 6, and Shin-Hanul unit 2 has received permission for commercial operation, expected to begin commercial operation in the first half of 2024.

Nuclear energy policy and R&D in Korea

The Korean government continues to prioritize nuclear energy and has announced a new policy to strengthen its nuclear strategy in 2023. In May 2022, the new government completely abandoned the previous nuclear policy to phase out nuclear power and instead resumed the construction of the Shinhanul Units 3 and 4 nuclear power plants, which had been suspended under the previous regime.

The new government plans to expand R&D and industry based on "National Strategic Technologies". Next-generation nuclear systems have been selected as one of the National strategic technologies. SMRs, Gen IV nuclear systems, and waste management have been identified as key technologies for next-generation nuclear systems. Therefore, the government will promote the demonstration of advanced SMRs and the development of various applications of nuclear reactors. In the future, the development of Gen IV systems will be promoted as a public-private partnership project in which the government and the private sector cooperate. It is expected to form a new value chain in the field of nuclear industry through public-private partnership projects. In particular, the government will propose a technology development model in which the Korean government and private companies cooperate to accelerate the demonstration of MSR and HTGRs. The government is looking forward to the realization of nuclear technology with industry and its entry into the market.

Sodium-cooled Fast Reactor (SFR)

Korea's SFR program has made consistent progress in developing an SFR-based power generation reactor known as SALUS, along with technology demonstrations for transuranic transmutation. SALUS is designed to generate 100 MWe of electric power, focusing on being a long-fuel-cycle core power generation reactor suitable for the near-term SMR market. Ongoing R&D projects in the SFR domain encompass three core areas: 1) enhancing sodium experimental capabilities, such as the STELLA program 2) advancing modeling and simulations, including high-performance computing technologies; and 3) innovating in metal fuel technologies, covering manufacturing fuel irradiation tests, the general supply chain, and more.

Very High Temperature Reactor (VHTR)

VHTR R&D is ongoing through the "Very High Temperature System Key Technologies Development" project, which consists of three sub-projects: 1) developing performance evaluation technologies for design and analysis codes; 2) verifying the performance of high-temperature materials; and 3) exploring coupled technologies between very high temperature steam and a HTSE hydrogen production system. The first two sub-projects began in April 2020, while the remaining sub-project was initiated in April 2021, with all sub-projects slated for completion in 2024. There is increasing interest in the utilization of high-temperature heat from HTGRs across various industries, leading to the establishment of the Alliance for Nuclear Heat Utilization in August 2023. This alliance, comprising 13 organizations including the KAERI, Gyeongsangbuk-do local government, six plant construction companies and five end-user companies, plans to develop high-temperature gas-cooled reactors for domestic process heat supply through a public-private partnership project.

Molten-Salt Reactor (MSR)

Korea has initiated a new Gen IV project focusing on MSR-based SMRs. The primary goal of this project is to develop key technologies for MSRs. Recognizing the significance of collaboration with private companies to expedite MSR demonstration, the government has established a Molten Salt Reactor Development Agency. As private companies have shown their keen interest in MSR technology development, KAERI is actively collaborating with several leading Korean companies with the aim of realizing MSR technology in the near future.

Russia

In the first half of 2023, electricity generation by power units of all 11 nuclear power plants operating in the Russian Federation, including the floating nuclear thermal power plant in Chukotka, exceeded 100 billion kWh. This amount of electricity generated prevented greenhouse gas emissions into the atmosphere of almost 50 million tons of CO_2 equivalent.

Development of nuclear fuel for modernized floating power units intended for use in Baim Mining and Refining Plants continues. Each modernized floating power unit is expected to be equipped with two RITM-200S reactors. To provide electricity to the Baim project, the Rosatom State Corporation proposed the use of four units with two new RITM-200S reactors at each power unit, with an installed capacity of at least 106 MW per unit.

Work on foreign projects is proceeding according to the schedule, with 23 power units under construction.

Work on the construction of the Multipurpose Fast Research Reactor (MBIR) continues successfully. In January 2023, the MBIR reactor vessel was installed in the design position. This is one of the key events of the reactor unit assembly, which makes it possible to complete the construction of the reactor unit dome.

Within the framework of the federal project "New Materials and Technologies" work is underway to create a research liquid salt reactor (RLSR) with a circulating fuel based on lithium and beryllium metal fluoride salt melts. The design documentation for the RLSR is expected to be finalized by the end of 2026, and the reactor should be commissioned in 2030.

Implementation of the strategic project area "Breakthrough", designed to open the way to the basic "green" energy of the future based on fast reactors and the closed nuclear fuel cycle, continues. In accordance with the schedule, a unique pilot and demonstration energy complex, which has no analogues in the world, is being constructed. The complex will include three interconnected pilot facilities: 1) an innovative fast neutron reactor with lead coolant BREST-OD-300 with a capacity of 300 MWe; 2) a module for production of mixed uranium-plutonium nuclear fuel; and 3) a module for reprocessing of spent fuel. The design of an industrial power complex is being developed, consisting of two power units with high-capacity lead-cooled BR 1200 reactors, industrial robotic fabrication and mixed fuel reprocessing facilities.

Full-scale simulations have been completed and all neutron characteristics were experimentally confirmed for the BREST-OD-300 active zone with nitride fuel at the critical experiment facility BFS in Obninsk.

By the end of 2024, the production of mixed dense nitride uranium-plutonium fuel (MNUP-fuel) for new generation fast neutron reactors will be launched in Seversk. In 2023, work is underway to supply and commission equipment.

In March 2023, at the fourth unit of the Beloyarsk NPP with a BN-800 reactor, fuel assemblies with MOX fuel were loaded into the core during the latest refueling. At the time, the share of this type of fuel in the reactor was 98%.

South Africa

In August 2023, the National Energy Regulator issued a decision, a concurrence under Section 34 of the Electricity Regulation Act of 2006 for the Department of Mineral Resources and Energy, to proceed with the procurement of a nuclear build program to the extent of 2 500 MW at a pace and scale that the country can afford.

Eskom submitted a license application and a safety case report for long term operation to the National Nuclear Regulator in July 2022 to allow for the continuous operation of the Koeberg Nuclear Power Plant for an additional 20 years.

Unit 1 of the Koeberg Nuclear Power Station reached a significant milestone in July 2023 when the mechanical work of the replacement of the steam generators was completed. Activities for return to service of Unit 1 are currently ongoing and includes amongst others:

- pre-testing on the primary side and safety systems;
- reactor fuel reloading and synchronization to the grid.

A feasibility study is ongoing for the establishment of the Centralised Interim Storage Facility. Once completed and a shovel readiness status is obtained, procurement for the facility will commence.

The South African Nuclear Energy Corporation is forging ahead with the development of Multi-Purpose Research Reactor Project to complement the current Research Reactor, which is now almost 60 years old. The project has completed the Feasibility Study which is under review before procurement.

Switzerland

GIF activities

The long-term aim of the research on molten salt reactors (MSR) in Switzerland is the safety and fuel cycle sustainability, without particular focus on specific MSR concepts. This research relies on national and international projects and student education. In the safety part, the project dedicated to Dual Fluid Reactor concept was continued in 2023, analyzing its system behavior with TRACE/PARCS code. Also, the MSc thesis of M. Krstovic entitled “Applicability of CRAM on Bateman equations, point kinetics equations and their combination” was defended. This work addressed the drift of delayed neutrons precursors, where all nuclides were explicitly tracked. In the fuel cycle sustainability part, several modifications of the fuel cycle procedure EQL0D were finished and released as EQL0D v4. Also, a MSc thesis entitled “Identification and evaluation of transmutation criteria for selected reactors” was defended. In this thesis, 22 different reactor concepts were evaluated from the closed fuel cycle perspective, of which ten systems were based on molten salt fuel.

Sodium fast reactor (SFR) research studies have been carried out within the framework of the Euratom ESFR SIMPLE project, which is dedicated to European sodium fast reactor (ESFR) research and coordinated by the CEA. At the PSI, researchers and students have focused on conducting a comparative analysis of the neutronic aspects of two ESFR core options: one with oxide fuel and the other with metallic fuel. The initial design of a metallic fuel core for ESFR was completed in collaboration with the Argonne National Laboratory, yielding promising results for the neutronic core performance. These results suggest higher breeding potential and lower fuel temperatures when compared to the oxide fuel core, which could form the foundation for improved safety. The analysis of transients and fuel performance will be conducted in subsequent phases.

On the material research side, the focus continues on the development of new analytical tools for high temperature materials. The PhD Thesis dealing 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 is in the completion phase. Using results from synchrotron-based tomography, of composite, a complete model could be realized, containing the pores and fibers and a cross comparison between the conductivity experiments and the microstructure could be realized. A very strong anisotropy of the thermal conductivity has been identified and is being reported in a PhD thesis. The two PhD theses reported in the 2022 GIF Annual Report (irradiation induced creep and additive manufacturing) are ongoing as planned.

Politics and regulation

There is no relevant change in the Swiss Government Strategy to reach CO_2 NetZero in 2050.

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

Operation of the Swiss nuclear power plants and waste management

The dismantling of the Mühleberg boiling water reactor is ongoing as planned. All other reactors are in operation and operating at nominal power.

Nagra, the Swiss National Cooperative for the Disposal of Radioactive Waste, has started to prepare the general license applications for the site and the encapsulation plant. Nagra plans to submit it to the Federal Council in 2024.

Nuclear power related research in Switzerland

The key activities of the Nuclear Energy and Safety division at the Paul Scherrer Institute is the education of the next generation of nuclear experts, the scientific support for the safe operation of LWRs, the delivery of the scientific basis for the assessment of the deep geological repositories safety, and the technology monitoring including research work on Gen IV concepts.

The Swiss Master program in Nuclear Engineering has been selected as a kind of "role model" by the IAEA Knowledge Management Section. The IAEA conducted a Knowledge Management Assist Visits in PSI and EPFL (Institution level 3 = institutions with very well-established programs) in October. These visits are used to establish best practices in nuclear education and training and will flow into the IAEA's recommendations on how to successfully establish nuclear education in developing countries, a clear recognition of the quality and actuality of the Swiss Master's program.

In spite of the non-association of Switzerland, the participation of Swiss scientist to EURATOM and HORIZON-Europe research initiative are ongoing as planned.

United Kingdom

Nuclear energy update in the UK

The British energy security strategy, published in April 2022, will see a significant acceleration of nuclear energy, with an ambition of up to 24 GW by 2050. In 2022, low-carbon sources generated 183 terawatt hours (TWh) of the UK's total 326 TWh electricity generated, with nuclear accounting for 48 TWh.

Great British Nuclear

In April 2023 Great British Nuclear (GBN) launched the first phase of the technology selection process to select the best small modular reactor (SMR) technologies in the form of a market engagement exercise. In July 2023 it launched the second phase of the process, which invited SMR technology vendors to register their interest by responding to a Selection Questionnaire, which closed in August 2023. Six designs have been selected to progress in a government competition supporting the development of this innovative technology for greater energy security. The designs chosen are considered by the government and Great British Nuclear as the most able to deliver operational SMRs by the mid-2030s.

The government will provide co-funding to be deployed by GBN to support the development of these selected technologies, and will work with successful bidders on ensuring the right financing and site arrangements are in place, in line with its commitment to take two final investment decisions next parliament.

Advanced modular reactors (AMR)

The AMR Research Development and Demonstration program aims to demonstrate High Temperature Gas Reactor (HTGR) 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 UK, should be shaped by end-user requirements and should incentivize private investment in HTGRs by removing technical risk. The following two organizations were awarded grants as part of Phase B of the AMR RD&D Project:

- Ultra Safe Nuclear Corporation (USNC); and
- National Nuclear Laboratory (NNL).

After AMR Phase A, a decision was made to split advanced nuclear fuel development into a separate workstream called "Coated Particle Fuel (CPF) Demonstration – Step 1". The objectives for Step 1 are to overcome the RD&D and innovation challenges associated with CPF production to maintain the option for a sovereign CPF process and supply for HTGRs. A match-funded grant has been awarded to NNL as part of the CPF program.

Nuclear Fuel Fund

In July 2023, GBP 22.3 million was offered to eight projects that will support the UK's nuclear fuel supply chain to develop the capabilities needed to meet current and future fuel demand in the UK and globally.

Future Nuclear Enabling Fund

The UK Government launched the Future Nuclear Enabling Fund of up to GBP 120 million to provide targeted support for new nuclear development. Three applications for potential grants have been identified. The combined total of the shortlisted potential grants is GBP 77.1 million. This funding will accelerate advanced nuclear business development in the UK and support advanced nuclear designs to enter UK regulation.

AMR Knowledge Capture

Government is providing funding for the AMR Knowledge Capture Project, as a complementary project to the AMR RD&D program. The project seeks to facilitate knowledge capture and sharing to reduce the time, risk and cost of AMR RD&D program delivery.

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.

First new reactor build connected to the US Grid in three decades

Vogtle Unit 3 entered commercial operation in Waynesboro, Georgia on 31 July 2023, becoming the nation's first new reactor to connect to the grid since 2016. Unit 3 is the most advanced light water reactor system in the United States, and it leverages Westinghouse's AP1000 technology that can shut down without operator action or external power for 72 hours. Georgia Power plans to have Vogtle Unit 4 operating in 2024.

Advanced Reactor Licensing

The US NRC continues its activities to develop technology-inclusive, performance-based, risk-informed licensing practices for advanced reactors. Multiple proposed rules are currently pending.

NuScale Power

The US Nuclear Regulatory Commission (NRC) issued its final rule in February to certify NuScale Power's 50-megawatt power module. The company's advanced light-water system is the first small modular reactor certified by the NRC and just the seventh reactor design cleared for use in the United States. It will help pave a path forward for other domestic SMRs currently under development to deploy their technologies.

Kairos Power

In December 2023, the NRC approved construction of Kairos Power's Hermes reactor, which will be built in Oak Ridge, Tennessee as early as 2026. Hermes is the first Gen IV reactor to receive an approved construction permit from the NRC. Hermes will help inform the development of the company's commercial fluoride salt-cooled high-temperature reactor.

Support for clean hydrogen generation

DOE has been working to further integration of hydrogen production processes with nuclear power plants through real world demonstration projects at nuclear power plants. The first of these projects began producing hydrogen in February 2023 and two others are expected to produce hydrogen in 2024. Additionally, the DOE announced USD 7 billion to stand up seven regional clean hydrogen hubs across the country that will be funded through the Bipartisan Infrastructure Law. Three hubs will use nuclear energy as part of their projects to generate clean hydrogen for the regions.

Fueling future reactors

Centrus Energy Corporation produced the US's first 20 kilograms of high-assay low-enriched uranium (HALEU), a crucial material required by many advanced reactor designs. The production was the first of its kind in the United States in more than 70 years and completed a key milestone in the DOE's HALEU Demonstration project in Piketon, Ohio. The HALEU will be used to help fuel the initial cores of the DOE's two demonstration reactors awarded under the Advanced Reactor Demonstration Program and it will also support fuel qualification and other testing of new reactor designs. Centrus is expected to ramp up its production rate of HALEU material to 900 kilograms per year starting in 2024.

Advanced Reactor Demonstration Program (ARDP)

Advanced Reactor Demonstration Pathway

- Natrium by TerraPower, LLC: Progressed design maturity of all key nuclear reactor facilities, as well as the design of the Fuel Fabrication Facility and the design of the Sodium Test and Fill Facility. Completed site characterization at Kemmerer, WY, which is the planned location for Natrium. Completed the preparation of its draft Construction Permit Application (CPA) and plans to formally submit the CPA to the NRC in March 2024. Initiated long lead procurements for the major designed nuclear island equipment and developed various programs to support Natrium's readiness. Continued support for securing a HALEU supply and fuel fabrication, and for National Environmental Policy Act activities.
- Xe-100 by X Energy, LLC: Transitioned all reactor systems into final design, completed the conceptual design of the TRISO-X fuel fabrication facility, partnered with Kinectrics to design, construct, and operate one of the first commercial-scale Helium Test Facilities, and continued regular prelicensing engagements with NRC staff. Received safety evaluations for three topical reports. Initiated long lead procurements for the major designed Nuclear Island equipment and developed various programs to support Xe-100's readiness. Continued support for securing a HALEU supply, fuel fabrication, and licensing activities. Completed the purchase and build-out of a new training and simulator facility in Maryland. Dow and X-energy announced a collaboration, called Project Long Mott, to deploy X-energy's Xe-100 technology at the Dow Seadrift Operations site in Seadrift, TX, where site characterization activities were initiated.

Risk Reduction for Future Demonstration Pathway

- Hermes Reduced-Scale Test Reactor by Kairos Power, LLC: NRC approved Construction Permit for Hermes. Kairos Power initiated salt operations of a non-nuclear mockup of Hermes.
- eVinci™ Microreactor by Westinghouse Electric Company, LLC: Scaled-up and enhanced heat pipe manufacturing operations and finalized design for irradiation test of moderator material.
- BWXT Advanced Nuclear Reactor by BWXT Advanced Technologies, LLC: Initiated fabrication of TRISO fuel specimens to support irradiation testing.
- Holtec SMR-160 Reactor by Holtec Government Services, LLC: Issued plans for nuclear steam supply system equipment manufacturing.
- Molten Chloride Reactor Experiment (MCRE) by Southern Company Services Inc.: Completed design of the fuel salt synthesis line. Completed National Environmental Policy Act process resulting in a Finding of No Significant Impact.

National Reactor Innovation Center (NRIC)

In 2023, the NRIC initiated construction of the Demonstration and Operation of Microreactor Experiments (DOME) test bed. DOME will be capable of siting experiments to support microreactor technologies. Additionally, the DOE, through the NRIC, awarded USD 3.9 million to three advanced reactor developers to design experiments to potentially test microreactor designs in the DOME test bed. Radiant, Ultra Safe Nuclear Corporation, and Westinghouse will further their microreactor designs through a front-end engineering and experiment design (FEEED) process. The FEEED process supports developers in planning for the design, fabrication, construction, and testing of fueled reactor experiments. Testing in DOME could start as early as 2026.

Advanced Reactor Concepts-20 (ARC-20)

- 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 report documenting analysis of reactor passive safety and completed fabrication of fuel rodlets for irradiation in the Advanced Test Reactor.
- Horizontal Compact High Temperature Gas Reactor by Massachusetts Institute of Technology: Completed preliminary manufacturing and licensing assessments and completed conceptual core design. Demonstrated adequate performance by the passive decay heat removal system.

Microreactor Applications, Research, Validation, and Evaluation (MARVEL)

The DOE 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. In 2023, the MARVEL team completed 90% final design for the reactor and successfully built a full-scale prototype to support the project.

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Appendix 2. List of abbreviations and acronyms

3S	Safety, security and safeguards
ADS	Accelerator-driven system
AFA	Alumina-forming austenitic
ALFRED	Advanced Lead Fast Reactor European Demonstration Project
ALLEGRO	European Gas Fast Reactor Demonstrator Project
AMME-WG	Advanced Manufacturing and Materials Engineering Working Group
AMR	Advanced modular reactor
ANSTO	Australian Nuclear Science and Technology Organization
ANTSER	Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review
ARAMIS	Advanced Reactor for Actinides Management in Salt (France)
ARWA	Australian Radioactive Waste Agency
BREST-OD-300	Russian acronym for 300 MW passive safe pilot demonstration fast neutron reactor
CEA	Alternative Energies and Atomic Energy Commission (France)
CFD	Computational fluid dynamics
CFR	Chinese sodium fast reactor
CIAE	China Institute of Atomic Energy
CiFR	Chinese integrated fast reactor nuclear energy system
CLEAR	China Lead-based Reactor
CNL	Canadian Nuclear Laboratories
CNRI	Canadian Nuclear Research Initiative
CNRS	National Center for Scientific Research (France)
COP28	28th Meeting of the Conference of Parties, held in December 2023 as part of the United Nations Climate Change Conference
CPA	Construction permit application
CPF	Coated particle fuel
Cr	Chromium
CVR	Centrum výzkumu Řež (Czechia)
DHR	Decay heat removal
DOE	Department of Energy (United States)
DNS	Direct numerical simulation
dpa	Displacements per atom
ECC-SMART	Joint European Canadian Chinese – Small Modular Reactor Technology project
EDF	Électricité de France
ELFR	European Lead Fast Reactor
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
ESNII	European Sustainable Nuclear Industry Initiative
ETWG	Education and Training Working Group
EU	European Union
GBN	Great British Nuclear

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 ton of heavy metal
GWt	Gigawatt thermal
HALEU	High-assay low-enriched uranium
HTGR	High-temperature gas-cooled reactor
HTR	High-temperature reactor
HTR-PM	High-Temperature Reactor – Pebble-bed Module
HTSE	High-temperature steam electrolysis
HTTR	High Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
INET	Institute of Nuclear and New Energy Technology (China)
INL	Idaho National Laboratory (United States)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
ISAC	Innovative System for Actinide Conversion (France)
ISAM	Integrated safety assessment methodology (GIF RSWG)
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre (Euratom)
JSFR	Japanese sodium-cooled fast reactor
KAERI	Korea Atomic Energy Research Institute
KMKP	Knowledge management and knowledge preservation
LBM	Lattice Boltzmann Method
LFR	Lead-cooled fast reactor
LOCA	Loss-of-coolant accident
LTE	Low temperature electrolysis
LWR	Light water reactor
MOX	Mixed oxide (fuel)
MCRE	Molten Chloride Reactor Experiment (United States)
MSR	Molten salt reactor
MTA EK	Hungarian Academy of Sciences Center for Energy Research
MW	Megawatt
MWe	Megawatt electrical
MWt	Megawatt thermal
MPa	Megapascal
MSFR	Molten salt fast reactor (Euratom)
NaF	Sodium fluoride
NCBJ	National Centre for Nuclear Research (Poland)
NEA	Nuclear Energy Agency
NEANH	Non-electric applications of nuclear heat
NNL	National Nuclear Laboratory (United Kingdom)
NPIC	Nuclear Power Institute of China
NRC	Nuclear Regulatory Commission (United States)
NRCan	Natural Resources Canada
ODS	Oxide dispersion-strengthened
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory (ORNL)
PGSFR	Prototype Generation IV Sodium-cooled Fast Reactor
PMB	Project management board
PR	Proliferation resistance

PRPPWG	Proliferation Resistance and Physical Protection Working Group
PP	Physical protection
PSI	Paul Scherrer Institute (Switzerland)
pSSC	Provisional System Steering Committee
PWR	Pressurized water reactor
RANS	Reynolds-averaged Navier-Stokes (model)
R&D	Research and development
RSWG	Risk and Safety Working Group
SAIGA	Severe Accident In-pile experiments for Generation IV reactors (France)
SAMOSAFER	Safety Assessment for Fluid-Fuel Energy Reactors
SCWR	Supercritical water-cooled reactor
SCW-SMR	Supercritical water-cooled small modular reactor
SCK CEN	Belgian Nuclear Research Centre
SDC	Safety design criteria
SDG	Safety design guidelines
SFR	Sodium-cooled fast reactor
Si	Silicon
SIAP	Senior Industry Advisory Panel
SiC	Silicon carbide
SMR	Small modular reactor
SSC	System Steering Committee
SST	Shear stress transport
SSTAR	Secure Transportable Autonomous Reactor (United States)
STELLA-2	Large-scale Sodium Integral Effect Test Facility (Korea)
TRISO	Tri-structural isotropic (nuclear fuel)
TRL	Technology readiness level
TRU	Transuranics, actinide elements heavier than uranium
UCO	Uranium oxycarbide (fuel)
URANUS	Ubiquitous Robot Accident Forgiving Nonproliferating Ultra-lasting Sustainer (LFR design, Korea)
USNC	Ultra Safe Nuclear Corporation
VHTR	Very high-temperature reactor
VUJE	Nuclear Power Plant Research Institute (Slovak Republic)
WGNT	Working Group on New Technology (NEA)
ZrC	Zirconium carbide

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Appendix 3. Selection of GIF publications (2023)

Cipiti, B.B., L. Cheng, B. Boyer and B. van der Ende, "Generation IV Proliferation Resistance and Physical Protection: Transitioning from R&D to Deployment", *Proceedings from the INMM/ESARDA Meeting*, Vienna, Austria, May 2023.

Daniels, B. and P. Vácha (2023), "D4.2 GFR Needs for Nuclear Standardization and Codes", *Deliverables of the SafeG Project, Euratom Research and Training Programme on Nuclear Energy within the Horizon 2020 Programme for Research and Innovation*, www.safeg.eu/fileadmin/user_upload/images/SafeG_D4.2_GFR_Needs_for_Nuclear_Standardization_and_Codes_1.1.pdf.

GIF PRPPWG (2022), *Proliferation Resistance and Physical Protection Crosscutting Topics, GIF/PRPPWG/2022/004*, Generation IV International Forum, Paris, www.gen-4.org/gif/jcms/c_209989/prpp-crosscutting-topics-report-final.

GIF PRPPWG (2023), *Bibliography compiled by the Proliferation Resistance and Physical Protection Working Group (PRPPWG)*, Revision 10, September 2023, Generation IV International Forum, Paris, www.gen-4.org/gif/jcms/c_216710/2023-gif-prppwg-bibliography-final.

GIF PRPPWG and MSR SSC (2023), *GIF Molten Salt Reactor Proliferation Resistance and Physical Protection White Paper*, GIF/PRPPWG/2023/001, Generation IV International Forum, Paris, www.gen-4.org/gif/jcms/c_218175/gif-msr-prpp-white-paper-final.

GIF RSWG and VHTR SSC (2023), *Safety Design Criteria for Generation IV Very High Temperature Reactor System, GIF/VHTR-SDC/2023/001*, Generation IV International Forum, Paris, www.gen-4.org/gif/jcms/c_212748/vhtr-sdc-report-final.

GIF RSWG (2023), "A Risk-Informed Framework for Safety Design of Generation IV Systems", GIF/RSWG/2023/001, Generation IV International Forum, Paris, www.gen-4.org/gif/jcms/c_212805/riapproachpositionpaper-final.

Mikityuk, K, M. Ferreira, B. Hatala, J.L. Kloosterman and M. Šípová (2023), "Review of Euratom projects on design, safety assessment, R&D and licensing for ESNII/Gen-IV reactor systems", *EPJ Sciences & Technologies*, Vol. 9/18, https://doi.org/10.1051/epjn/2022048.

Paviet, P. (2023), "Informing and Engaging the Future Workforce on Generation IV International Forum Reactor Systems" Paper #40428, CONTE 2023 Conference, Amelia Island, 9 Feb 2023.

Smith, C.F. and L. Cinotti (2023), "Lead-cooled Fast Reactors (LFRs)", in Pioro, I (ed) *Handbook of Generation IV Nuclear Reactors*, Second edition, Woodhead Publishing.

Van der Ende, B. (2023), *Joint RSWG/PRPPWG/VHTR-SSC effort for establishing safeguards, safety, and security interfaces for VHTR system*, Contribution to the Joint IAEA-GIF Workshop on the Safety of Non-Water-Cooled Reactors, May 30 to 2 June + IAEA Interregional Workshop on Safety, Security and Safeguards by Design in Small Modular Reactors (SMRs), Idaho Falls, Idaho, 11-15 September 2023.

Yamano, H. et. al., "Activities of the GIF safety and operation project of sodium-cooled fast reactor systems", *Joint IAEA-GIF Workshop on the Safety of Non-Water Coolant Reactors*, Vienna, May 2023.

Yamano, H. et. al. (2023), "Activities of the GIF safety and operation project of sodium-cooled fast reactor systems," *Proceedings of the 30th International Conference on Nuclear Engineering*, ICONE30-1117, Kyoto, May 2023.

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, Czechia, 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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