GIF R&D Outlook for Generation IV Nuclear Energy Systems:

2018 Update

GEN IV International Forum℠

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Foreword

by François Gauché
Chairman of the Generation IV International Forum (GIF) Policy Group (2016-2018)

The purpose of this 2018 update to 2009 GIF *R&D Outlook for Generation IV Nuclear Energy Systems* is twofold: firstly, to present the major outcomes achieved by the Forum partners in pursuing the R&D objectives specified in the 2014 update to the *GIF Technology Roadmap*; and secondly, to identify (and compare to the respective timeline) remaining or new challenges and the R&D needed to overcome them on the path to viability, performance demonstration or, in some cases, the demonstration/deployment of the respective fourth generation (Gen IV) system.

Since the publication of the *GIF R&D Outlook for Gen IV Nuclear Energy Systems* in 2009, nuclear energy has faced numerous, new challenges, including:

- unstable electricity prices in liberalised markets;
- electricity market structures that do not provide investment signals for low-carbon technologies;
- insufficient carbon pricing to promote nuclear investment;
- explicit government support for renewables in some countries;
- uncertain and changing political support;
- unstable licensing frameworks;
- poor social and political perception of nuclear energy safety (impact of the Fukushima Daiichi nuclear power plant accident);
- budget and schedule overruns during the construction of new nuclear power plants;
- the long-term nature of capital investment and significant financing costs.

Meeting these varied challenges is the driving force behind GIF's Gen IV systems R&D and methodology development work.

GIF is a multinational co-operative endeavour organised to carry out the R&D needed to establish the feasibility and performance capabilities of the next generation of nuclear systems. These systems aim to achieve improved sustainability, economics, safety and reliability, as well as proliferation resistance and physical protection. GIF's activities are striving to meet these objectives through technical, institutional and organisational innovation.

Gen IV concepts complement existing and evolutionary Gen III/III+ reactors, which will be deployed throughout the century, by providing additional options and applications, such as:

- optimisation of resource utilisation;
- multi-recycling of fissile materials/used fuel and reduction of the footprint of geological repositories for high-level waste;
- low-carbon heat supply for co-generation and high-temperature industrial applications (e.g. process steam, synthetic fuels, hydrogen production);
- enhanced integration of nuclear energy with other low-carbon sources.

Gen IV systems are taking into account lessons learnt from the Fukushima Daiichi accident by reinforcing the defence-in-depth approach against external events and promoting the robustness of the safety demonstration. Current R&D work focuses on enhancing the safety characteristics of Gen IV systems, aimed at excluding radioactive releases to the environment in case of accidents, eliminating the need for emergency measures, and minimising the impact on populations.

Some Gen IV systems may enter the demonstration/deployment phase in the next decade. To support these systems, GIF will ensure:

- best use of available experimental R&D infrastructure;
- support for the co-ordination of national programmes among GIF countries to avoid unnecessary duplication of facilities and ensure availability of key experimental infrastructure;
- best use of (multi-scale/multi-physics) simulation, as well as verification, validation and qualification (VVQ) tools as a complement to experimental programmes, to expedite the demonstration phase;
- best use of digital and product life management (PLM) tools to support the licensing phase and reduce time to market for innovative designs;
- support for technical and methodological innovation to reduce the costs of investment (overnight capital cost), to shorten and master the duration of construction (financing cost), and to optimise the costs of licensing, operation and maintenance (O&M), the fuel cycle, waste management and decommissioning (compared to existing nuclear power plants) at the design stage, in order to be competitive in the market;
- promotion of synergy between non-proliferation, physical protection and a robust safety design, thus strengthening a safety culture that is aimed at optimal integration of safety, security and safeguards for advanced fuel cycles and reactor concepts in order to achieve greater public acceptance.

Cross-cutting R&D activities will be pursued collaboratively to advance the technology in areas where common research and technology development needs have been identified (e.g. safety design methodologies, decay heat removal systems, advanced fuels and materials, advanced manufacturing, modelling and simulation, as well as VVQ tools).

GIF recognises the importance of attracting and educating the younger generation of nuclear energy scientists and engineers, and will continue to strengthen its efforts in education and training on advanced nuclear systems.

Finally, GIF will continue to engage with regulatory authorities and technical support organisations with the long-term goal of reaching the harmonisation of requirements and a better understanding of licensing approaches.

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Executive summary

Since the publication in 2009 of *Generation IV International Forum (GIF) R&D Outlook for Generation IV (Gen IV) Nuclear Energy Systems* [1], nuclear energy has been facing numerous challenges. These challenges include safety concerns after the Fukushima Daiichi nuclear power plant (NPP) accident, cost and regulatory uncertainties in a context where many electricity markets are dysfunctional as a result of low-cost gas and subsidised renewables, and construction cost and scheduling difficulties in relation to the ongoing construction of some new nuclear facilities.

Meeting these varied challenges is the driving force behind GIF's Gen IV systems research and development (R&D) and methodology development work.

GIF is a multinational, co-operative endeavour organised to carry out the R&D needed to establish the feasibility and performance capabilities of the next generation of nuclear systems. These systems aim to improve sustainability, economics, safety and reliability, proliferation resistance and physical protection. The Forum's activities strive to meet these objectives through technical, institutional and organisational innovation.

GIF is pursuing technological innovations and advanced reactor designs aimed at reducing costs, increasing revenues by addressing new market opportunities (supplying heat, hybrid energy systems, dispatchable energy, etc.) and boosting thermal efficiency. Gen IV systems take advantage of high-temperature coolants such as helium, liquid metals, liquid salts, or supercritical water, which increases design flexibility, allowing for an up to 40% improvement in thermal efficiency. It also provides the option of industrial heat applications that can vastly displace fossil fuel use. When used with an advanced fuel cycle, Gen IV systems can deliver up to a 100-fold improvement in uranium use. Additionally, advanced recycling options enable the reduction of the volume and radio-toxicity of waste to be stored in deep geological repositories.

These performance advantages can be realised while maintaining or improving upon today's stringent safety standards for new nuclear plants. GIF is also investigating the need for dispatchable energy in electricity and heat markets, which is considered to be a major strategic issue. In fact, some countries are moving towards hybrid energy mixes with a high percentage of renewables, and a strong need for energy storage and flexibility in energy supply.

GIF also recognises the need to support institutional innovation worldwide, for example through the effective sharing of international safety standards. A dedicated GIF Task Force is working to define safety design criteria (SDC) and guidelines (SDG) for next-generation sodium-cooled fast reactor (SFR) designs. Its work represents an important first step towards helping regulators become familiar with the technical characteristics of non-water-cooled reactor technologies such as Gen IV systems and the associated safety research conducted within GIF. Exchanges with experts from the International Atomic Energy Agency (IAEA) in this topical area, and the extension of this work to other Gen IV systems, will certainly help pave the way towards the development of international safety standards for non-water-cooled reactor technologies.

In respect of organisational innovation, the interest of private capital in advanced reactors (small modular reactor [SMR] and Gen IV systems) in some countries constitutes a positive signal for the nuclear industry. Moreover, it represents an obvious opportunity to attract young skilled scientists and engineers. The GIF members are confident that the Forum can deliver new ideas and help establish new business models for the development and application of innovative nuclear energy systems. Considering the importance of renewing the workforce in the nuclear energy field, in 2015 GIF launched a task force to provide a platform to enhance open education and training (E&T), and to facilitate networking between individuals and organisations involved in the development of Gen IV systems.

Having defined the objectives of the next ("fourth") generation of nuclear systems (sustainability, safety and reliability, economic competitiveness, and proliferation resistance and physical protection), the original GIF *Technology Roadmap* of 2002 identified the following six nuclear energy systems as being the most promising to meet its objectives, assuming a deployment horizon beyond 2030 [2]:

- sodium-cooled fast reactor (SFR);
- very high temperature reactor (VHTR);
- gas-cooled fast reactor (GFR);
- molten salt reactor (MSR);
- lead-cooled fast reactor (LFR);
- supercritical water-cooled reactor (SCWR).

The roadmap then went on to define, for each of the six Gen IV systems, the research and technology development efforts needed to achieve the stated objectives. It established indicative system development timelines based on three phases (viability, performance, and demonstration/deployment).

Ten years later, in preparing an updated *Technology Roadmap*, GIF has taken stock of its achievements and has charted a path forward. In doing so, it became evident that funding for, and the pace of research and technology development activities of, Gen IV systems differed from one GIF country to another. This difference results mainly from the pace and/or the reorientation of national programmes, which has led to uneven technical progress being achieved for the six Gen IV systems. More specifically, the funding levels in GIF countries involved in the SFR and VHTR R&D have exceeded those of the other Gen IV systems, primarily for historical reasons. This situation has led to the revision of the system development timelines in the GIF *Technology Roadmap Update* published in January 2014 [3]. By the same token, the update identified, for each of the six Gen IV systems, the major R&D objectives and development milestones for the next decade.

The main purpose of this 2018 update of the *Gen IV R&D Outlook* is to:

- review progress achieved for each system against the major R&D objectives specified in the updated *Technology Roadmap*;
- identify (and compare to the respective, revised timeline) the remaining obstacles, and the R&D efforts needed to surmount them, on the path towards the demonstration of viability or performance, or in some cases, towards the demonstration/deployment of the respective Gen IV system.

Chapter 1 of this report describes the critical need for nuclear energy in meeting the world's energy challenges and how GIF works to advance its contribution. Chapter 2 provides a description of the six Gen IV systems. Chapter 3 summarises the major achievements of GIF partners in response to the R&D challenges identified in the updated 2014 roadmap, as well as the 2018 outlook. The chapter specifies the challenges that Gen IV systems face in the current decade, the R&D needed to meet them and respective R&D plans. Chapter 4 summarises the major achievements and the outlook of the three methodology working groups (economics, risk and safety, proliferation resistance and physical protection). Chapter 5 is dedicated to the initiatives implemented in support of Gen IV R&D (education and training, regulatory challenges, market opportunities, the Senior Industrial Advisory Panel, research infrastructure needs, and advanced materials engineering and manufacturing).

The report concludes with a summary of GIF's major accomplishments and an outlook for the period 2018–2028, covering overall vision, challenges, R&D needs and plans for Gen IV systems, methodological working groups, task forces and cross-cutting initiatives in view of their respective objectives (i.e. viability, performance, or demonstration/deployment).

References

[1] GIF (2009), *R&D Outlook for Generation IV Nuclear Energy Systems*, www.gen-4.org/gif/upload/docs/application/pdf/2013-10/gif_rd_outlook_for_generation_iv_nuclear_energy_systems_2013-09-30_15-49-32_599.pdf.

[2] GIF (2002), *A Technology Roadmap for Generation IV Nuclear Energy Systems, GIF-002-2002*, www.gen-4.org/gif/jcms/c_40473/a-technology-roadmap-for-generation-iv-nuclear-energy-systems.

[3] GIF (2014), *Technology Roadmap Update for Generation IV Nuclear Energy Systems*, www.gen-4.org/gif/upload/docs/application/pdf/2014-03/gif-tru2014.pdf.

Chapter 1. An essential role for nuclear energy and the contribution of the Generation IV International Forum

The world's population is expected to expand from 7.6 billion people today to over 9 billion people by the year 2050, with each individual striving for a better quality of life. As the earth's population grows, so does the demand for energy and the benefits that it brings: improved standards of living, better health and longer life expectancy, improved literacy and opportunities, and much more. Simply expanding the use of energy along the same mix of today's production options, however, does not satisfactorily address concerns over climate change and depletion of fossil resources. For the earth to support its population, while ensuring the sustainability of humanity's development, it will be important to increase the use of clean, safe and cost-effective energy supplies, which can serve for both basic electricity production and other primary energy needs. Prominent among such supply options is nuclear energy.

The International Energy Agency (IEA) [1] estimates that the emission of almost 60 gigatonnes of CO_2 ($GtCO_2$) has been avoided through the use of nuclear energy since 1971. Hydro and nuclear currently dominate clean electricity production. Nuclear electricity, for its part, avoids global emissions of about 1.7 $GtCO_2$ annually. While producing no carbon emissions during electricity generation, nuclear energy has a very small life cycle carbon footprint originating from fossil fuels used during mining, and the manufacturing and transport of materials and components. Moreover, nuclear energy can help smooth the integration of renewable, albeit intermittent, capacity into electric grid systems. Consequently, the IEA projects that nuclear energy will have a cornerstone role in climate-friendly scenarios because the technology allows for the most cumulative CO_2 emission avoidance [2]. For a strategy designed to limit global temperature increase to 2°C, nuclear capacity would need to more than double by 2050, and represent 17% of worldwide electricity production.

In 2009, the Generation IV International Forum (GIF) published its *R&D Outlook for Generation IV Nuclear Energy Systems* [3]. The main purpose of the present update is, firstly, to review progress achieved for each of the six systems against the major R&D objectives specified in the GIF *Technology Roadmap Update* [4], and secondly, to identify (and compare to the respective, revised timeline) remaining obstacles, as well as the R&D efforts needed to surmount these obstacles on the path towards the demonstration and/or deployment of the respective systems.

Since 2009, nuclear energy has been facing numerous challenges, unstable electricity prices in liberalised markets;

- electricity market structures that do not provide investment signals for low-carbon technologies;
- insufficient carbon pricing to promote nuclear investment;
- explicit governmental support for renewables in some countries;
- uncertain and changing political support;
- unstable licensing frameworks;
- poor social and political perceptions of the safety characteristics of nuclear energy (impact of the Fukushima Daiichi NPP accident);
- budget and scheduling overruns at new, nuclear plant construction sites;
- the long-term nature of capital investment and significant financing costs.

Meeting these varied challenges is the driving force behind the GIF's Generation IV (Gen IV) systems R&D and methodology development work. GIF is a multinational co-operative endeavour organised to carry out the R&D needed to establish the feasibility and performance capabilities of the next generation of nuclear systems. These systems aim to achieve improved sustainability, economics, safety and reliability, proliferation resistance and physical protection. GIF activities are striving to meet these objectives through technical, institutional and organisational innovation.

GIF is pursuing technological innovations and advanced reactor designs aimed at reducing costs, increasing revenues by addressing new market opportunities (supplying heat, hybrid energy systems, dispatchable energy, etc.) and boosting thermal efficiency. Gen IV systems take advantage of high-temperature coolants such as helium, liquid metals, liquid salts, or supercritical water, which increases design flexibility, allowing for an up to 40% improvement in thermal efficiency. It also provides the option of industrial heat applications that can vastly displace fossil fuel use. When used with an advanced fuel cycle, Gen IV fast-neutron systems can deliver up to a 100-fold improvement in uranium use. Additionally, advanced recycling options enable the reduction of the volume and radio-toxicity of waste to be stored in deep geological repositories.

These performance advantages can be realised while maintaining or improving upon today's stringent safety standards for new nuclear plants. GIF is also investigating the need for dispatchable energy in electricity and heat markets, which is considered to be a major strategic issue since some countries are moving towards hybrid energy mixes with a high percentage of renewables and a strong need for energy storage and flexibility in energy supply.

GIF recognises the need to support institutional innovations worldwide, for instance through the effective sharing of international safety standards. One example is the harmonisation of regulatory requirement activities within the framework of the OECD/NEA Multinational Design Evaluation Programme (MDEP) sub-committee on safety goals. The objective of this sub-committee is to achieve as much convergence in light water reactor (LWR) safety goals as possible [5]. A dedicated GIF task force is working to define safety design criteria (SDC) and safety design guidelines (SDG) for next-generation, sodium-cooled fast reactor (SFR) designs. The work of this task force represents an important first step towards helping regulators become familiar with the technical characteristics of Gen IV systems and the associated safety research conducted within GIF. Exchanges with experts from the International Atomic Energy Agency (IAEA) in this topical area, and the extension of this work to other Gen IV systems, will help to pave the way towards the development of international safety standards.

In terms of organisational innovations, the interest of private capital in advanced reactors (small modular reactors [SMR] and Gen IV systems) in some countries constitutes a positive signal for the nuclear industry. Moreover, it represents an obvious opportunity to attract young, skilled scientists and engineers. The GIF members are confident that the Forum can deliver new ideas and help establish new business models for the development and application of innovative nuclear energy systems. Considering the importance of renewing the workforce in the field of nuclear energy, in 2015 GIF launched a task force to provide a platform to enhance open education and training (E&T) and facilitate networking between individuals and organisations involved in the development of Gen IV systems.

References

[1] IEA (2014), *World Energy Outlook 2014*, Paris: OECD/IEA.

[2] IEA (2015), *Energy Technology Perspectives 2015*, Paris: OECD/IEA.

[3] GIF (2009), *R&D Outlook for Generation IV Nuclear Energy Systems*, www.gen-4.org/gif/upload/docs/application/pdf/2013-10/gif_rd_outlook_for_generation_iv_nuclear_energy_systems_2013-09-30_15-49-32_599.pdf.

[4] GIF (2014), *Technology Roadmap Update for Generation IV Nuclear Energy Systems*, www.gen-4.org/gif/upload/docs/application/pdf/2014-03/gif-tru2014.pdf.

[5] NEA (2011), *The Structure and Application of High-Level Safety Goals: A Review by the Multinational Design Evaluation Programme Sub-committee on Safety Goals*, Paris, OECD, www.oecd-nea.org/mdep/documents/MDEP-SAHLSG-Jan2011.pdf.

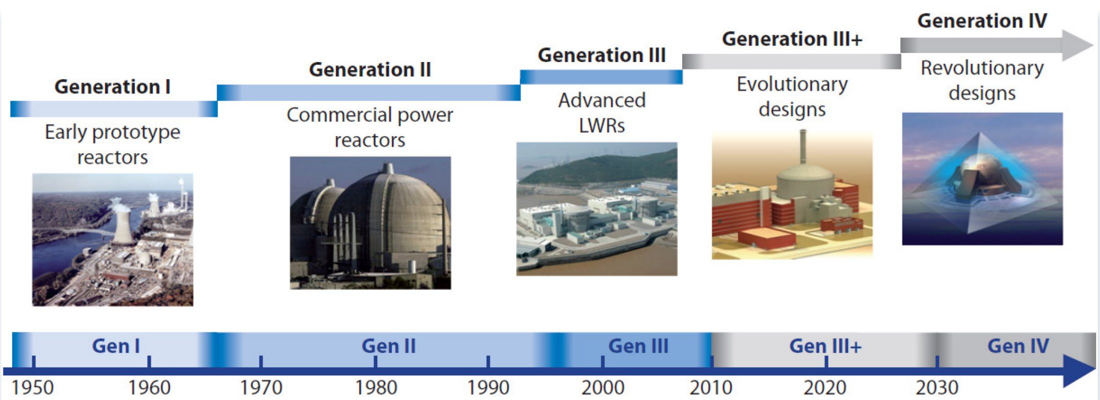
Chapter 2. Generation IV nuclear energy systems

2.1. High-level R&D goals for Generation IV nuclear energy systems

The Generation IV International Forum (GIF) defined four goals in its 2002 *Technology Roadmap* [1] to move nuclear energy forward into its next, fourth generation (i.e. Gen IV) (see Figure 1):

- sustainability;
- safety and reliability;
- economic competitiveness;
- proliferation resistance and physical protection.

Figure 1: **The four generations of reactor designs**



The roadmap identified the following six nuclear energy systems as being the most promising to meet its objectives, assuming a deployment horizon beyond 2030:

- sodium-cooled fast reactor (SFR);
- very high temperature reactor (VHTR);
- gas-cooled fast reactor (GFR);
- molten salt reactor (MSR);
- lead-cooled fast reactor (LFR);
- supercritical water-cooled reactor (SCWR).

The *Technology Roadmap Update*, published in 2014, also defined and planned the R&D required to achieve the four goals listed above and to enable the deployment of Gen IV nuclear energy systems from 2030 [2]. Gen IV nuclear energy systems include the nuclear reactor and its energy conversion systems, as well as the requisite fuel cycle technologies. Moreover, the roadmap established indicative system development timelines based on three phases (viability, performance and demonstration phase).

2.2. Missions for Gen IV nuclear energy systems

While the evaluation of systems for their potential to meet all four goals was a central focus of the roadmap, it was recognised that countries would have a range of perspectives on the priority applications or objectives for Gen IV systems.

GIF today identifies the following missions for Gen IV systems: the generation of electricity; the ability to produce non-electric products such as hydrogen or to process heat; the minimisation of waste; and cost-effective integration of Gen IV systems in a global low-carbon energy system.

All six systems have electricity applications. The higher temperature systems (VHTR, GFR, LFR and MSR) have potential applications in the production of hydrogen or industrial process heat for such chemical processing facilities as petroleum refineries. The three fast reactor systems and the MSR have the capability to transmute actinides for effective waste management.

It is likely that the role of intermittent renewables will increase to a significant level when Gen IV reactors are ready for deployment, requiring Gen IV reactors to be even more flexible to meet the variable demand from the grid and to provide frequency regulation. Developers of the Gen IV systems will have to take into consideration this flexible operation requirement and associated thermal cycling and fatigue, reactivity control and fuel optimisation in the design of the reactors.

However, policy makers and engineers could also investigate other options to ensure a robust, cost-effective and flexible energy supply system, such as:

- large-scale energy storage and cogeneration (hydrogen, heat) applications allowing for flexible operation of Gen IV reactors;
- hybrid energy systems, consisting of coupled nuclear and renewable generators with adequate energy storage and cogeneration applications that would meet flexible demand from the grid while operating power generators at full capacity, thus ensuring overall economically viable operation.

As recommended by the Senior International Advisory Panel (SIAP), GIF is in any case progressing towards a "system" approach to flexibility in a broad sense, addressing operational flexibility (manoeuvrability, compatibility with hybrid systems, island mode operation, diversified fuel use), deployment (scalability, siting, constructability), and products (electricity, process heat).

It will also be necessary to establish metrics to evaluate Gen IV concepts with respect to flexibility (and other criteria tied to design maturity), and to recognise that deployment of new build is sensitive to drivers beyond cost (e.g. policies, types of service and types of electricity markets). See for instance the EPRI white paper on "Expanding the Concept of Flexibility" [3].

In preparing its updated *Technology Roadmap* [2], GIF has taken stock of its achievements and has charted a path forward. It has recognised that both funding for, and the pace of, research and technology development activities in relation to the Gen IV systems has differed from one GIF country to another. This difference results mainly from the pace and/or reorientation of national programmes, which has resulted in uneven technical progress in the case of the six Gen IV systems. More specifically, the funding levels in GIF countries involved in SFR and VHTR R&D have exceeded those focused on the other Gen IV systems, primarily for historical reasons, which has led to the revision of the system development timelines in the updated *Technology Roadmap*. By the same token, the update identified, for each of the six Gen IV systems, major R&D objectives and development milestones for the next decade.

2.3. Description of the six Gen IV nuclear energy systems

Sodium-cooled fast reactor (SFR)

The SFR uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction. While the oxygen-free environment prevents corrosion, the sodium reacts chemically with air and water and thus requires a sealed coolant system.

Plant size options under consideration range from small, 50 to 300 megawatt electrical (MW_e) modular reactors to larger plants up to 1 500 MW_e. The outlet temperature range is 500-550°C for the options, which affords the use of materials developed and proven in prior fast reactor programmes. The reactor unit can be arranged in a pool layout or a loop layout.

Reference concepts

Several SFR design concepts are being considered within the GIF framework:

- The Japan Sodium Fast Reactor (JSFR) and the European Sodium Fast Reactor (ESFR) represent the large size loop-type and pool-type reactor concepts, respectively, with mixed uranium-plutonium oxide fuel, and potentially minor actinides, supported by a fuel cycle based on advanced aqueous processing at a central location serving a number of reactors. The Russian BN-1200 reactor, which is a large size, pool-type reactor, has recently been added to the concepts being considered.
- The Korean KALIMER-600 (KAERI) represents the intermediate size pool-type reactor concept with oxide or metal fuel.
- The AFR-100 (US Department of Energy or DOE) represents small size modular-type reactors with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor.

Figures 2 and 3 below are sketches of the AFR-100 and BN-1200, two of the most recent SFR design concepts being considered by GIF.

Figure 2: **AFR-100 reactor**

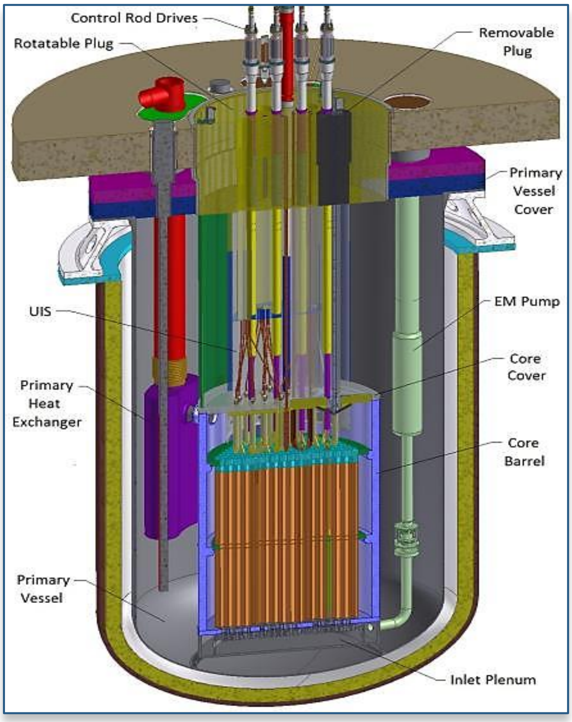
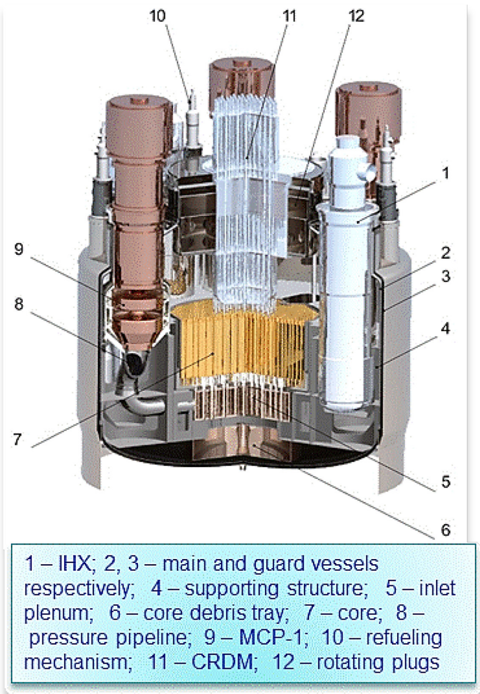


Figure 3: **BN-1200 reactor**



IHX – Intermediate heat exchanger; MCP – main coolant pump; CRDM – Control rod drive mechanism.

Performance

The SFR closed fuel cycle enables regeneration of fissile fuel and facilitates management of high-level waste – in particular, plutonium and minor actinides. However, this regeneration requires that recyclable fuels be developed and qualified for use.

Important safety features of the Gen IV systems include a long thermal response time, a reasonable margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the power conversion system. Water/steam, nitrogen and supercritical carbon dioxide (CO_2) are considered working fluids for the power conversion system to achieve high performance in thermal efficiency, safety and reliability.

Much of the basic technology for the SFR has been established in former fast reactor programmes (Russia, France, United States, United Kingdom, Japan). As for recent achievements, the Chinese Experimental Fast Reactor (CEFR) reached 100% power in 2014. In Russia, the BN-600 achieved 86% load factor in 2015, and the BN-800 achieved 100% power in 2016. As for new construction plans, China is planning to build the CFR-600 by 2023, Russia is considering the construction of the BN-1200, and India is planning to operate its Prototype Fast Breeder Reactor (PFBR) in 2018.

Very high temperature reactor (VHTR)

The VHTR is the next generation in the development of high temperature reactors with ceramic fuel, using graphite as a moderator and helium gas as a coolant. The technology basis, current development status and co-operation in GIF are described in [4]. The VHTR is of interest for high efficiency power generation, including in arid regions, and for cogeneration of electricity and process heat, e.g. for bulk hydrogen production or process heat in numerous industry branches. Owing to the high efficiency of cogeneration, the VHTR can greatly reduce the emission of CO_2

and noxious pollutants produced by the combustion of fossil fuels in power plants and industry, and thus enhance security of energy supply in those countries where this is an issue.

Design objectives

Hydrogen can be produced from water by using thermochemical, electrochemical or hybrid processes. The original target for the GIF VHTR outlet temperature was set at 900-1 000°C because one of the main drivers for the technology was initially large-scale bulk hydrogen production with the iodine-sulphur (I-S) process. This process consumes heat at 850°C and thus, accounting for temperature cascades in heat exchangers, calls for 900-1 000°C at the reactor outlet, keeping in mind that temperatures above 950°C would require development and qualification of innovative materials such as new super alloys, ceramics and compounds.

In the meantime, market studies in several GIF signatory countries have confirmed the existence and development potential of a significant near-term market for lower-temperature applications, especially process steam below 600°C, which would necessitate less stringent reactor outlet temperatures of approximately 750°C. This steam can then be used either for electricity generation (exploiting commonalities with the most advanced coal-fired power plant machinery to reduce cost), and/or for industrial applications where the steam is used as a reactant for chemical processes or as a heat carrier.

Therefore, the VHTR system is being considered for process heat in general and not specifically dedicated to hydrogen production via the I-S process. Several nearer-term VHTR projects are looking at steam production for most or all energy needs of their customers. Direct helium gas turbines or indirect (gas mixture turbine) Brayton-type cycles are also being considered for near-term application in several countries.

Performance

VHTRs have demonstrated inherent safety features in a number of past and current prototypes and engineering-scale tests, and all modern concepts are designed to remove decay heat passively. Modular construction and the elimination of active safety systems are expected to result in low equipment, operation and maintenance costs. Two baseline options are available for the VHTR core: the pebble bed type and the prismatic block type, both of which rely on fully ceramic fuel elements containing tristructural-isotropic (TRISO) coated particles that have been demonstrated to very reliably contain radioactivity even at high burnup and under extreme loss of forced cooling scenarios.

A robust fuel and graphite moderation enable high once-through burnup (150-200 gigawatt days per tonne of heavy metal [GWd/tHM]) in low-enriched uranium (LEU) fuel enriched to 10-20%. High conversion and the use of transuranics can be achieved with thorium or recycled plutonium, within the physical limits of a thermal neutron system. High conversion thorium-uranium fuels were used in early demonstration reactors (Fort St Vrain and Thorium Hochtemperatur Reaktor [THTR]), and other fuel cycles were investigated extensively in the early programmes. Deep burn and the incineration of weapons-grade plutonium were also deemed feasible in more recent studies. Existing reprocessing technologies can be used on TRISO fuel with the addition of a suitable head-end process required to separate the fuel from the graphite and breach the particle layers.

Reference concepts

The basic technology for the VHTR was established in former high-temperature gas reactors such as the US Peach Bottom and Fort St Vrain plants, as well as the German Arbeitsgemeinschaft Versuchsreaktor (AVR) and THTR prototypes. The technology has benefitted from extensive R&D projects internationally, as well as from former and ongoing projects, such as the High-Temperature Reactor-Pebble bed Module (HTR-PM, China), the Gas Turbine High-Temperature Reactor (GTHTR300C, Japan), PBMR (South Africa), ANTARES (France, United States), the Nuclear Hydrogen Development and Demonstration (NHDD, South Korea) project, the Gas Turbine Modular Helium Reactor (GT-MHR, United States, Russia) and the Next-Generation Nuclear Plant (NGNP, United States), led by several plant vendors and national laboratories.

Experimental reactors, such as the High-Temperature Engineering Test Reactor (HTTR) (Japan, 30 megawatts thermal [MW_{th}]) and HTR-10 (China, 10 MW_{th}), support this advanced reactor concept development, together with the cogeneration of electricity and hydrogen or a range of other nuclear heat applications. Another concept, the fluoride-salt-cooled high-temperature reactor (FHR), uses a combination of VHTR fuel and molten salt instead of helium coolant. Within GIF, this technology is covered in the MSR system. VHTR development also has limited synergies with the Gen IV GFR system, for example in the area of structural materials and components.

Figure 4 schematically presents three recent VHTR designs, which were submitted to the GIF SIAP for advice by the designers in 2017 with selected performance parameters, listed in Table 1. The performance numbers refer mainly to electricity generation, although all reactors are suitable for cogeneration with various fractions of heat and power.

Figure 4: **Examples of recent VHTR designs submitted to the SIAP**

HTR-PM

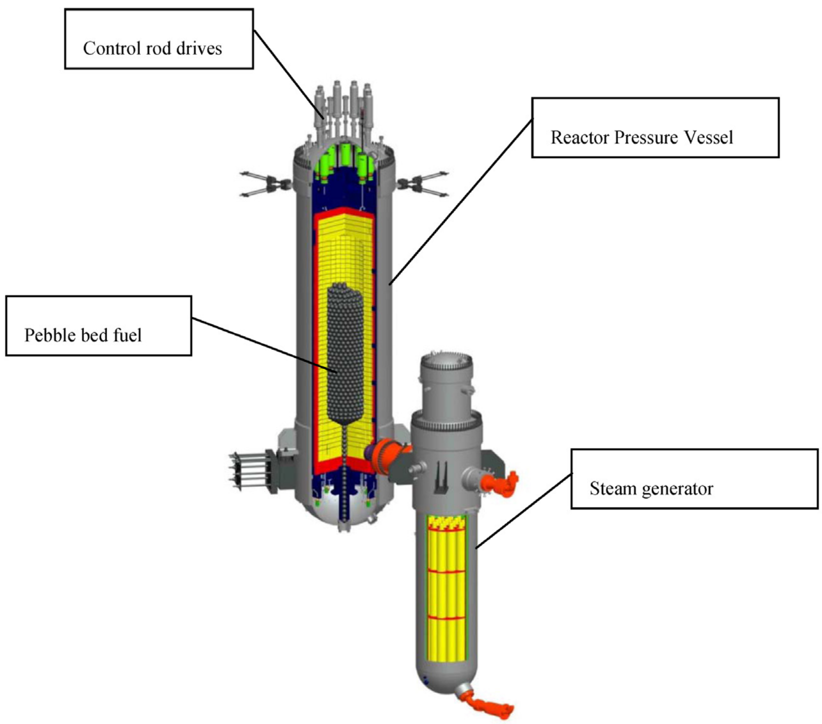
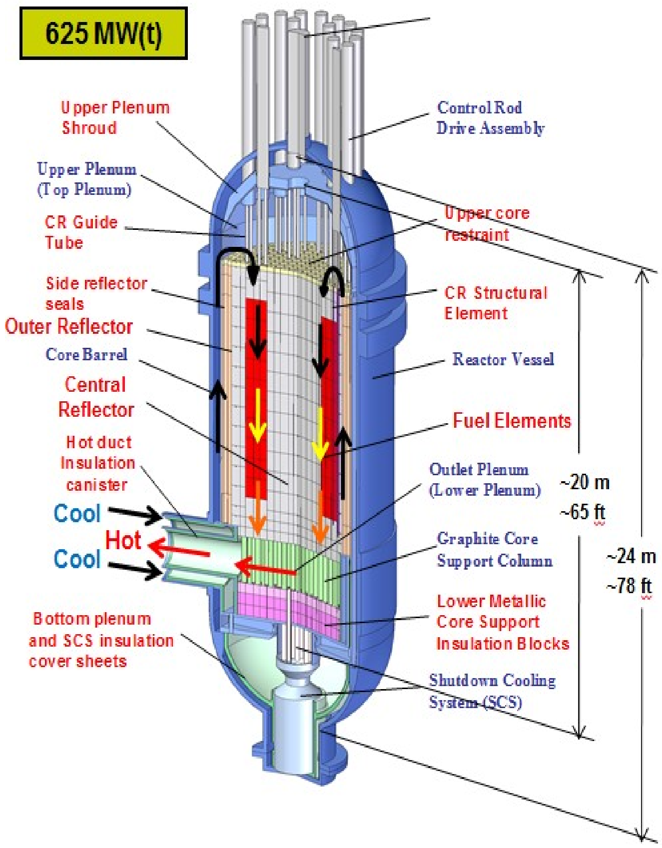
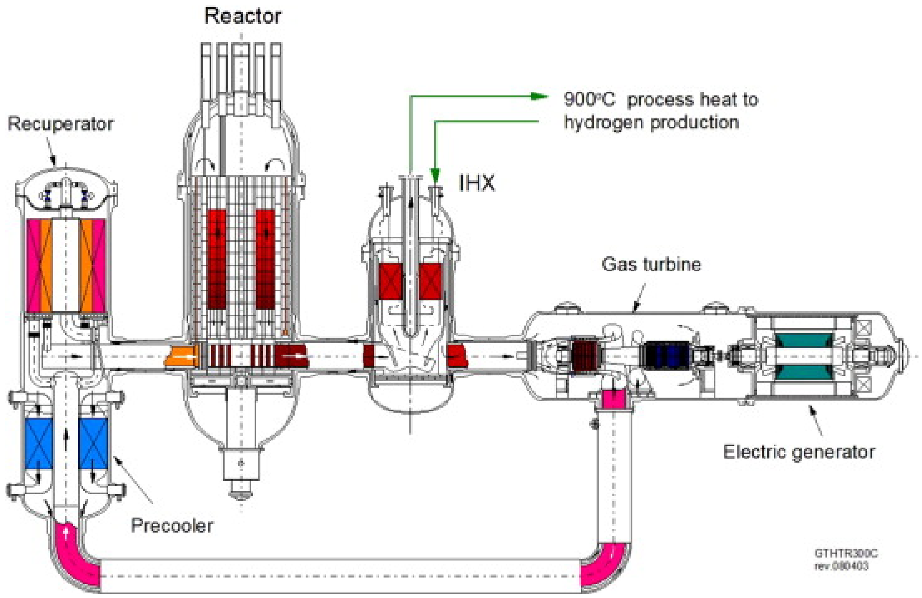


Figure 4: **Examples of recent VHTR designs submitted to the SIAP** *(cont'd)*

SC-HTGR



GTHTR300C



Note: SC-HTGR = Steam Cycle High-Temperature Gas-cooled Reactor.

Table 1: **Selected performance parameters for recent VHTR designs**

	HTR-PM	SC-HTGR	GTHTR300C
Development status	Pre-FOAK under construction	FOAK design	FOAK design
Designer	INET (China)	Framatome (United States)	JAEA (Japan)
Power plant architecture	Modular	Modular	Modular
Thermal output [MW_{th}]	2 × 250	n × 625	n × 600
Electric output [MW_e]	210	272	274
Core configuration	Pebble bed Cylindrical 420 000 pebbles/core	Prismatic block Annular 102 columns 10 blocks/column	Prismatic block Annular 90 columns 8 blocks/column
Fuel	TRISO-coated particles in pebbles UO_2 8.9% enriched once-through	TRISO-coated particles in compacts UCO 10.36-15.5% enriched once-through	TRISO-coated particles in compacts UO_2 14.3% enriched once-through
Primary He pressure [MPa]	7	6	6.9
He inlet/ outlet temperature [°C]	250/ 750	325/ 750	587-663/ 850-950
Secondary coolant	Water/steam	Water/steam	Helium
Power conversion	Rankine Steam turbine	Rankine Steam turbine	Direct Brayton Gas turbine H_2 production
Number of steam generators/HX per core	1	2	1
Max. secondary pressure [MPa]	13.24	16.7	tbd (heat transfer circuit to H_2 production)
Max. secondary temperature [°C]	566	566	tbd (heat transfer circuit to H_2 production)

Notes: FOAK = first of a kind; He = helium; HX = heat exchangers; H_2 = hydrogen; MPa = megapascal; tbd = to be defined; UCO = uranium oxycarbide; UO_2 = uranium dioxide.

2.3.3 GFR

The gas-cooled fast reactor (GFR) is an innovative nuclear system that combines fast neutrons and high temperature. The GFR cooled by helium is proposed as a longer-term alternative to sodium-cooled fast reactors. The main advantages of the GFR, besides allowing adoption of the closed fuel cycle, are:

- a high operating temperature, allowing increased thermal efficiency and generation of high temperature heat for industrial applications similar to the VHTR;
- a chemically inert and a non-corrosive coolant (helium);
- a single phase (no boiling) coolant;
- relatively small (albeit positive) coolant void reactivity coefficient;

- a coolant that does not dissociate nor activate;
- a coolant that is transparent, facilitating in-service inspection and repair, as well as fuel handling.

The main drawbacks are related to:

- the need to operate under pressurised conditions;
- the low cooling efficiency of helium, in particular in decay heat removal conditions by natural convection;
- the need for additional coolant inventories to make up for coolant loss through leakage.

Design objectives

Key reference objectives are high outlet temperature (850°C) for high thermal efficiency and hydrogen production, and direct cycle for compactness. Unit power will be considered in a range from 200 MW_e (modularity) up to a larger 1 500 MW_e.

The objective of high fuel burnup together with actinide recycling results in used fuel characteristics (isotopic composition) that are unattractive for handling.

Designers agree to aim at minimising feedstock usage with a self-sustaining cycle, which only requires depleted or reprocessed uranium feed, which calls for a self-generating core with a breeding gain near zero. It is recommended that the initial plutonium inventory in the GFR core be not much higher than 15 tonnes per GW_e. in order not to penalise the long-term deployment of GFR, based on consideration of both the available plutonium stockpiles foreseen (mainly derived from water reactor irradiated fuels) and the period of time for GFR fleet development.

Reference concept

The reference concept for GFR is a 2 400 MW_{th} [5] plant having a break-even core, operating with a core outlet temperature of 850°C, enabling an indirect combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high power density necessary for good neutron economics in a fast reactor core.

The core consists of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed carbide-fuelled pins contained within a ceramic hex tube. The favoured material at the moment for the pin clad and hex tubes is silicon carbide fibre-reinforced silicon carbide (SiCf/SiC). The whole primary circuit with three loops is contained within a secondary pressure boundary, the guard containment. The produced heat is converted into electricity in the indirect combined cycle with three gas turbines and one steam turbine. The cycle efficiency is approximately 48%.

A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle containing a helium-nitrogen mixture, which in turn drives a closed-cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator, which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plants and so represents an established technology, with the only difference in the GFR case being the use of a closed-cycle gas turbine.

Figure 5: **Reference GFR design (2 400 MWth)**

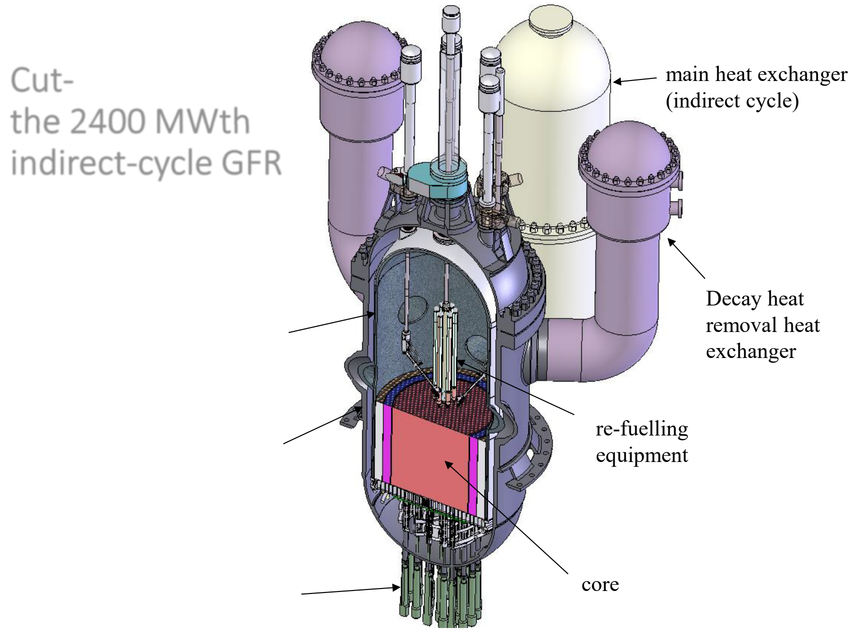


Table 2: **Reference GFR characteristics**

Nominal power (thermal/electrical)	MW	2 400/1 150 (48%)
Fuel/cladding		UPuC/SiC-SiCf
Power density	MW/m^3	100
Pu content	%	16.3
Breeding gain		0
Number of primary loops		3
Primary coolant		Helium
Primary pressure	MPa	7
Core inlet/outlet temperature	°C	400/850
Number of secondary loops		3
Secondary coolant		Helium-nitrogen
Secondary pressure	MPa	6.5
Power conversion system		Closed-cycle gas turbine and steam generator

Notes: MW = megawatt; MW/m^3 = megawatt per cubic metre; UPuC = uranium-plutonium carbide.

Molten salt reactor (MSR)

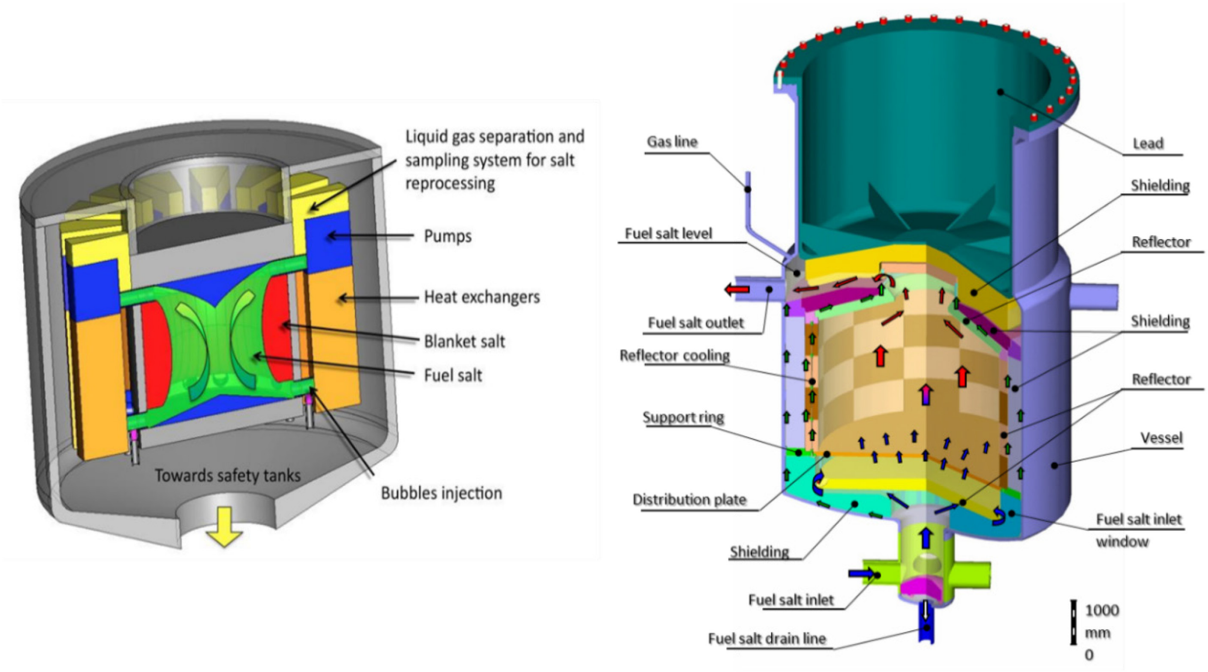
An MSR is a reactor that, to a significant degree, uses molten salt within the core as fuel carrier or coolant. There are two main MSR subclasses:

- Fissile material is dissolved in the molten salt, which serves both as fuel carrier and coolant in the primary circuit.
- Molten salt is the coolant in a graphite-moderated core fuelled with ceramic fuel, similar to that employed in VHTRs. This solid-fuel variant is typically referred to as an FHR, in order to distinguish it from the previous one.

Reference concepts

In liquid fuel MSRs, the fissile material is part of the liquid coolant. Significant elements of the MSR technology were developed and demonstrated in the 1950s and 1960s in the United States. While early MSR development focused on thermal neutron spectrum concepts, the liquid fuel MSR concepts under development by GIF members, following GIF's sustainability objectives, have been fast neutron spectrum reactors with circulating fluoride-based fuel in a closed fuel cycle. Specifically, the MSR designs under development within the GIF framework are the liquid fluid Molten Salt Fast Reactor (MSFR) (European Union) and Molten Salt Actinide Recycler and Transmuter MOSART (Russia), as well as the solid-fuel FHR demonstration reactor (DR) (United States). Figure 6 depicts MSFR and MOSART designs.

Figure 6: **Liquid fuel MSFR and MOSART designs**



The 1 400 MW_e thorium-uranium MSFR consists of a 2.25 metre equal diameter and height cylinder made of a nickel-based alloy, filled with the liquid fuel salt under ambient pressure conditions, operating at temperatures up to 750°C. The fuel salt in the primary circuit is pumped around in upward direction through the central core zone and in downward direction through the heat exchangers located circumferentially around the core. In between is a thorium-fluoride salt-filled container to increase the breeding gain.

The fast spectrum reactor can be operated in the full range from breeder to burner mode. This flexibility is facilitated by the fact that the fuel salt composition can easily be adapted during reactor operation, since there is no need to manufacture solid fuel elements. In addition to the large MSFR considered so far and in view of identifying the potential limits of such concepts, small and medium-sized reactor (SMR) designs might mitigate some of the safety constraints identified for large power concepts, with their high fuel salt power density. Because of this, these SMRs may be the first type of MSFR to be developed, and related R&D studies are due to be launched in the near future to evaluate industrialisation viability and define the industrialisation process.

The 1 000 MW_e MOSART can operate as an efficient transuranics burner, or as a breeder when using the thorium fuel cycle. The homogeneous Li, Be/F MOSART core without graphite moderator has an optimum fast neutron spectrum with neutron flux levels of nearly $1x10^{15} n \times cm^{-2} \times s^{-1}$. The MOSART concepts offers several attractive features:

- simple configuration (no solid moderator or structural materials exposed to high neutron irradiation fields);
- proliferation resistant, multiple recycling of actinides and proliferation resistance features (separation coefficients between the transuranics and the lanthanide groups are high, but very low within the transuranics group);
- proven container materials (Ni-based alloys) and system components (pump, heat exchanger, etc.) operating in the fuel circuit at temperatures below 750°C;
- inherent safety characteristics (large negative temperature reactivity coefficient around -3.7 pcm/K);
- long, soluble fission product removal period (one to three equivalent full-power years).

Following trends in diversification of concepts using solid fuels and molten salt coolants, the GIF MSR preliminary System Steering Committee has included the FHR DR in its work scope. The FHR DR, a 100 MW_{th} reactor, uses TRISO particle fuel within prismatic graphite blocks. FLiBe (stoichiometric 2:1 compound $^7Li_2BeF_4$ of ^7LiF and BeF_2) is the reference primary coolant. The FHR DR is designed to be small, simple and affordable; its development is a necessary intermediate step to enable near-term commercial FHRs. Lower-risk technologies are purposely included in the initial FHR DR design to ensure that the reactor can be built, licensed and operated within an acceptable budget and schedule. These technologies include TRISO particle fuel, replaceable core structural material, use of that same material for the primary and intermediate loops, and tube-and-shell primary-to-intermediate heat exchangers.

LFR

The lead-cooled fast reactor (LFR) features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used to manage actinides from used fuel and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of a relatively inert coolant, provided that the effects of weight, and the corrosive and erosive nature of lead, can be overcome. The LFR has the potential to meet the electricity needs of remote sites as well as to be deployed as a large grid-connected power station.

Reference concepts

The designs that are currently proposed as reference concepts in the frame of Gen IV activities are three pool-type reactors:

- a reference design of 600 MW_e based on the European Lead Fast Reactor (ELFR), representing a large central electrical power generation unit;
- a 300 MW_e option developed by Russia, which is in a well-advanced stage of development (BREST-OD-300);
- a small modular design of 20 MW_e, based on the Small Secure Transportable Autonomous Reactor (SSTAR), representative of the small size initiatives related to LFR.

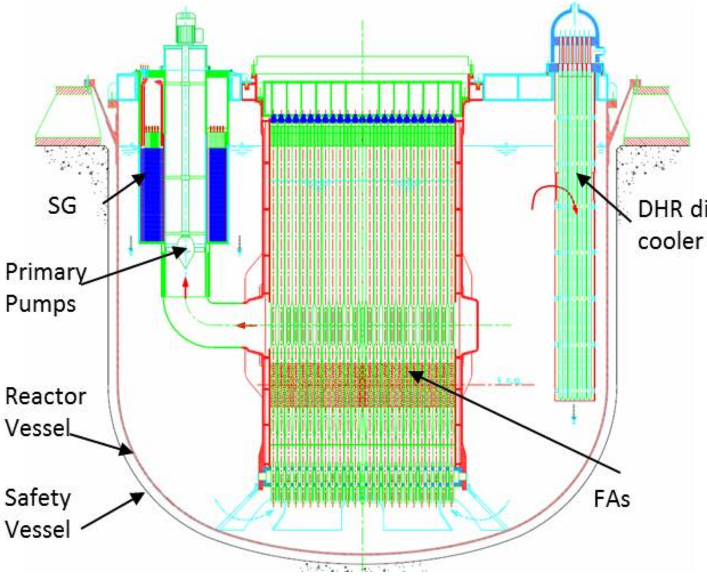
All of these three reference designs are cooled by pure lead, although lead-bismuth eutectic remains a coolant option. Figures 7 to 9 schematically present the three reference options, and Table 3 summarises the main characteristics of the systems.

The present LFR reference designs feature simple primary circuits with an objective of maximising removability of internal components to promote competitive electric power generation and long-term investment protection.

Table 3: **Key design parameters of the GIF LFR reference systems**

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

Figure 7: **ELFR**



Notes: SG = steam generator; DHR = decay heat removal; FA = fuel assembly.

Figure 8: **BREST-OD-300**

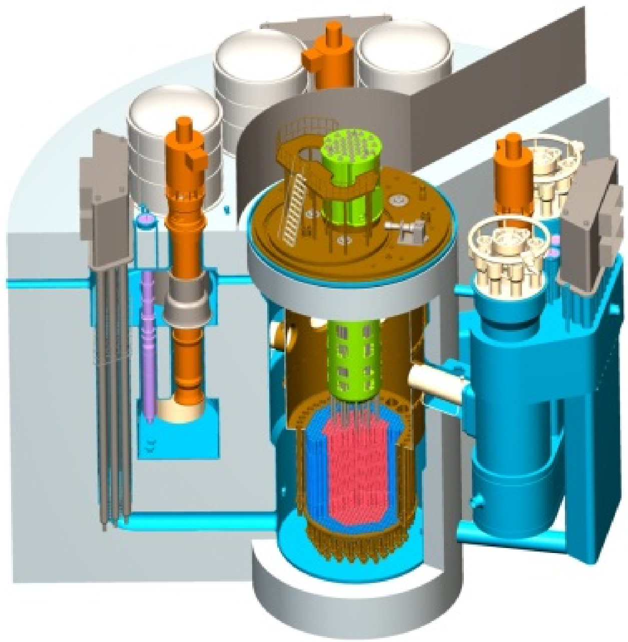
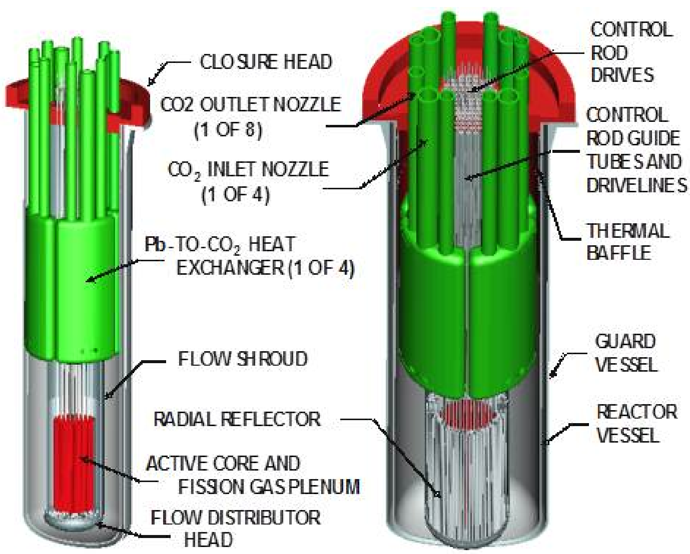


Figure 9: **SSTAR**



Performance

Two of the reactor concepts rely on secondary water loops with superheated once-through steam generators and a steam Rankine cycle able to reach a net efficiency well above 40%. The third system (SSTAR) achieves a similar level of efficiency by relying on a supercritical CO_2 power conversion system in a direct Brayton cycle. The simplicity of these LFR system configurations is expected to reduce both the capital cost and construction time, contributing in this way to their economic attractiveness. Simplicity also enhances compactness, for example through the elimination of an intermediate cooling system, as well as through the design of reduced-height internal components. The core consists of an array of open (or wrapped) fuel assemblies surrounded by reflector assemblies. The transportable 20 MW_e option employs natural circulation in the primary lead circuit, eliminating the need for coolant pumps and associated ancillary equipment.

The current reference design for the ELFR exploits the results of many years of extensive R&D programmes on heavy-metal cooled critical/subcritical systems and basic lead technologies, as performed under the 5th, 6th and 7th Framework Programmes of Euratom and in the Euratom Horizon 2020 programme for research and innovation. The conceptual reactor design represents the industrial, large size LFR to be used as the reference for the development of dedicated demonstrator initiatives. The ELFR also benefits from activities dedicated to the Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA) project, developed in Europe as a demonstrator for an accelerator driven system and flexible irradiation facility, cooled by lead-bismuth eutectic.

The reference design for the SSTAR is a 20 MW_e natural circulation reactor in a transportable reactor vessel. The design features molten lead coolant, a nitride fuel containing transuranic elements, a fast spectrum core and a small size. These combine to provide a unique approach to proliferation resistance by enabling a long core life, autonomous load following, simplicity of operation, reliability and transportability, and a high degree of passive safety.

The BREST-OD-300 reactor is a pilot demonstration reactor of 700 MW_{th} with an electrical power of 300 MW_e using lead coolant and nitride fuel, featuring a closed fuel cycle with related facilities (nitride fuel manufacturing and reprocessing) co-located on the reactor site. BREST-OD-300 is considered as a prototype of future commercial reactors of the BREST family for large-scale nuclear energy deployment with a very high level of safety.

The design uses the properties of the heavy lead coolant and high-density, heat-conducting uranium-plutonium nitride fuel to create an environment for efficient fuel utilisation in the core. Along with a small fuel temperature power effect, this enables power operation with a small reactivity margin, which constitutes an important prevention measure against unprotected transient overpower (UTOP) accident initiators. To prevent loss-of-coolant (LOC)-type of accidents, the BREST design uses an integrated layout with a multilayer reinforced metal-concrete vessel. The lead coolant and the main primary circuit equipment are located in the vessel. There are no coolant valves in the primary circuit. BREST implements a system of emergency cooling with convective circulation, removing heat directly from the primary circuit to the ultimate heat sink (air).

The detailed design of BREST-OD-300 was submitted to the safety authority in 2016, and construction licensing review is under way. In addition, a dedicated fuel manufacturing plant for nitride fuel is being built on the BREST site.

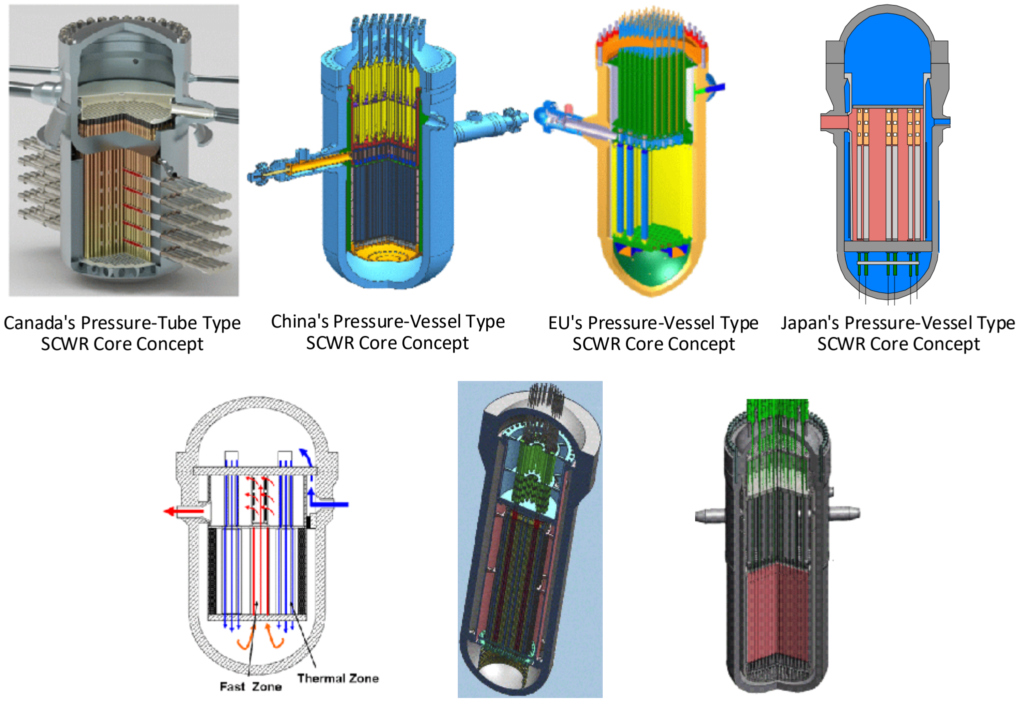
Supercritical water-cooled reactor (SCWR)

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).

Reference concepts

Two types of SCWR core configuration are being pursued: pressure vessel and pressure tube. These core designs are based on thermal neutron, fast neutron or mixed (thermal and fast neutron) spectra. Figure 10 illustrates the SCWR core configurations. Uranium-based fuel has been adopted for the pressure-vessel type, with thorium-based fuel for the pressure-tube-type thermal-spectrum SCWR concepts. Mixed oxide-based fuel is selected for the fast-spectrum SCWR concepts. Figure 11 illustrates SCWR fuel assembly configurations. The majority of plant concepts are being developed for power generation higher than 1 000 MW_e at operating pressures of about 25 MPa and reactor outlet temperatures up to 625°C.

Figure 10: **SCWR core configurations**



Canada's Pressure-Tube Type SCWR Core Concept

China's Pressure-Vessel Type SCWR Core Concept

EU's Pressure-Vessel Type SCWR Core Concept

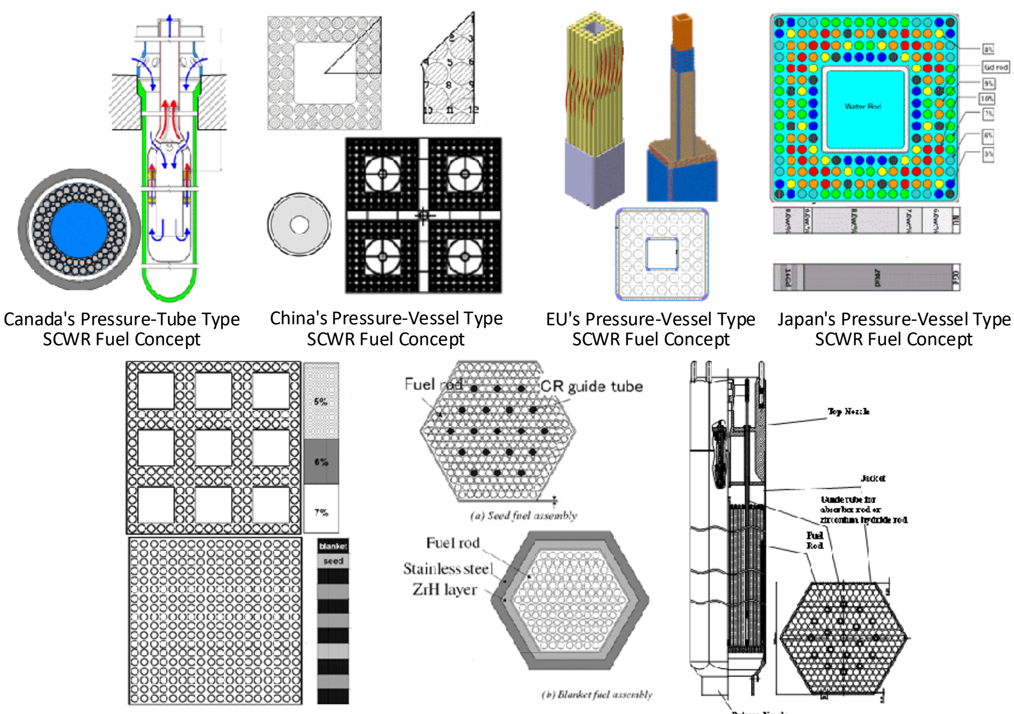
Japan's Pressure-Vessel Type SCWR Core Concept

China's Mixed-Spectrum SCWR Core Concept

Japan's Fast-Spectrum SCWR Core Concept

Russian Federation's Fast Spectrum SCWR Core Concept

Figure 11: **SCWR fuel assembly configurations**



Canada's Pressure-Tube Type SCWR Fuel Concept

China's Pressure-Vessel Type SCWR Fuel Concept

EU's Pressure-Vessel Type SCWR Fuel Concept

Japan's Pressure-Vessel Type SCWR Fuel Concept

China's Mixed-Spectrum SCWR Fuel Concept

Japan's Fast-Spectrum SCWR Fuel Concept

Russian Federation's Fast Spectrum SCWR Fuel Concept

The SCWR balance-of-plant is considerably simplified because the coolant does not change phase (boil) in the reactor, which eliminates the need for the moisture separator reheaters and recirculation pumps found in boiling water reactors or the steam generators found in pressurised light water and heavy water reactors. Its configuration is the same as that of a fossil-fired power plant, for which there is over 50 years of design and operation experience. The safety systems of the SCWR are similar to those of advanced boiling water reactors. Additional passive safety features have been incorporated to enhance the SCWR safety characteristics.

Performance

The main mission of the SCWR is to generate electricity efficiently, economically and safely. All developed concepts could generate electricity with thermal efficiencies ranging from 43% to 48%, which is better than the 35% achieved by current nuclear reactor fleets. Improvement in thermal efficiency reduces the amount of fuel usage for power generation, improving resource utilisation and reducing waste generation from turbines and condensers to minimise environmental impact (enhanced sustainability).

From an economic perspective, simplification of the balance-of-plant systems reduces the capital cost of the plant. Preliminary assessments show a reduction in total capital investment cost of up to 20% for SCWRs compared to the advanced boiling water reactor [6]. The number of plants needed to generate the required amount of electricity is also reduced in view of improvements in thermal efficiency compared to the Gen II or Gen III nuclear reactors (enhanced sustainability and economics).

The use of thorium-based fuel in the thermal-spectrum SCWR and mixed oxide fuel in the fast spectrum SCWR would close the fuel cycle, while both the pressure-tube type thermal-spectrum and the fast-spectrum SCWR concepts have the capability to transmute actinides for effective waste management, resulting in greater environmental sustainability.

Passive safety systems (e.g. the containment cooling system in the pressure-vessel type SCWR and the passive moderator cooling system in the pressure-tube type SCWR) have been introduced to enhance the safety characteristics of the SCWR. These systems maintain the fuel and cladding integrities during normal operation and postulated accident scenarios. Deterministic safety analysis has demonstrated that temperatures for both the cladding and the fuel centreline are well below the melting points during normal operation and postulated accident scenarios. Core damage frequencies are similar to those of light water reactors for the pressure-vessel type SCWR and are reduced for postulated accident scenarios in the pressure-tube type SCWR (by at least an order of magnitude).

Safeguard features of thermal-spectrum SCWRs are similar to those of the light water reactors (i.e. batch-fuelled, solid fuel, light water coolant, etc.). The introduction of the advanced thorium fuel cycle to the thermal-spectrum SCWR, and the resulting production of the fissile ^{233}U, need to be addressed from the proliferation resistance point of view. The difficulty in separating ^{233}U and ^{238}U, as well as high gamma-emitting ^{232}U, act as deterrents for use in nuclear weapons ^{233}U. Other mitigation techniques (such as addition of natural or depleted uranium to the fuel) could minimise the attractiveness of ^{233}U extraction from used fuel.

SCWRs are developed to generate high-temperature steam, ranging from 500°C to 625°C at the outlet of the core, which facilitates the use of a direct cycle as in the boiling water reactors, but without the need for the moisture separator, nor recirculation pumps or steam generators. The simplification of system configurations in the SCWR reduces considerably the size of the containment and the reactor building, as well as the footprint of the plant, compared to the current fleet of reactors. In addition to the capital cost saving, the reduction in the plant footprint provides the opportunity for enhancing physical protection.

The high outlet temperature of the SCWR core is applicable for hydrogen production (via the high-temperature steam electrolysis process or the copper-chlorine cycle), industrial process heat for chemical processing facilities such as petroleum refineries, and steam generation for oil production (via the steam-assisted gravity drainage process). Waste heat from the plant is ideal for potable water production using the desalination process.

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Chapter 3. Major achievements and current outlook for Generation IV systems

3.1. Sodium-cooled fast reactor (SFR)

The long-term vision for the SFR is for this reactor to be highly sustainable, realising the resource extension and waste reduction benefits possible with actinide management. In addition, technological advances to achieve cost reduction and safety refinement are also being pursued. The SFR benefits from the worldwide operational experience of sodium-cooled reactors, with several national programmes aimed at near-term development and construction of prototype Generation IV (Gen IV) reactors. Based on existing, broad experience, R&D goals were identified in the 2014 update of the Gen IV International Forum (GIF) *Technology Roadmap* [1]. For SFR systems, remaining R&D challenges exist, as outlined in the complementary areas listed below.

Safety systems and analysis tools:

- improvement in core inherent safety and instrumentation and control;
- passive shutdown systems;
- decay heat removal (DHR) (e.g. natural circulation, system diversification, ultimate heat sinks);
- prevention and mitigation of severe accidents with large energy releases.

Component design and operation (for improved economics):

- system configurations (e.g. fuel handling);
- energy conversion systems without steam/water;
- steam generators with high reliability;
- prevention and mitigation of sodium fires;
- in-service inspection and repair (e.g. under sodium viewer).

Actinide management:

- development and demonstration of recycle fuels;
- high-performance core/fuel.

Although SFR technology was identified as the most mature of the GIF concepts, the capital costs of previous experimental SFRs were high. Recent cost studies estimate that the capital costs of current designs may be 25% greater than conventional light water reactors (LWRs). Given the importance of achieving SFR economic competitiveness to enable commercial deployment, the GIF SFR design options incorporate significant technological innovations to reduce capital costs through a combination of configuration simplifications, advanced fuels and materials, and refined safety systems. In parallel, a major goal is to apply innovation to further enhance safety and proliferation resistance. Gen IV collaborative R&D on SFRs is focused on several design tracks, as contributed by members. These concepts use a variety of system configurations, fuel types and power levels [2].

To promote a common understanding of the safety approach and licensing framework, the GIF SFR collaboration contributed extensively to the development of safety design criteria (SDC) co-ordinated by the GIF SDC Task Force; more specifically, to the technical concept input requested for the SFR SDC Phase 1 report; and to the revisions based on feedback received from the International Atomic Energy Agency (IAEA), the Nuclear Regulatory Commission (NRC, United States), the Institute for Radiological Protection and Nuclear Safety (IRSN, France), and the National Nuclear Safety Administration (NNSA, China). Related efforts to develop SFR safety design guidelines (SDG) continue (see Section 4.2 of this report).

The major achievements in meeting the aforementioned challenges can be summarised as follows.

Safety systems and analysis tools

After the Fukushima Daiichi NPP accident, R&D on the development of simulation tools, as well as studies and experiments related to safety, have focused on the four main related topics listed below. These studies have furthered the understanding of the inherent favourable behaviour of SFR systems as regards safety (e.g. natural circulation for DHR) and strengthened severe accident prevention and mitigation features. Complementary safety-related activities also addressed innovative shutdown systems, external hazards, and core design optimisation for enhanced safety [3–10].

Decay heat removal (DHR) and natural circulation

- ex-vessel DHR systems modelled for performance evaluation;
- studies of passive safety tests using the Joyo experimental fast reactor;
- definition of analytical conditions for plant dynamics evaluation in Monju turbine trip test;
- sodium experiment performed on fully natural circulation systems for DHR in the Japan Sodium Fast Reactor (JSFR);
- development of natural circulation analytical model in Super-COPD code and evaluation of core cooling capability in Monju during station blackout;
- study on reactor vessel coolability of SFRs under severe accident conditions (water experiments using a scale model).

Probabilistic safety assessment (PSA)

- development of severe accident evaluation technology (level-2 PSA) for SFRs and identification of the dominant factors in the initiation and transition phases of unprotected events;
- development of reliability evaluation methodology on natural circulation heat removal in Level-1 PSA for JSFR.

Severe accidents, sodium boiling, fuel degradation and core catcher

- development of subassembly total instantaneous blockage (TIB) deterministic and probabilistic models;
- development of simplified models for severe accident calculations;
- modelling of sodium boiling and related experiments;
- analyses by SIMMER-III of the integral verification of the COMPASS Code on fuel-pin disruption and low-energy disrupted core motion;
- elimination of severe re-criticality events in the core disruptive accident of JSFR, aiming for in-vessel retention of the core materials;

- SIMMER-III analysis of EAGLE-1 in-pile tests focusing on heat transfer from molten core material to steel wall structure;
- study of the mechanism of upward fuel discharge during core disruptive accidents in SFRs;
- establishment of in-vessel retention of core disruptive accident in the context of the safety strategy for JSFR;
- development of assessment methods for a self-levelling behaviour of debris bed and analysis of experiments;
- investigation of clad disruption in fuel element simulators under static sodium at transient heat load conditions;
- sodium boiling analysis in a seven-rod bundle SFR fuel subassembly mock-up;
- experimental investigation of the degradation of fuel pin claddings on model assemblies.

Containment

- transport of radioactive corrosion products in primary system of the Monju sodium-cooled fast neutron reactor;
- development of evaluation methods for the transfer behaviour of corrosion products in the primary cooling system;
- evaluation of hydrogen transport behaviour in the Monju power rising test.

Component design and operation achievements

A variety of innovative technology options are being evaluated to reduce the capital and operations cost of SFR reactors [11–14]. Key achievements in five main areas are noted below:

Advanced cycles for energy conversion

- Several aspects of supercritical CO_2 (sCO_2) cycles have been investigated. The use of dry air cooling has been evaluated positively for the Brayton sCO_2 cycle. The modelling of the sCO_2 cycle, and in particular the Plant Dynamics Code, has been validated with comparison of results obtained on small loops.
- Progress on the understanding of the water–sCO_2 interaction: new experiments were performed with innovative instrumentation (such as acoustic detection) to allow better understanding of reactions depending on the initial conditions (temperature).
- Development of sodium-nitrogen heat exchanger: compact plate-type heat exchangers are proposed with innovative channel design. Models were qualified on small mock-up experiments (40 kilowatts [kW] in representative high-pressure gas and high-temperature sodium). Header optimisation has been studied both by simulation and experiments.

Sodium modelling

- In order to lower sodium leakage occurrence, the leak-before-break (LBB) strategy is being consolidated. The objective is to establish a rational LBB assessment method. The fracture mechanics parameter assessment procedures employed in the Japan Society of Mechanical Engineers LBB standard are proposed.
- Kinetic study of sodium-water reactions: a reaction model (gas phase reaction and surface reaction) with *ab initio* calculation is being used for predicting the reaction pathway and determining the direction of R&D on reaction kinetics, with thermal analysis to obtain the kinetic parameters.

Steam generator reliability monitoring

- Proposal for a new technique to detect sodium-water reactions, based on passive acoustic detection with innovative signal treatment techniques and application to different recorded signals (Phénix data).
- Investigation of the potential uses of eddy current flowmeter for gas leak detection: a first theoretical approach and modelling are proposed, along with simple experiments to confirm the approach.
- For steam generator tube inspection (wall thinning, pitting and crack detection), development of a new kind of sensor, combining remote field eddy current testing (RFECT) and magnetic flux leakage (MFL). RFECT testing and performance demonstration have been conducted on tubes with different kinds of grooves.
- To better evaluate the behaviour of cracks, fatigue crack growth and creep crack growth, carrying out of tests on P91 specimens. Mathematical models of fatigue and creep crack growth were obtained from the tests. Conservatisms of the crack growth models (RCC-MRx) were quantified.
- Conducting of self-wastage experiments using specimens of pinhole and fatigue crack, and material assay: the applicability of the existing self-wastage correlation equation for 9-Cr steel was confirmed.
- Development of the mechanistic sodium-water reaction analysis model: the SERAPHIM code predicts the temperature distribution and the environment of the liquid droplet impingement erosion under practical steam generator conditions.

In-service inspection and repair (ISIR)

- Optimisation of the ultrasonic waveguide sensor for ranging: manufacturing optimisation, theoretical formulation of the waveguide radiation problem, implementation in CIVA code (acoustic wave transmission) and validation. First in-water tests performed and promising results obtained by detecting obstacles between core and upper core structures.
- Under-sodium viewing detection development for non-destructive testing, telemetry and imaging. Major accomplishments include a specific robotic arm developed and operated in hot liquid sodium. Tests demonstrated the capability of a single-element, ultrasonic transducer to produce under-sodium images and to detect small grooves in a metallic plate placed in a near-field area.
- For long-distance imaging, initial experiments started using water as a simulant of sodium and using available multi-array sensors. In parallel, the development of a multi-array sensor able to operate in hot liquid sodium is ongoing.
- To detect localised flaws in large welded areas, non-destructive examination using a topological energy method has been developed, with experimental validations using acoustic transducers.

Information exchange experimental platforms

- progress of the Sodium Thermal Hydraulic Test Programme (STELLA) at the Korean Atomic Energy Research Institute (KAERI): design review of the large-scale sodium thermal hydraulic test facility;
- Large-Scale Sodium Facility (METL) at the Argonne National Laboratory (ANL, United States);
- versatile sodium pot and sodium loop hybrid concept (MECANA) at the Alternative Energy and Atomic Energy Commission (CEA, France).

Actinide management achievements

Metal and oxide fuels have been identified as short-term candidates for transmutation fuels, whereas carbide and nitride fuels remain longer-term options [15].

The Global Actinide Cycle International Demonstration (GACID) project has concluded. It was aimed at demonstrating that the SFR can effectively manage all actinide elements, including uranium, plutonium and minor actinides (MAs). The project included fabrication and licensing of MA-bearing fuel, pin-scale irradiations, material property data preparation, irradiation behaviour modelling and post-irradiation examination (PIE), as well as transport of MA raw materials and MA-bearing fuels. Bundle-scale demonstration was also included.

The irradiation behaviour (e.g. americium migration) of the MA-bearing fuel irradiated in the Joyo reactor (AM1 irradiation) was analysed and investigated in detail based on the PIE results.

(U,Pu,Am,Np)OX fabrication was demonstrated at the CEA. Irradiation test preparation and implementation, PIE and calculations of fuel behaviour under irradiation were performed for oxide and metallic fuels. In particular, the Idaho National Laboratory (INL) completed non-destructive examination of americium-bearing oxide fuel irradiated up to very high burnups in its advanced test reactor, within the framework of the AFC-2 campaign. In parallel, PIE of AFC-2D (U0.75, Pu0.2, Am0.03, Np0.02)O1.95 fuel yielded visual examination, neutron radiography, gamma spectroscopy and dimensional measurement results. To evaluate the pellet structural change of MA-bearing mixed-oxide (MOX) fuel, the pore migration model was improved by introducing the O/M dependence in MA-MOX vapour pressure calculations. It was shown that the central void formation observed in Am-bearing MOX irradiated in Joyo was simulated well by this improvement.

The GACID Project was terminated after the decision to decommission the Monju reactor. Therefore, design, fabrication and qualification at representative scale of a fuel element containing MA diluted in the oxide driver fuel could not be achieved.

Another key topic for effective actinide management, pursued within the framework of the Advanced Fuels project, was the development of innovative fuels to reach high burnup, and of claddings and wrappers capable of withstanding high neutron doses and temperatures. The Advanced Fuels project also included research on remote fuel fabrication and material manufacturing techniques, as well as on performance under irradiation [16, 17].

Preparation work is ongoing for an irradiation test up to a medium burnup of U-Zr-type fuels in the High Flux Advanced Neutron Application Reactor (HANARO). The effect of the oxygen potential during sintering on $(U,Pu)O_2$ microstructure has been investigated, as have the corrosion resistance of americium-bearing oxide fuels in liquid sodium and new fuel fabrication routes. As regards cladding development, fabrication and characterisation of ferritic/martensitic cladding tubes is ongoing [18-20].

Looking forward to the next decade, the major challenges for safety, component design and operation, as well as actinide management guiding the SFR R&D collaboration within the GIF framework, are summarised below:

Several safety-related topics were identified for common projects, as listed below.

- Natural circulation in sodium systems (first priority)
  - design issues: thermal stratification, flow redistribution or reversal, freezing, thermal stress;
  - evaluation methods: phenomena identification and ranking table, model selection, plant-scale validation, uncertainty quantification;
  - fundamental models: heat capacity, pressure loss and property correlations;
  - experimental measurement techniques.
- Reactivity control system options (e.g. hydraulic, fusible devices, Curie-point, GEMs [gas expansion modules], ARC [autonomous reactivity control])

- Ex-vessel cooling system options
  - reactor vessel air cooling system (e.g. air natural circulation, forced oil convection, modelling approaches).
- Sodium boiling (e.g. timing and location, stability, codes and methods, experiments)

With regard to component design and operations, some challenges remain in the following areas:

- Advanced energy conversion systems
  - testing of small-scale compact diffusion-bonded heat exchangers;
  - testing of small-scale sCO_2 compressors;
  - sodium-CO_2 interactions;
  - CO_2 corrosion of various austenitic and ferritic steels;
  - analysis of SFRs incorporating sCO_2 Brayton cycles;
  - plant dynamic analysis for SFRs with sCO_2 cycles;
  - investigation of plugging in sodium channels of sodium-to-CO_2 heat exchanger.
- Steam generators
  - steam generator development, including investigation of sodium-water reactions and development of advanced inspection technologies for a Rankine-type steam generator.
- Sodium leakage and interactions
  - development of LBB assessment procedures;
  - instrumentation to detect sodium leakage;
  - sodium fire and aerosol consequence analysis and mitigation approaches;
  - creep-fatigue crack initiation and growth behaviour in ferritic martensitic steels;
  - sodium leakage detection, in particular improvement of time response and limitation of false alarms;
  - improved knowledge of sodium fire in order to limit its consequences and design appropriate mitigation systems.
- ISIR technology
  - development of various innovative inspection technologies, e.g. transducer development and testing, waveguide development and testing, sodium wetting behaviour, eddy current flowmeter for fuel assembly outlet flow measurement, CIVA code applications, under-sodium viewing techniques.
- Sodium operation technology and new sodium testing facilities
  - sharing and/or development of information/experience in the fields of sodium drainability, wetting, plugging and purification, freezing and remelting;
  - exchange of information related to existing and/or new sodium testing facilities (e.g. AtheNa, STELLA-2, and METL).

In the area of actinide management, innovative fuel development remains an ongoing challenge. The SFR collaboration will focus on the following three items:

- Non-MA-bearing driver fuel evaluation, optimisation and demonstration. Based on the results of the first project phase, the identified gaps of knowledge on fuel performance and fabrication technologies will be further investigated. Experimental and analytical evaluation efforts will be required, including fabrication and irradiation tests. The goal is the demonstration of fabrication and adequate irradiation performance at assembly level for non-MA-bearing nitride, oxide and metal driver fuels.

- MA-bearing transmutation fuel evaluation, optimisation and demonstration. MA-bearing fuels as driver fuels and targets dedicated to transmutation will be further investigated, addressing both the homogeneous and heterogeneous modes of MA transmutation. The objective is to achieve effective utilisation and destruction of MAs. An additional objective is the optimisation and demonstration of fabrication and irradiation performance on the fuel pin/bundle level for MA-bearing driver fuel. For the MA-bearing transmutation targets, the demonstration at the fuel pin level is a long-term goal.
- High-burnup fuel evaluation, optimisation and demonstration. High burnup fuel performance potential and safety performance characterisation will continue. The scope is extended to irradiation testing and further development of core materials (cladding, wrapper) for high burnup fuels. Optimisation and demonstration of high burnup fuel performance is targeted at the fuel pin level.

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3.2. Very high temperature reactor (VHTR)

Major R&D progress has been achieved on the VHTR, in response to the R&D challenges identified in the 2014 GIF *Technology Roadmap Update* [1]. References documenting highlights are given only indicatively and are not exhaustive.

In addition to the restricted information exchanged on the GIF platform, much of the progress made internationally on the VHTR system is reported every two years in the HTR conference series [2], which began in 2002. Moreover, a status report was published in 2014 [3].

Fuel/fuel cycle

The VHTR fuel cycle is initially a once-through fuel cycle specified for high burnup (150-200 gigawatt days per tonne of heavy metal [GWd/tHM]) using low-enriched uranium. The used fuel, fuel cycle, enrichment and target burnup are the result of an overall cost optimisation that may differ from one country to another. Solutions to improve the fuel cycle back end have been developed, while the possible use of thorium for a closed thorium-uranium fuel cycle is considered a longer-term option.

Despite the alternative (pebble or prismatic) fuel designs, the two baselines have many technologies in common that allow for a unified R&D approach. The well-known UO_2 tristructural isotropic (TRISO)-coated particle fuel (with a UO_2 kernel and SiC/PyC coatings) is the origin of the VHTR's benign safety performance and may be used in either fuel design or further enhanced with a UCO fuel kernel (as shown by the US programme) and/or advanced coatings through additional research.

The primary emphasis in fuel development has been on its performance at high burnup, power density and temperature. R&D has broadly addressed its manufacture and characterisation, quality assurance methods, irradiation performance and accident behaviour. Irradiation tests provided data on coated particle fuel and fuel element performance under irradiation that is necessary to support fabrication process development, to qualify the fuel design, and to support development and validation of models and computer codes on fission product transport relevant for safety assessment and licensing [4–8]. These experiments also provided irradiated fuel and material samples for post-irradiation and safety testing, which is undertaken to confirm that the fuel remains leak tight against fission product escape in all normal, transient or accident conditions (beyond design basis) where small fractions of the core can reach temperatures of the order of 1 600°C.

A strategy for waste minimisation and management has been established that considers, *inter alia*, sustainability criteria, economics and proliferation issues. Different approaches for used fuel management are possible and have been investigated:

- direct disposal of fuel elements;
- separation of the fuel element from the graphite moderator;
- separation of coated particles from the matrix graphite and treatment of both fractions;
- separation of kernels from coatings and recycling of fuel.

Materials and components

Several components are needed for the VHTR system, some of which may be similar to those of a gas-cooled fast reactor. These components include the reactor pressure vessel, circuit components such as valves or circulators, piping, thermal insulation, seals to minimise expensive helium leakage, helium handling and purification systems, instrumentation, intermediate heat exchangers and Brayton-cycle turbo machinery. The pressure vessels are unique because of their dimensions: depending on the reactor size, they can range from road-transportable to larger than modern boiling water reactor vessels. Their development has included welding and fabrication methods, as well as means to ensure high thermal emissivity of the outer vessel walls.

The intermediate heat exchanger must be a highly reliable, compact and thermally efficient boundary between the primary and the secondary coolants. Printed-circuit heat exchangers or plate-fin type compact heat exchangers are favoured because of their size and high efficiency. However, high-temperature materials, manufacturing processes and the components themselves need to be qualified, together with the development of suitable design codes and standards for the required temperatures, thermal and/or pressure cycling and the required lifetime in a potentially corrosive environment.

For core outlet temperatures up to about 950°C, existing materials can be used; however, temperatures above this, including safe operation during off-normal conditions, require the development and qualification of new materials. The research focused on: a) graphite for the reactor core and internals [9-11]; b) high-temperature metallic materials for internals, piping, valves, high-temperature heat exchangers [12, 13], and gas turbine components; and c) ceramics and composites for control rod cladding and other core internals, as well as for high-temperature heat exchangers and gas turbine components.

Reactor systems and balance of plant

The VHTR has two established baselines for the core: pebble bed or prismatic block. Each system has its specific advantages. Several signatories of the VHTR System Arrangement pursue their own design projects and much of the information is shared within GIF; however, no common design effort is currently under way within the SSC. For power conversion, several nearer-term (and lower-temperature) VHTR projects use steam cycle technology [14, 15], since much of the process heat market can be captured with steam at these temperatures and because of the well-established commercial infrastructure supporting steam-based power conversion. Direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles are also being considered for deployment in several countries in the near future [16, 17], especially with higher temperature (~950°C) VHTR concepts.

Regardless of the choice of power conversion, supply of process heat will require an intermediate heat exchanger connected to the primary circuit [18]. Near-term concepts for these components are being developed using existing materials, and more advanced concepts are stimulating the development of new materials. An intermediate heat exchanger is also required for thermochemical hydrogen production and may use a heat transfer fluid such as helium, a gas mixture, or molten salt.

As a specific process heat application, the SSC organises the collaborative development of hydrogen production processes [19, 20]. Several facilities combining the three elementary processes of the iodine-sulphur process to a thermochemical cycle have been built in Japan and China, and operated at continuous, stable operation at 20-60 litres per hour for almost four days.

Progress has been made in the area of materials, components, optimisation of elementary processes, process control, safety, system integration and cost evaluation. The main alternatives to the iodine-sulphur cycle are high-temperature steam electrolysis (HTSE) or hybrid cycles using various fractions of high-temperature process heat and electricity. A lab-scale HTSE experiment was successfully executed at the Idaho National Laboratory (INL) in the United States, and now a pilot-scale HTSE plant is being constructed. These two alternatives deserve further development in areas such as viability of the basic processes, materials for electrolysis cells or reaction vessels, and scale-up and control of large processes.

Beyond process equipment, research is ongoing into the coupling of a nuclear reactor with industrial processes [21]. This involves the analysis of safe and reliable control and operation, including the combined effects of each of the systems. It also branches out into the conceptual design, licensing and economics of systems for various petrochemical and other applications.

In the absence of a separate GIF VHTR project on system integration and assessment, several of the signatories investigated the integration of VHTR systems into electricity grids and heat networks with fossil and variable renewable fractions [22–24].

In support of design and licensing, a project on computational methods, validation and benchmarking was prepared for signature in 2018. The project signatories have already exchanged significant information while the project remained provisional [25, 26]. The project encompasses neutronic and thermal fluid model validation using a number of experiments provided by the signatories. In 2017, the project was presented with guidelines for validating system and computational fluid dynamics models that have been accepted by the US Nuclear Regulatory Commission. A common set of validation guidelines will be adopted and applied within the GIF research.

Economics

Detailed economic studies have been performed for both electricity production and process heat applications by several GIF signatories, and these have been shared informally. The US results suggest that modular VHTRs are competitive with new LWRs for electricity production. For process heat and cogeneration applications, the VHTR can be competitive with conventional combined-cycle gas turbine systems producing steam and electricity when the cost of natural gas is higher than USD 8 per million British thermal units. Carbon taxes may reduce this threshold. Currently, the cost of natural gas varies widely across the globe. Thus economic viability depends largely on the financial and regulatory climate in individual countries.

The inherent safety features of the VHTR may benefit the economics index indirectly. Similar to certain other small and medium-sized reactor concepts, the VHTR can also take credit for lower infrastructure requirements such as easier integration in smaller electricity grids, lower cooling requirements and proximity to industrial sites and agglomerations. Market research has also been performed by several of the GIF signatories [27], which was shared informally and corroborates the motivation to pursue this system.

The SSC has worked together with several companies to provide technical and economic performance information on three different VHTR designs thus far for analysis by the Senior Industry Advisory Panel.

Safety objectives

A VHTR can be designed with a high degree of inherent safety, allowing passive coolability and core/fuel integrity under all circumstances. The SSC has interacted closely with the Risk and Safety Working Group to provide a white paper and safety self-assessment document. Further efforts to generate a set of safety requirements [28] and safety design criteria for VHTRs are currently under way under the auspices of the IAEA (co-ordinated research project [CRP] to be finalised in 2018) and will be completed by the SSC. The SSC has also reported to the NEA Working Group on the Safety of Advanced Reactors (WGSAR) to consult regulators and transmission system operators for input.

As regards the path forward, R&D objectives and needs to guide the SSC's future R&D efforts have been identified. The SSC is currently carrying out four R&D projects (materials, fuel and fuel cycle; hydrogen production; computational methods; validation and benchmarking). The path forward in these areas is detailed and updated annually in the individual project plans. In the coming years, the SSC will direct its work towards further improved technology and market readiness and the demonstration of the technology in the following areas:

- completion of fuel testing and qualification capability (including fabrication, quality assurance, irradiation, safety testing and PIE), to be completed in certain countries. Waste reduction and fuel recycling;
- qualification of graphite, hardening of graphite against air/water ingress, e.g. by SiC infiltration; management of graphite waste;
- coupling technology and related components (e.g. isolation valves, intermediate heat exchangers);
- establishment of design codes and standards for new materials and components;
- advanced manufacturing methods (co-operation with the GIF Cross-cutting Task Force).
- cost-cutting R&D and interaction with the GIF Economics Modelling Working Group (EMWG) and industry to optimise the VHTR design;
- licensing and siting: verification and validation of computer codes for design and licensing;
- system integration with other energy carriers in hybrid energy systems;
- follow-up on the High Temperature Reactor – Pebble Bed Module (HTR-PM) demonstration tests, enhancing information exchange with several start-ups, private investors and new national programmes;
- High-Temperature Engineering Test Reactor (HTTR): safety demonstration tests and coupling to the H_2 production plant (subject to regulator approval for restart).

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3.3. Gas-cooled fast reactor (GFR)

The following summarises major R&D achievements on the GFR in response to the R&D challenges identified in the 2014 GIF *Technology Roadmap Update* [1].

Fuel and materials

Exploratory design studies on the commercial GFR to identify a promising set of design options – in particular fulfilling the objective of self-generating cores, which has a very strong impact on fuel design – indicated the necessity for high, heavy metal content in the core. Studies show that dense fuels like carbides or nitrides can assist in the quest for high actinide content in the core. In addition, their good thermal conductivity lowers the fuel temperatures to the required level. Detailed comparisons between carbide and nitride fuels concluded that both remain serious candidates and that only ongoing R&D can provide new insights. Nevertheless, core options in which the oxide could be used as the actinide compound have not been neglected and are considered to be in a transition period before the full development of carbide fuels. The pin type solid solution fuel has been selected. For the cladding, SiCf/SiC is the ambitious choice, while metallic refractory or semi-refractory claddings could be envisaged as the backup solution.

Regarding core structural material of the GFR (cladding, fuel assembly structure, control rods, stand pipes, reflector), studies have shown that ceramic materials (CER) and composite materials (e.g. CER/CER, CERf [fibres]/CER, CER/MET) present the greatest promise for use in a high-temperature nuclear environment ranging from 500°C to about 1 200°C in normal operating conditions, and up to 1 600–1 800°C in (anticipated) accidental transients.

Neutronics

The ENIGMA experimental programme in MASURCA provided data for the validation of neutronic tools. It involved a spectrally representative reference core made of $(U,Pu)O_2$, UO_2, graphite and void rods, and a steel reflector. The refurbishment process at MASURCA delayed this programme. However, the reference core was charged in MASURCA in 2006 and was characterised just before the shutdown for refurbishment [2].

Core thermal hydraulics (air tests)

The objective of the experimental programme performed in CEA's ESTHAIR was to perform thermo-hydraulic tests in air under conditions similar to those in the ALLEGRO initial core and in subsequent GFR-type refractory cores. It included pressure drop and heat exchange measurements, as well as hot spot risk estimations for the various types of subassembly designs that can be considered (i.e. wire spaced pins for the MOX core, plate bundles for the refractory or Vanadium plate core, and pin bundles with and without grid spacers for the refractory pin core).

The test programme on the wire spaced pins was completed and allowed comparisons to open literature correlations [3].

Safety

In the safety area, efforts were mainly devoted to the analysis of the feasibility of the ALLEGRO demonstrator. Design improvements were tested in relation to the power and power density of the stainless steel cladded oxide start-up core, as well as on the safety systems (accumulators for safety injection and increased backup pressure for the guard containment) [4, 5].

Benchmarking exercises on the thermal-hydraulic system codes (CATHARE, RELAP, MELCOR) used for the safety analysis completed the activity.

Technology/components

At the pre-design stage of circulator modelling, a mean line analysis model was developed and studied. Although this type of flow analysis is a strong simplification of a complex three-dimensional system, it was able to describe the behaviour of a compressor with sufficient accuracy at the pre-design stage.

The validation of the model was done based on experimental results in air [6, 7], and helium machines [8, 9]

The Von Karman Institute in Belgium performed design work on a helium circulator for the previous 50 megawatt (MW) ALLEGRO concept. A two-zone model and a computational fluid dynamics (CFD) three-dimensional analysis completed this work. As far as pressure ratio is concerned, the levels given by the simulations are in agreement with the reference data. The location of the peak value is well predicted. Simulations of efficiencies are of acceptable quality.

Helium purification

A preliminary sizing of the ALLEGRO purification system was performed, summarising the experience and knowledge obtained from the operation of systems for purification of the high-temperature gas-cooled reactor coolant. A purification circuit, parallel with the primary circuit, has been designed; it contains filters, columns with a catalytic bed (CuO), adsorbers with molecular sieves, low-temperature adsorbers cooled by liquid nitrogen at -160°C and a retaining bed (delay trap) for fissure product trapping.

In addition, experimental results were obtained for the helium recovery from the guard vessel [10].

Mechanical design of structures

As regards mechanical design rules and acquisition of material data, many activities were conducted at the CEA in France on martensitic 9Cr steels, which are candidate materials for some GFR structures. Much progress has been made on the following items: characterisation of the cyclic and creep behaviour of the material (cyclic softening effect), validation of creep-fatigue analysis rules, determination of the negligible creep domain, and determination of crack propagation laws and weld joint behaviour in creep and fatigue.

The ALLEGRO project

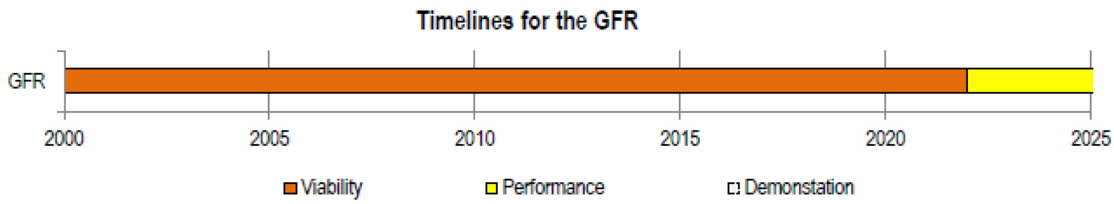
ALLEGRO is an experimental fast reactor cooled with helium that constitutes an important step on the way to the GFR [11]. It is one of the six Gen IV concepts and one of the three fast reactors supported by the European Sustainable Nuclear Energy Technology Platform.

Starting from a reference design studied at CEA prior to 2009, the project explored a new target of nominal power (in the range of 30-75 MW thermal [MW_{th}]) and power density (in the range 50-100 MW per cubic metre [MW/m^3]) compatible with the safety limits and the design requirements (Table 4). At the same time, the feasibility of a low-enriched uranium and uranium dioxide (LEU UOX) start-up core as an alternative to a standard MOX core is being considered. This start-up core, to be used in the first period of the reactor operation, will include experimental positions dedicated to the refractory fuel development.

Looking ahead to the next 10-20 years, the GFR system is expected to conclude the viability phase, enter the performance phase and most likely initiate the demonstration phase (Figure 12). The following major steps in GFR development are expected:

- finalisation of the design of a small experimental reactor, ALLEGRO;
- decision on launching the licensing process for the experimental reactor.

Figure 12: **Anticipated GFR development**



An experimental reactor is an essential step to establish confidence in the innovative GFR technology. The proposed experimental reactor, ALLEGRO, would be the first-ever GFR to be constructed. The objectives of ALLEGRO are to demonstrate the viability and to qualify specific GFR technologies such as fuel, the fuel elements and specific safety systems, in particular, the DHR function, as well as demonstrate that these features can be integrated successfully into a representative system.

Table 4: **ALLEGRO main design characteristics**

ALLEGRO MAIN DESIGN CHARACTERISTICS			
Nominal power (thermal)	75 MW	MW	Reduced power is being considered in the 30-75 MW range
Nominal power (electrical)	0	MW	
Power density	100 MW/m^3	MW/m^3	Reduced power density is being considered in the 50-75 MW/m^3 range
Fuel	MOX/stainless steel cladding		Start-up core; feasibility of LEU UOX for the start-up core is being investigated
	UPuC/ SiCSifC cladding		Long-term core
Type of fuel assembly	Hexagonal wrapper and wired fuel rods		
Number of fuel rods per assembly	169		
Number of fuel assemblies	81		
Number of experimental fuel assemblies	6		
Number of control + shutdown rods	10		
Primary circuit coolant	Helium		
Secondary circuit coolant	Water		Gas is being investigated
Tertiary circuit coolant	Air		Atmosphere
Primary pressure	7.0 MPa	MPa	
Core inlet/outlet temperatures	260°C/516°C		Should be upgraded for full core refractory fuel
Number of primary loops	2		
Number of secondary loops	2		
Number of DHR loops	3		Directly connected to the primary vessel
DHR circuit coolant	Helium		
DHR intermediate circuit coolant	Water		
DHR heat sink	Water pool		
DHR exchangers nominal capacity per loop	2.4 MW	MW	
Number of accumulators	3		Filled with nitrogen

Note: MPa = megapascal.

The development of an acceptable fuel system that meets the target criteria (*viz.* 1 000°C normal operation clad temperature, no fission product release at 1 600°C clad temperature over a period of a few hours, and maintenance of the core-cooling capability up to 2 000°C clad temperature) is a key viability issue for the GFR system. It is necessary to develop an initial cladding material that meets the core specifications of length, diameter, surface roughness, apparent ductility, level of leak tightness (including the potential need for a metallic liner on the cladding), compatibility with helium coolant (plus impurities), and the anticipated irradiation conditions. Performance measures include fabrication capacities and material characterisation under normal and accidental conditions for fresh and irradiated fuel.

The GFR also requires a specific, dense fuel element that can withstand very high temperature transients, as a result ofthe lack of thermal inertia of the system. Uranium plutonium carbide (UPuC) pin fabrication methods, as well as their behaviour under irradiation, must be studied. Ceramic or refractory metal cladding should be selected, developed and qualified. Such a programme requires material property measurements, selection of different materials, their arrangement and their interaction, out-of and in-pile tests up to qualification, and demonstration tests.

The specific operating conditions of the ALLEGRO oxide fuel pins (*viz.* maximum fuel temperature below 1 000°C, linear pin power below 100 watts per centimetre [W/cm]) are not covered by fuel pin behaviour codes. The results of such codes appeared to be very sensitive to fission gas release predictions. In this context, a programme of post-irradiation examinations of selected pins of the PHENIX SFR irradiated CPed6106 standard fuel subassembly is suitable for obtaining experimental data on fission gas release under operating conditions similar to those foreseen for the ALLEGRO oxide core fuel pins. Alternatively, a specific irradiation programme in other, potentially available fast reactors (e.g. JOYO, BOR60, or MBIR) could be envisaged.

In addition to the qualification of the oxide fuel, development of the reference GFR fuel (i.e. carbide fuel with composite SiC and fibre-reinforced SiC cladding) must continue. In particular, it is necessary to develop a ceramic cladding that meets specifications, as discussed above.

In the area of neutronics, existing calculation tools and nuclear data libraries have to be validated for gas-cooled fast reactor designs. The wide range of validation studies on sodium-cooled fast reactors must be complemented by specific experiments that incorporate the unique aspects of gas-cooled designs: slightly different spectral conditions, innovative materials and various ceramic materials (UC, PuC, SiC, ZrC, Zr3Si2). In addition, some unique abnormal conditions (e.g. depressurisation, steam ingress) must be considered. It is hoped that the planned systematic validation efforts will be able to rely on data from the experimental ENIGMA programme that was proposed in the past and could be implemented in the zero-power MASURCA facility once it is restarted in the future [2].

Given the high temperature environment of the ALLEGRO ceramic core, the design margins considered for material characteristics, as well as in the applied thermal hydraulics correlations, must be as low as possible. Therefore, air followed by helium tests on subassembly mock-ups under representative temperature and pressure conditions are necessary to assess the heat transfer and pressure drop uncertainties for the specific GFR design.

Moreover, large-scale air and helium tests will be required for the licensing process of ALLEGRO to demonstrate the passive DHR function.

As regards component development and qualification, future GFR system R&D activities should focus on the following areas:

- Specific blowers and turbo machines are needed to cope with a wide range of pressure operation (from 7.0 to 0.1 MPa) and with rotating parts, while retaining their leak tightness.
- GFR-specific solutions for the thermal barriers that protect the metallic structures from the hot helium during normal and transient conditions must be developed and qualified in experimental facilities.

- Valves and check-valves are critical safety components of the GFR. Qualification tests of candidate technologies for these components are needed and must be performed using a dedicated helium loop.
- The design of passive shutdown systems will require related R&D to validate the technology, based on experimental programmes to be defined.
- The development of instrumentation performing under GFR conditions is one of the main challenges of this Gen IV system. In particular, the main safety issue concerns the helium temperature measurement at the core outlet, in order to be able to detect hotspots on fuel cladding or fuel assembly plugging.
- Various fuel handling solutions will have to be qualified under representative conditions.
- Seals will have to be developed (including static helium leak tightness qualification tests).

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3.4. Molten salt reactor (MSR)

The main MSR R&D challenges as identified in the 2014 *Roadmap Update* were:

a) Compatibility of salts with structural materials for fuel and coolant circuits, as well as for fuel-processing components. This challenge is addressed through academic lab-scale studies aimed at improving basic knowledge on high-nickel alloys and other advanced materials that are available, as well as through integrated corrosion studies in loops or demonstrator facilities aimed at testing the same materials under realistic thermal and hydraulic conditions and for long exposure times.

b) Instrumentation and control of liquid salts. This challenge requires the development of in situ measurement methods and tools to monitor the redox potential, which influences the corrosion of the structural materials in both the liquid fuel reactor and the fluoride salt-cooled high-temperature reactor (FHR) concepts. It also affects the composition of the salt in the fissile and fertile elements of the liquid fuel MSR.

c) Comprehensive understanding of the key physical and chemical properties of the salts as they impact the physical-chemical behaviour of the fuel and/or coolant salts and, notably, the coupling mechanisms between neutronics, thermal hydraulics and chemistry. This understanding is of paramount importance for the development and qualification of appropriate simulation tools to study normal and accidental MSR behaviour.

d) Availability of inactive salt testing loops. Such facilities are needed to support salt preparation and management studies; as well as for chemical control, accidental leak and freezing management; validation of thermal hydraulic models; process instrumentation and component testing (including heat exchanger, pump, valves); and gaseous and volatile fission product and particle behaviour and separation. Both forced and natural convection loops have to be considered to better understand heat and mass transfer and long-time material exposure to fuel and coolant salts.

e) Development and demonstration of on-site fuel processing concepts.

f) Design, construction and operation of a mock-up demonstrator without induced fission, capable of full-scale prototypical reactor component testing.

g) Availability of a demonstrator with induced fission for in-pile and online chemical potential control, monitoring of the evolution of the salt composition, measurement of corrosion in a neutron field, fission product removal through helium bubbling in a fuel salt environment, and testing of maintenance techniques.

When reviewing the major achievements in response to these challenges, it must be noted that, in spite of an increase in private initiatives, MSRs suffered from a lack of public funding that curtailed the volume of R&D work within the GIF framework and slowed down its pace of development. Therefore, the aforementioned challenges could be tackled only partly. Progress achieved on the seven challenges described above is summarised as follows.

a) Compatibility of salts with structural materials. The R&D focused on the compatibility of high nickel alloys (US Hastelloy-N; Russian HN80MTY, Czech MONICR, Chinese GH3535 and French EM-721) with molten Li,Be/F, Li,Be,U/F and Li,Be,Th,U/F salt mixtures as considered in the Molten Salt Fast Reactor (MSFR) and Molten Salt Actinide Recycler and Transmuter (MOSART) designs. These studies included:

   - a compatibility test between high-nickel alloys and fuel/coolant salts in a natural convection loop with measurement of the redox potential;
   - study of the effect of PuF_3 addition to the fuel salt on compatibility with high-nickel alloys;
   - tellurium corrosion study for fuel salt and high-nickel alloys in stressed and unloaded conditions with measurement of the redox potential.

   The results [1, 2] show that uniform and inter-granular corrosion rates for all alloys under consideration correlate with the measured redox potential and temperature of the fuel salt. The HN80MTY alloy has the best corrosion and mechanical properties.

   Experiments were carried out on hydrogen isotope permeation through a HN80MTY alloy. Hydrogen permeability, diffusion and solubility constants were measured as a function of temperature. Major findings are summarised in "Experimental study on critical issues of nuclear energy systems employing liquid salt fluorides" [3].

   Corrosion studies of multiple-structure materials in fluoride salts, and studies of the infiltration of fluoride salts into graphite were performed within the framework of the Thorium Molten Salt Reactors (TMSR) programme in SINAP (China).

b) Instrumentation and control of liquid salts. The voltammetric technique was used to measure the redox potential in molten Li,Be/F, Li,Be,Pu/F, Li,Be,U/F and Li,Be,Th,U/F mixtures. The results show that corrosion rates for all high-nickel alloys under consideration correlate with the measured redox potential and the temperature of the fuel salt [4].

c) Comprehensive understanding of the key physical and chemical properties of the salts. The method of isothermal saturation was used to measure the individual solubility of UF_4, PuF_3 and AmF_3 in binary LiF-BeF_2. The joint solubility of UF_4, PuF_3 and CeF_3 in ternary LiF-NaF-KF and LiF-ThF_4-UF_4 melts was measured in the temperature range 550-800°C. The effect of metal trifluoride additivities on the viscosity of molten salt mixtures was also measured [5-7].

Measurements of the chemico-physical properties of UF_4, ThF_4 and PuF_3 containing salt systems has been completed focusing on melting point behaviour, vapour pressure, heat capacity, phase diagram data and enthalpy of transitions. A series of phase diagrams of the key systems have been assessed according to the Calphad method, resulting in an extensive thermodynamic database, recognised as the JRCMSD (Joint Research Centre Molten Salt Database), describing fluoride and chloride salt systems. The database was used to optimise the fuel salt composition coupling neutronics and thermodynamics to find the minimum melting point [8-12].

The transient fission matrix (TFM) approach has been developed specifically as a neutronic model able to take into account the precursor motion-associated phenomena, and to perform coupled transient calculations with an accuracy close to that of Monte Carlo calculations for the neutronics while incurring a low computational cost. These tools have been used for the preliminary safety assessment of the MSFR, to confirm the excellent safety behaviour of the fuel circuit of the reactor, to optimise its design and to start the definition of dedicated operation procedures.

Several multi-physics code systems have been developed using rigorous calculation methods for both the neutronics and CFD calculations. The code systems have been benchmarked, and validation efforts are planned based on experimental data from newly built setups [13-19].

d) Inactive salt testing loops. The SWATH-W and SWATH-S facilities were designed and commissioned to investigate salt heat transfer and phase change phenomena. Helium bubbling and liquid–gas separation tests have been performed in the FFFER (Forced Fluoride Flow for Experimental Research) facility. The data will be used to improve numerical models for molten salt design and safety studies [20].

An electrically heated and thermally insulated, forced convection FLiBe loop was built and commissioned in the Research Centre Řež. The loop is intended for MSR and FHR material research and component testing [21].

The liquid salt test loop (LSTL) was created at the Oak Ridge National Laboratory (ORNL). It is a versatile facility in support of the development and demonstration of FHR components [22].

Finally, existing test loops and loops being constructed within the framework of the TMSR programme are establishing an important experimental complex in support of future R&D on MSR.

e) On-site fuel processing. The process to extract lanthanum, neodymium, europium and samarium trifluorides from a 73LiF-27BeF_2 melt (mole %) into liquid bismuth at 600-610°C was studied [23-25]. Technologies such as fluorination for uranium recovery, distillation for salt purification and fluoride electrochemical separation for uranium recovery were developed and tested in "cold" experiments.

Within the framework of the TMSR programme, hot cells for future irradiated fuel experiments were constructed. Moreover, TMSR has conducted research on tritium striping using a bubbling system, tritium separation from noble gases using cryogenics, tritium storage using alloys, and continuous tritium monitoring.

In the area of MSR chemistry and the chemical technology of MSR, present efforts and future directions cover the development and experimental verification of fused salt volatilisation techniques proposed for the extraction of uranium (in the chemical form of UF_6) from the MSR fuel salt. The present effort in electrochemistry is focused on the development and verification of quantitative electrochemical extraction of uranium and thorium on removal of main neutron poisons (fission products) from the MSR carrier salt ($LiF-BeF_2$). Special attention will be paid to the electrochemical studies of protactinium [26].

f) Mock-up demonstrator without induced fission. The mock-up facility TMSR-SF0 was designed (and is currently under construction) within the framework of the TMSR programme. It will provide data for the validation of thermal hydraulics and safety analysis codes.

g) Demonstrator with induced fission. The irradiation experiment SALIENT-01 (SALt Irradiation ExperimeNT) of small $78LiF-22ThF_4$ salt samples in graphite crucibles was planned and is currently being conducted at the Petten High Flux Reactor (HFR). In parallel, a concept design was developed for a 125 kW (fissile) molten salt loop driven by neutrons from HFR Petten as a demonstrator for in-pile performance of an integrated system.

In the next phase, two reactor units will be needed to demonstrate final technology readiness (TRL8/9). The first, named MONO, would be a small power unit (up to 100 MW_{th}), probably representative of a single loop of the larger reactor plant, and the second a 1 000 MW_{th} reactor named DEMO, which would have 16 circulation loops. DEMO is necessary to establish the basis for obtaining the approval of safety authorities: to demonstrate the control of the reactor; and test the management of the active fuel salt (start-up, transition to equilibrium, drain-out, shut down etc.) with its volatile and fission products. It will allow testing of all the structural materials under real conditions.

The conceptual design of the TMSR-LF1 test reactor ($LiF-BeF_2-ThF_4-UF_4$ fuel, thermal neutron spectrum) is ongoing. The reactor is scheduled to reach criticality in 2020 using existing TMSR funds.

MSR development needs for the next ten years can be expressed in terms of the following grand challenges:

- Identifying, characterising and qualifying successful salt and material combinations for use in MSRs.
- Developing integrated reactor performance modelling and safety assessment capabilities that capture the appropriate physics and fuel chemistry needed to evaluate plant performance over all appropriate timescales and to license MSR designs.
- Demonstrating the safety characteristics of the MSR at laboratory level and beyond.
- Establishing a salt reactor infrastructure and economy that includes affordable and practical systems for the production, processing, transport and storage of radioactive salt constituents for use throughout the lifetime of molten salt reactor fleets.
- Developing a licensing and safeguard framework to guide research, development and demonstration.

As regards progress towards demonstration, it is noted that MOSART development efforts are focused on the realisation of an advanced reactor with the objective of closing the nuclear fuel cycle for all actinides, including Np, Pu, Am and Cm. A possible site for the construction of the Li,Be/F MOSART reactor plant is the Mining and Chemical Combine (MCC). The unique technical and technological capabilities of the MCC site provide the opportunity to locate an experimental fast neutron spectrum MOSART unit with thermal power of up to 100 MW in close proximity to the VVER used-fuel reprocessing facilities, linking it to the infrastructure of the Experimental Demonstration Centre. MCC and the NRC "Kurchatov Institute" are advancing programme plans for the development of the experimental MOSART reactor [27-28].

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3.5. Lead-cooled fast reactor (LFR)

The LFR offers a long-term option for a class of reactors that promise high levels of economic and safety performance and sustainability. At the same time, LFRs require additional development in a number of technical areas before reaching full industrial maturity. The main R&D objectives for the development of the LFR were identified in the 2014 GIF *Roadmap Update* and are briefly summarised below:

- materials corrosion and development of a lead chemistry management system;
- core instrumentation;
- fuel handling technology and operation;
- advanced modelling and simulation;
- fuel development (MOX fuels for the first core, then incorporating MA in follow-on fuels);
- nitride fuel for lead-cooled reactors (BREST, SSTAR);
- actinide management (fuel reprocessing and manufacturing);
- ISIR techniques for opaque medium;
- seismic impact mitigation.

For each of the topics identified above, last years' major achievements are briefly summarised in the following sections.

Materials corrosion and lead chemistry management system [1-7]

The potential for corrosive attack by molten lead at high temperature has historically been one of the main obstacles to bringing lead technology to a demonstration phase. However, considerable research effort has been dedicated to this topic in the past 20 years, yielding a much deeper understanding of corrosion/erosion processes. This research has also led to the development of new corrosion-resistant materials and technical approaches for prevention and mitigation of corrosion/erosion.

Corrosion in lead (and, more generally in heavy liquid metals [HLM]) is mainly due to the dissolution of the main constituent elements of steel, such as nickel, chromium or iron, into the molten lead. Additionally, this process is accompanied by the oxidation of the steel constitutive elements. Material degradation can be further increased by high flow speeds of lead near material surfaces (erosion), the main motivation for the LFR designers to typically limit the maximum lead flow speed to about 2 metres per second.

Material corrosion protection strategies currently being developed rely on different approaches, but all of them are based on the formation of a thin oxide layer to protect the steel from corrosion. These strategies are accompanied by the control of operating conditions (temperature and oxygen content in liquid lead), and are underpinned by an understanding of the underlying corrosion mechanisms.

Russia developed for this purpose a ferritic-martensitic steel doped with silicon. In this case, the high silicon oxidation potential promotes the formation of a silicon oxide layer on the surface of steel, which, in controlled oxygen concentration conditions, is able to protect the underlying structural material.

A similar approach forms the basis of the ongoing development and qualification of alumina-forming austenitic (AFA) steels within Euratom.

Other approaches, typically used for the structures experiencing the highest temperature (e.g. cladding), involve the application of corrosion-protective layers by different techniques to provide corrosion/erosion protection through deposition of alumina layers (coatings), or through surface treatment by laser or electron beam.

Experts in the field are in any case convinced that the technology developed in the past 20 years in relation to materials and prevention/mitigation approaches to corrosion/erosion is sufficiently advanced, up to the level necessary to support construction of a demonstrator reactor.

Instrumentation [8–10]

For the reactor core in particular, a large and redundant set of thermocouples is expected to be needed to monitor the outlet fuel assembly temperatures and to provide early detection of any deviations from normal operation (for example, those caused by flow blockage). Radiation detectors for in-core neutron flux monitoring are under development in several countries. Outside the primary circuit, the reactor can be equipped with instrumentation already developed for LWRs.

For experimental facilities, the design and implementation of prototypical instrumentation (including flow meters, pressure transducers, resistivity probes and bubble tubes) is already a reality.

Special competence has been developed in the contactless control of HLM flows by magnetic fields and in the development of various measurement techniques (in particular, for integral and local HLM velocities).

Fuel-handling technology and operation [11]

Refuelling in lead is a critical operation. While giving due consideration to coolant opacity and to the need for proper engineering solutions to move/store used fuel, the relatively inert chemical nature of lead in contact with air, combined with the very low vapour pressure of molten lead, should allow relaxation of the otherwise stringent requirements for gas tightness of the reactor head and possibly allow the adoption of a simplified fuel handling scheme. For the European Lead Fast Reactor (ELFR), the proposed approach is based on the extension of the fuel assemblies (FAs) to allow fuel handling using a handling machine operating in the cover gas at ambient temperature under full visibility.

For BREST the strategy is more “conventional”. BREST fuel assembly heads are below the lead level and the reactor is equipped with in-vessel fuel storage. Fuel assemblies and reflector blocks are reloaded with the aid of rotary plugs, using an in-pile reloading machine and a set of mechanisms for out-of-pile handling (using a technology already developed and used for other liquid metal reactors).

Advanced modelling and simulation [9-10, 12-15]

Modelling and simulation have always been among the main tools used by LFR designers to identify new solutions and provide confidence and verification for proposed design choices. The activity started with the introduction – in the so-called "system codes" already in use for LWRs or other liquid metal cooled reactors – of the available lead (and lead-bismuth) properties and heat transfer correlations. Then, a substantial effort was dedicated to the generation of appropriate experimental data for code validation and for the development of new LFR-specific correlations.

CFD has also been extensively used for the simulation and analysis of specific reactor parts where three-dimensional approaches are needed to correctly take into account the physical phenomena. This methodology was also the subject of verification and validation activities.

Neutronic data have been improved within the framework of several Euratom collaborative projects with the aim of reducing the uncertainties of the simulation tools.

More recently, molecular dynamics tools are also being used to investigate in more detail the interaction of the heavy liquid metal coolant with materials. These studies may bring additional insights for the development of corrosion/erosion resistance approaches.

MOX fuel development [16]

The European Sustainable Nuclear Industrial Initiative (ESNII) is promoting MOX fuel in the European Union as the baseline fuel option for fast reactors. It is important to note that this can be considered the most important cross-cutting topic between different fast reactor designs (SFR, LFR and GFR), and as such is addressed within the Euratom framework. In 2017, a new Euratom collaborative project (INSPYRE) was launched, focusing on the investigation of fast reactor MOX fuel in support of the licensing of fast reactor demonstrators/prototypes. However, the main difficulty in Europe remains the absence of the industrial capability for the production of such a fuel.

Nitride fuel development [17-18]

In Russia, a fuel manufacturing plant for nitride fuel is being built on the BREST site. A co-located nitride fuel reprocessing plant is planned for future construction to implement on-site closure of the fuel cycle. Nitride fuel testing and qualification started in 2013 and is currently under way with nitride fuel assemblies being irradiated in both BOR-60 and BN-600 reactors. The first results obtained from such tests support the choice of the nitride fuel for BREST.

Actinide management [19]

Actinide management is a very important cross-cutting topic and is not peculiar to LFR technology. It relates to the closed fuel cycle and sustainability of the Gen IV reactors, and as such it is common to all fast reactor designs. In Europe, the LFR demonstrator will use fresh MOX fuel and no recycle is foreseen. While the recycling of MOX fuel has already been demonstrated in Europe (for example in Phénix and Superphénix), it should be noted that the development of such capability is presently beyond the scope of the LFR demonstrator.

In Russia, the present strategy is based on an LFR plant co-located with both fuel manufacturing and reprocessing facilities designed to achieve homogeneous on-site recycling, which implies that the actinides will be recycled with the objective of reaching a plutonium equilibrium content in the fuel.

ISIR for opaque medium [20-21]

Most of the techniques developed for other liquid metal coolant systems can be used for the LFR as well. The additional developments needed relate to the heavy fluid characteristics (buoyancy forces) and higher inspection temperatures. Ultrasonic view systems can be used and provide very good image definition results. In general, however, the present GIF designs under consideration are based on the concept of maximising the number of components that can be extracted from the primary pool, in order to simplify inspection and repair.

In addition, techniques to clean components that have been removed from the reactor to eliminate residual lead coolant have been developed and are already routinely used for experimental facilities.

Seismic behaviour [22]

Concerning seismic behaviour, the SILER project co-funded by Euratom provided the demonstration that an LFR rated at 600 megawatt electrical (MW_e) can be mounted on seismic isolators, substantially improving the seismic response behaviour. Other LFR designs, like BREST, use a different approach based on the development of a novel primary vessel concept using a multilayer concrete-metal structure with relatively small interconnected pools.

R&D activities for the next decade will be driven by the specific needs of each country and entity pursuing the common objective of LFR system deployment. While in Russia the technology readiness level was judged to be able to support the construction licensing of BREST, in Europe additional investigations are needed to support detailed design, safety assessments and the licensing of envisaged concepts, specifically in relation to their chosen reference temperatures. In any case, a number of topics still deserve the full attention of the scientific and industrial communities devoted to LFR development.

The main R&D topics to be pursued over the next decade are:

- **Phenomenology of lead-water/steam interactions:** Experimental investigation and code qualification are needed with the aim of demonstrating that steam generator tube ruptures are of no safety concern in terms of their effects on the primary side as well as on water/steam transport to the core. Experimental facilities to address this topic are being planned in the European Union.
- **Prevention mitigation of sloshing:** In relation to seismic response, the sloshing effect needs to be investigated on specific designs, both experimentally and numerically. New activities to address this issue are being planned to support the completion of new LFR designs.
- **New corrosion-resistant materials (including surface modifications):** Material compatibility is clearly another main topic to which R&D efforts should be devoted. Present lines of research planning envisage the development of AFA steels, oxide dispersion-strengthened (ODS) steels and other innovative structural materials that may become available in the near future. These activities are organised in the European Union inside the Joint Programme on Nuclear Materials of the European Energy Research Alliance (EERA–JPNM) and are subject to periodic revisions as the research brings new results. Parallel to these efforts, the development of so-called “coating” (surface modification) technologies is also under way. These technologies are expected to have qualification needs similar to those required for the “new” materials. Additional efforts also need to be devoted to mastering coolant chemistry and purification control.
- **Operation and maintenance (O&M):** O&M is not well developed for some LFR concepts and therefore needs further R&D, in particular activities to address innovative technologies for ISIR and fuel/component handling. Development and qualification of some O&M aspects will need to be finalised as the first realisation of an LFR takes place.
- **Fuel** and fuel reprocessing require very important investments, not only dedicated to LFR technology but also in general to allow nuclear energy to become sustainable, and in particular in relation to innovative MA-bearing and high burnup fuels. The industrial production capacity of MOX fuel in particular is to be addressed more specifically in the European Union.
- **Advanced modelling and simulation** play an important role for safety assessments as well as for reactor design and analysis, while also providing new insights into the design of experiments, new material development, coolant interaction with materials, etc. In this context, further development and qualification of LFR-specific modelling

methodologies and tools are also needed, including mass-transfer in the primary coolant and design extension condition accident modelling.

- **Design codes and standards** need an "LFR-dedicated" section, taking into account the environmental effects of heavy liquid metals on structural materials and welds and providing, *inter alia*, guidelines for the design of mechanical components of LFRs. Efforts are being carried out through lines of research identified by EERA–JPNM in the European Union (e.g. in the frame of the GEMMA H2020 collaborative project) aimed at providing recommendations for possible adaptations of the existing RCC-MRx Design Code. Within GIF, dedicated safety design criteria and guidelines for GIF LFRs are also being further developed [23].

However, some of these developments are expected to be fully realised only when the first demonstrator is commissioned, and an adequate return from experience has finally been collected.

Severe accidents deserve special attention related to several aspects and phenomenology. These include fuel-coolant thermodynamic and chemical interactions, including the behaviour of dispersed fuel, and retention of fission and activation products in heavy liquid metal.

In Europe, with the use of MOX fuel, some efforts have been devoted to demonstrating that in the case of a temperature excursion and cladding failure, the fuel may be dispersed in the coolant without compaction, preventing core re-criticality. This topic nonetheless necessitates more detailed investigations, and collaborative projects need to be proposed in this area.

In Russia, where nitrides are being considered, the adopted safety approach was not able to identify any initiating event leading to the generalised (whole) core melt. Although the intrinsic characteristics of the fuel (melting temperature of 2 700°C and high thermal conductivity) are expected to prevent contact of melted fuel with the coolant, further investigations in this area will need to be performed.

Far from being exhaustive, the above discussion simply indicates priority topics for LFR development. It is nevertheless important to note once again that some basic and fundamental steps will be made only after the first Gen IV LFR demonstrator becomes a reality.

Progress towards demonstration/deployment of a Gen IV LFR

Since the early efforts to develop HLM technology, numerous improvements, design innovations and material developments have been implemented. As a first example of these innovations, the concept and design of the BREST-OD-300 vessel developed in Russia can be highlighted. The vessel (if one may still use this word) is comprised of a multilayer structure of metal and concrete, which allows for the design of a very innovative primary system, and which could conceptually be considered as an amalgam of pool and loop configurations. The vessel is a 26-metre-diameter cylinder about 17.5 metres in height, and in which compartments are provided to host the main components, including the core, steam generators, pumps and safety systems. The individual compartments are connected by horizontal channels to provide the coolant flow paths in the primary system. This design concept is a prime example of a technological innovation able to provide substantial improvements in the design of the primary system.

Substantial advances have also been made in the design of various components, including steam generators, pumps and DHR systems. For steam generators, the helical-type once-through steam generator is one of the reference designs presently being considered, with the tube bundle directly immersed in the LFR primary system. Several different pump designs have been investigated, and mechanical pumps have been selected to provide the volumetric flow required to cool the reactor. Bayonet-tube heat exchanger bundles were investigated as a part of the DHR safety system, using double-wall exchanger tubes to provide a very high degree of reliability and continuous monitoring of the status of the tube bundle itself. Regarding in-service inspection, one of the strategies adopted is to specify in the conceptual design that all the important components must be removable. Most of the designs considered follow this strategy in order to minimise the impact and enhance the reliability of ISIR.

When employing lead as a coolant, in fact a number of new design options become possible by carefully considering the basic characteristics of the lead coolant and fully exploiting these characteristics from the early phases of the design. As an example, another design innovation that profits from the high density of the coolant can be cited: namely, in certain LFR design concepts, control or shutdown rods are located below the core and are inserted passively into the core by buoyancy force, since the rods are lighter than the displaced lead coolant. The same characteristic requires that fuel assemblies be held down into the core since they naturally float in lead. Moreover, the relative inertness of lead coolant in contact with air and water, while advantageous for safety, also provides an opportunity for the elimination of an intermediate cycle.

These examples illustrate design considerations that depend on the specific characteristics of the coolant, and the many possibilities for innovation that the coolant itself opens to a designer. During the last several years, a number of different proposals have been developed for advanced LFR components, and it is strongly believed that there is considerable space for the possible implementation of new ideas and innovation.

The construction of BREST-OD-300 in Russia is expected to start as soon as the licensing process is completed (documentation was submitted to the safety authority in 2016).

Recently, there is renewed interest worldwide in the development of the LFR technology. In the United States, for example, several private companies are pursuing development of LFR designs, largely as a result of the US Department of Energy signing the GIF-LFR memorandum of understanding in February 2018. There is also growing interest in China, with several promising initiatives being implemented there.

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3.6. Supercritical water-cooled reactor (SCWR)

The SCWR has a long-term vision for LWRs that requires significant development in technical areas (such as materials, chemistry, thermal hydraulics, neutronics, instrumentation and control). At the same time, the SCWR benefits from interest in LWRs, as well its use of established technology for supercritical water power cycle equipment in the fossil-fired power industry. The overall plan for the SCWR within GIF is to complete its viability phase research by about 2020 and operation of a prototypical fuelled loop test by about 2025, thereby preparing for construction of a prototype sometime after 2035. Main R&D activities identified in the SCWR System Research Plan include system integration and assessment, materials and chemistry, as well as thermal hydraulics and safety.

Major R&D achievements in response to the R&D challenges identified in the 2014 GIF *Technology Roadmap Update* are summarised in the following paragraphs.

System integration and assessment

Canada, the European Union and Japan have completed their development of the thermal-spectrum SCWR plants, which were reviewed by international peers for their viability. New features or components have been introduced to drive R&D efforts and improve the technologies. Information has been disseminated to the SCWR community through the information exchange meetings and the international symposia on SCWR.

Uranium fuel cycles have been adopted for pressure-vessel-type thermal-spectrum SCWRs. Levels of enrichment and neutron absorption in fuel rods vary depending on the fuel assembly concept and burnups. The fuel remains in the core for mostly three (batch) refuellings and is discharged for disposal afterwards (once-through cycle).

The thorium fuel cycle has been selected as the reference (uranium fuel cycles are also applicable) for the pressure-tube-type thermal-spectrum SCWR. This cycle is based on mixed plutonium and thorium fuel and has evolved from the fuel proposed for the Advanced Fuel Cycle Reactor (AFCR), which has adopted uranium and thorium fuel. Reactor-grade plutonium is extracted from the spent fuel of light and heavy water reactors to breed U-233 from thorium, which will then be used to replace plutonium to close the fuel cycle.

MOX fuel cycles are adopted as the seed fuel, while uranium (or depleted uranium) is selected as the blanket fuel of fast-spectrum SCWRs. Seed and blanket fuels are distributed either heterogeneously (in separate assemblies) or homogeneously (wafered in a fuel rod). MOX fuel can also be used in the pressure-tube-type thermal-spectrum SCWR.

Fuel assembly concepts for the thermal-spectrum SCWRs have been developed from current nuclear reactors. Two main differences in fuel assembly configuration between the SCWR and current reactors are the introduction of a central water channel to increase moderation, which is relatively low for supercritical water due particularly to the low density of the vapour-like fluid, and the use of wire-wrapped spacers. Fast-spectrum SCWRs consist of hexagonal-shaped fuel assemblies. Japan's concept separates the seed and blanket fuel pellets into two different assemblies. Russia's concept, however, combines the seed and blanket fuel pellets (wafered) within a single assembly.

Advanced in-reactor and out-of-reactor components have been developed to withstand the high pressure and temperature operating conditions of SCWRs [1, 2]. The outlet header has been included within the inlet plenum to reduce the pressure and temperature gradients over the header wall, significantly reducing the material thickness and associated thermal stress. Similarly, an insulation sleeve has been developed for the steam pipe at the core outlet to minimise the temperature gradient.

Most pressure-vessel-type SCWRs adopt the multi-pass coolant flow configuration. Advanced components have been developed to separate the core into regions for various flow paths [3]. Dividing headers have also been introduced to collect the coolant at each region and distribute the coolant between regions.

An innovative single-ended fuel channel has been introduced to the pressure-tube-type SCWR [4]. It is attached to the tubesheet of the inlet plenum and contains the fuel assembly. A central flow tube has been introduced in the fuel assembly for the relatively low-temperature coolant to travel from the inlet plenum to the bottom of the fuel channel.

Materials and chemistry

SCWR R&D has been focusing on the technological development of advanced materials. A key challenge for materials is certainly the selection of a cladding material for elevated temperatures beyond 600°C, where Zr alloys are not applicable.

The approach for the development of materials and components builds on: a) the evaluation of candidate materials with regard to corrosion and stress corrosion cracking, strength, embrittlement and creep resistance, and dimensional and microstructural stability; b) the potential for water chemistry control to minimise impacts as well as rates of deposition on fuel cladding and turbine blades; and c) the measurement of performance data in an in-pile loop.

Candidate materials for fuel cladding have been identified. Corrosion characteristics for one of these materials were studied through the round robin exercise at supercritical pressures and temperatures [5]. The weight-change data exhibited considerable discrepancies between laboratories. A follow-up campaign on the round robin corrosion exercise has therefore been initiated.

The corrosion resistance of several materials exposed to pressures of 0.1 MPa, 8 MPa and 29 MPa at 625°C for 1 000 hours has been investigated [6]. The result confirmed the qualitative agreement for corrosion testing in superheated steam at low pressures with that of the operating pressure of SCWRs (i.e. 25 MPa). A thinner scale was observed on Alloy 800H than on stainless steel SS316; it provides improved corrosion protection in the presence of hematite [7]. The initial stages of oxidation were studied by monitoring concentrations of dissolved metals, oxygen and hydrogen at the exit of an Alloy 800H tube exposed to oxygenated water at supercritical pressures [8]. Hydrogen evolution from an oxidised stainless steel SS-316 surface was found to be about six times larger than that of Alloy 800H [9]. The activity release rate of ^{60}Co due to corrosion was the largest for SS316 at 650°C and pH10.

A slow strain rate tensile test showed that the stainless steel SS-310S material has high creep cracking susceptibility at 500°C, which contributed to the overall crack growth rate in a supercritical water environment. Another experiment identified that creep is the major cracking mechanism contributing to the cracking of cold worked Alloy 690 in supercritical water. A round robin test programme on stress-corrosion cracking (SCC) was initiated using tapered test samples to define and compare the threshold SCC conditions of different alloys.

Austenitic stainless steels are the main candidate material group for the internals and fuel cladding [10]. Oxide dispersion-strengthened ODS) steels may be an alternative to replace austenitic steels at high operating temperatures, but key challenges in manufacturing processes need to be solved.

Different techniques have been examined in the past five years to improve the oxidation resistance of traditional low-alloyed austenitic steels (e.g. by using coatings and cold working) [11]. A combination of ex-situ analytical studies of the oxide film forming processes with modelling approaches was introduced [12].

Thermal hydraulics and safety

The thermal hydraulic characteristics of SCWRs during normal and off-normal operation, as well as during postulated accidents, were evaluated using analytical tools developed for the current fleet of light and heavy water reactors. Experimental data and analytical results are required for: a) the basic thermal hydraulic phenomenon of heat transfer and fluid flow of supercritical water in various geometries; b) critical flow; c) quantification of flow stability and transient behaviours due to the strong coupling of neutronic and thermal hydraulics; d) validation of computer codes; and e) definition of the safety and licensing approach as distinct from current water reactors, including the spectrum of postulated accidents.

Extensive databases on supercritical heat transfer have been compiled for tubes and bundle subassemblies in water and non-aqueous fluids. The water heat transfer database for tubes contained over 24 000 data points and is one of the largest compilations in the community [13]. These data have been applied in validating correlation and developing new prediction methods.

A number of test facilities have been constructed [14]. Most of these facilities were established for heat transfer tests with tubes, annuli and bundle subassemblies in water, carbon dioxide or refrigerant flows. Separate test facilities for critical flow and flow stability experiments at supercritical pressures were also constructed.

Experiments were performed with vertical annuli in upward water flow at supercritical pressures [15]. Heat transfer was enhanced in the annular test section. The presence of spacers enhances heat transfer significantly at the spacer location; the enhancement diminishes downstream of the spacer. Effects of the presence of the wire-wrapped spacer and the geometry on heat transfer in bundles were assessed using CFD software tools [16]. A strong non-uniformity of surface temperature was identified for the bare-rod bundle, but it is mitigated by the presence of the helical-wire spacers.

The effect of flow direction on heat transfer is of particular interest to SCWRs with multiple flow passes. Wall temperature measurements are similar for upward and downward flows at low heat fluxes, but are generally higher for upward than downward flows at high heat fluxes [17].

The majority of experimental studies focused on obtaining data on cladding surface temperature at bulk temperatures close to the pseudo-critical point [18]. Experiments with water or surrogate-fluid flow through small electric-heated bundle assemblies were performed to investigate the deteriorated heat transfer phenomena and the effect of spacers on heat transfer [19]. A recent blind benchmark exercise on heat transfer in an electrically heated rod bundle in supercritical water confirmed that further work is required to provide reliable predictions [20].

As regards the path forward, the following R&D objectives and needs have been identified to guide future SCWR system R&D efforts.

System integration and assessment

The core concept of China's thermal spectrum SCWR has been established. A plant concept based on the advanced boiling water reactor is being developed. An international peer review of the SCWR concept is planned.

Most SCWR concepts are developed for large baseload power generation of over 1 000 MW_e, which are considered excessive for small remote communities, small mining operations and oil production. With the modular configuration, SCWR concepts can be scaled down to meet the needs of local deployment. The development of small and very small SCWR concepts will be initiated.

The construction of a demonstration plant for generating a small amount of power has been planned with technical support from the international SCWR community. Its core is likely to be the pressure vessel type, although there is little difference between pressure vessel and pressure tube types for a small core. The design work could be initiated after the completion of the viability phase.

Uranium pellets inside the fuel rod behave similarly to those of the pressurised water-cooled reactor. Therefore, no additional requirements are needed. Optimisation of the fuel assembly (e.g. rod diameter, gap sizes) is continuing, with the focus on lowering the fuel and cladding temperatures and increasing the burnup to further enhance the economic and safety characteristics.

The current approach for the pressure-tube-type SCWR is to introduce the "once-through" thorium fuel cycle at start up. Qualification of the reprocessing technique for spent thorium fuel will be considered. Once the technique is qualified, the spent thorium fuel can be recycled, closing the fuel cycle. In addition, research has been planned to improve the homogeneity of mixing thorium and plutonium in the pellet. Different mixing techniques, using uranium as a surrogate of plutonium, will be considered.

Qualification testing of the SCWR fuel will be needed in a research reactor at representative operating conditions. A supercritical-pressure water loop has been constructed for installation into the LVR-15 reactor of the Řež Research Centre in the Czech Republic. Out-of-pile commissioning is being performed. Once ready, the loop will be installed into the research reactor for material testing. Plans have been made to apply for a licence to operate the loop for fuel testing. Another supercritical pressure water loop for fuel testing has been designed for a new research reactor at the Nuclear Power Institute of China.

Installation of wire-wrapped spacers on a long fuel rod is considered challenging. Current welding techniques change the microstructure of the cladding at the weld location, potentially weakening the material integrity. New welding techniques are currently being studied to minimise adverse impacts on the cladding material.

Insulation attached to the water channel is needed to maintain the outlet temperature of the core (power generation in the core needs to be raised to compensate for the increase in heat losses to the water channel). This design feature is particularly crucial for the pressure-tube-type fuel assembly concept, since insulation is needed not only for the water channel, but also for insulating the inner wall of the pressure tube from the high-temperature coolant. Experiments are currently being set up to measure thermal conductivity of the insulator (yttria-stabilised zirconia) at operating conditions in the fuel channel.

A lack of reactor physics data is evident in the case of validating the analytical predictions. A code-to-code benchmarking exercise is planned to reduce the uncertainties in reactor physics calculations for the proposed SCWR concepts. A neutron scattering experiment has been planned to obtain cross-section data with supercritical water to validate the nuclear data. Potential spallation neutron sources are being considered to perform the experiment. An experimental cell has been designed for operation in high-temperature and high-pressure conditions (up to 550°C and 25 MPa). Construction of the cell will be initiated after discussions with the organisation hosting the spallation neutron source.

Reactor physics data will be needed for designing, licensing and operating the demonstration plant and deployment of SCWRs. A feasibility study has been planned to design and install a supercritical pressure and temperature channel into the zero power research reactor (ZED-2 reactor) for reactor physics experiments. If feasible, design and construction of the experimental channel will be carried out.

An experimental study of the flow path through the fuel assembly of the pressure-tube-type SCWR has been initiated. It will provide data for validating the computational fluid dynamics tool. A transparent test facility is being designed for the experiment using water or other surrogate fluids at low pressures to simulate high-pressure water conditions. Construction and testing will be initiated after the design.

One of the key components in the fuel channel of the pressure-tube-type SCWR is the co-extruded joint between the zircaloy pressure tube and the stainless steel extension, which is welded to the tubesheet of the inlet plenum. The co-extrusion technique of two dissimilar materials was developed using a bench-top scale apparatus. It needs to be qualified and proven for industrial manufacturing. Two pieces of metal plate (one stainless steel and the other zircaloy) have been manufactured for the co-extrusion process.

A 3-D printing technology is being considered as an alternative approach to joining two dissimilar materials. It was used to lay different proportions of stainless steel and zirconium powder over each layer. Cracks were observed at each layer of the printed joint metal, affecting its mechanical strength. Different 3-D printing techniques will be pursued.

The economic benefit of the small SCWR will be assessed using the GIF G4ECONS analytical tool. Capital cost reduction will be compared against those for the reference SCWR and light water reactors. The high core outlet temperature of SCWRs facilitates cogeneration, such as hydrogen production, space heating and steam production. An economic assessment related to the adoption of the copper-chlorine cycle to produce hydrogen with a small SCWR will be performed using the GIF G4ECONS analytical tool.

Materials and chemistry

Candidate alloys have been identified to meet the performance requirements under most in-core conditions. Significant progress has also been made on the evaluation of candidate alloys for all key components in SCWRs. However, there is a lack of experimental data for any single alloy to unequivocally ensure its performance in a SCWR core. Mechanical properties of these candidate alloys need to be quantified at high temperatures (higher than 700°C) and pressures (more than 25 MPa).

Additional experiments will be performed to improve the understanding of long-term corrosion and stress corrosion behaviours of fuel cladding alloys at high temperatures. The effects of inter-granular carbides, surface finish and surface treatment on corrosion will also be examined. Modelling tools will be developed to evaluate the performance of candidate materials for long-term service life (> 10 000 hours). A second round robin corrosion test is currently under way to confirm the corrosion behaviour observed at various testing facilities.

The effect of slow microstructural degradation will be examined on fracture and SCC resistance of in-core and out-of-core materials. Round robin SCC testing has been initiated with material samples distributed to participating laboratories. Studies are ongoing to improve the bellows-based testing technology, together with miniature autoclaves capable of working in a hot-cell environment [21]. A post-irradiation study is planned to characterise the effects of irradiation damage and helium generation on the microstructure and the micromechanical properties of candidate alloys, which were previously irradiated in the High Flux Isotope Reactor (HFIR) at ORNL [22].

In addition to the focus on candidate alloys, the development of advanced coating techniques is ongoing, with the objective of mitigating the corrosion and SCC issues of in-core and out-of-core components. Appropriate coating techniques for in-core (such as fuel cladding) and out-of-core (such as the inlet plenum and outlet header) components will be identified.

Additional effort will be required to study the effect of chemistry and radiation damage on the behaviour of candidate core materials so as to ensure viability of these materials in SCWRs. A preliminary chemistry control strategy has been established. However, the effect of water radiolysis needs to be quantified. Experiments have been planned for candidate alloys with out-of-pile and in-pile loops under water chemistry regimes of interest to SCWRs. These experiments will facilitate the examination of the impact of chemistry on radiolysis and corrosion product transport, as well as the effect of neutron flux on corrosion rates and SCC (or related failure mode).

Additional experiments will be performed to measure the solubility of activation products (such as iron, nickel and chromium) in supercritical water and to evaluate the molecular sieve capture efficiency and sorption capacity. Requirements for chemistry monitoring will be established with a more enhanced understanding of the radiolysis effect on corrosion behaviours and the release transport of activation products.

The need for materials-related in-pile tests is important to study the effects of irradiation, in conjunction with other possible degradation modes, on materials performance in SCWR service, while taking into account additional effects (e.g. water radiolysis). There are opportunities for in-pile tests in Europe and China to examine the effects of irradiation on long-term mechanical properties.

Post-irradiation microstructural characterisation of Alloy 800H will be performed to assess the relative significance of phase instabilities, solute segregation, radiation hardening and helium generation. Confirmatory out-of-core tests on helium and proton irradiated specimens at temperatures of 350°C to 800°C will be performed to study helium embrittlement using micromechanical testing. Additional neutron irradiation studies to displacement-per-atom levels consistent with those of the fuel clad of SCWR concepts are planned for the HFIR at ORNL in the United States or the LVR-15 facility at the Řež Research Centre in the Czech Republic.

The zirconium alloy "Excel" has been selected as a candidate pressure-tube material with Zr-2.5Nb, an alternative material for Canada's SCWR. A limited amount of data on corrosion, hydrogen ingress, corrosion fatigue, delayed hydride cracking (DHC) and irradiation deformation are available at the operating temperatures of the pressure tube. In-reactor tests have been planned to determine the effects of irradiation under low temperature (390-500 K) on deformation, fatigue and fracture for pressure-tube material Excel. Similar tests are planned to examine corrosion resistance, corrosion fatigue, DHC and hydrogen pickup using the SCW loop at the LVR-15 facility in the Czech Republic.

Thermal hydraulics and safety

Hydraulic resistance over the fuel assembly affects flow distribution in the core and the design of pump capacity. A pressure-drop model has been developed for a wire-wrap-equipped fuel assembly by taking into account the increased friction due to the wrapped wires. It will be finalised and incorporated into the subchannel code for core flow analyses.

A heat transfer experiment is being performed using a 4-rod bundle with a symmetric-cosine axial power profile in water flow at supercritical pressures. Through an IAEA Coordinated Research Project, a benchmarking exercise against this experiment has been initiated. Analytical results will be compiled from benchmarking participants for comparison against the experimental data.

A lookup table approach to heat transfer coefficients for water in tubes has recently been developed, with an extensive database of over 22 000 data points. The approach is being implemented into the subchannel code in support of the development and optimisation of fuel assembly concepts. An assessment of the subchannel code with the implemented lookup table for heat transfer coefficients against experimental data for bundle assemblies has been scheduled.

A second benchmarking exercise is being prepared to examine the heat transfer deterioration phenomena in a 4-rod bundle equipped with wire-wrapped spacers. Participants are performing analyses with released test specifications. Results will be submitted to the organiser for comparison against the experimental data.

Separate effect experiments have been scheduled to examine the effects of spacers (wrapped wires or grid) on heat transfer in a 2x2 rod bundle with a short heated length. Experimental data will also be applied in quantifying the effect of heated length on heat transfer when compared against data obtained in experiments with bundles of a long heated length. Additional experiments are planned to study other separate effects (such as radial power profile, rod bowing) on heat transfer.

Heat transfer experiments with a simulator of the full-scale fuel assembly will be needed for the demonstration plant and deployment of SCWRs. The schedule for these experiments will be established once the optimised configuration of the fuel assembly is available.

Dynamic instability experiments were performed in single and two parallel channels, with water or carbon dioxide flow at supercritical pressures. Additional experiments will be carried out to understand the mechanism in detail and examine the effects of wall thickness and heating profile on dynamic instability. Furthermore, coupling analyses of thermal hydraulics and neutronics are planned to investigate dynamic instability behaviours in SCWR cores.

Previous safety analyses of postulated, large-break, loss-of-coolant accident events applied the critical-flow correlations developed for subcritical conditions, which were based on the separated-flow model. A critical-flow correlation has been developed with recent experimental data obtained at supercritical pressures. It will be implemented into the system codes for future analyses. Additional experiments with large size openings, relevant to the SCWR pipe break, will be performed to examine the impact of break size on critical flow characteristics. Furthermore, an experimental study of critical leak flow is planned to support the LBB analysis.

An integral testing facility is required to examine system behaviours during postulated accident events at SCWRs and confirm adequacy of the safety system. Ideally, it should be representative of the heat transport system of the SCWR for safety-related experiments (such as small- and large-break, loss-of-coolant accidents, loss-of-flow accidents). However, constructing such a facility would be costly and time consuming. A scaled facility would reduce considerably the construction cost and time, and yet be representative in simulating various phenomena. A scaling analysis would be required to establish the size, geometry and power requirements for the facility. A scaling analysis of the SCWR heat transport system is being performed. If feasible, design and construction of a scaled loop is planned.

Most SCWR concepts adopt the direct cycle, which is similar to that of the boiling water reactor. An indirect cycle is being considered for the small SCWR concept to minimise activity transport to turbines. It would increase the capital cost, but provide an opportunity to cool the fuel with natural circulation. A study is planned to examine the effectiveness of natural circulation for cooling the fuel and the core.

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Chapter 4. Major achievements and current outlook for the methodology working groups

4.1. Economics

The Economics Modelling Working Group (EMWG) was established under the Expert Group to develop a methodology to assess Generation IV (Gen IV) systems against the economic goals of the Generation IV International Forum (GIF). The economic assessment methodology released in 2008 consists of a comprehensive guideline for cost estimation, and an Excel-based software package G4ECONS v3.0. This methodology is publicly available through the Nuclear Energy Agency (OECD/NEA) in its role as GIF Secretariat and is already widely used within and outside GIF.

The EMWG completed benchmarking of its economic assessment tool G4ECONS against the International Atomic Energy Agency (IAEA) Nuclear Economics Support Tool (NEST) for both thermal reactors with a once-through fuel cycle and fast reactors with closed-loop fuel cycles, in collaboration with the IAEA. Results of the closed-loop fuel cycle benchmarking were distributed to the System Steering Committees (SSCs) of fast reactor systems (sodium-cooled fast reactor [SFR], gas-cooled fast reactor [GFR] and lead-cooled fast reactor [LFR]). A paper was co-authored with the IAEA on benchmarking work done to date, and the manuscript has been accepted for publication in a refereed journal [1]. The EMWG also published a paper demonstrating a top-down approach for cost estimation of Gen IV systems that are under development, as outlined in the EMWG's Cost Estimation Guidelines [2] using the supercritical water-cooled reactor (SCWR) as an example [3].

The EMWG has been reaching out to the SSCs to encourage use of economic methodologies from the early stages of concept development. A joint session of the EMWG and SCWR SSC was useful to demonstrate the G4ECONS economic tool and the cost estimation methodology. Subsequently, a paper on the economics of SCWR was published, as noted above. EMWG representatives also made a presentation at the very high temperature reactor (VHTR) SSC meeting. Several publications already exist on the economics of high-temperature gas-cooled reactors, including hydrogen cogeneration. The results of benchmarking of the IAEA and GIF economic assessment tools for fast reactors were documented in a note and distributed to all fast reactor SSC chairs (SFR, GFR, LFR and MSR) for distribution to other members. Most of the SSCs do not yet have good estimates of the capital cost or the fuel cycle costs – and associated uncertainties – that are required for economic analyses.

The EMWG will continue to reach out to the SSCs through communications on economic methodologies as well through feedback surveys. Additionally, the EMWG will continue to monitor the feedback from G4ECONS users, and will also keep a close watch on work being carried out in universities and other research institutions on nuclear economics for further development of its methodology.

The EMWG is currently working on the Policy Group Vice-Chair's initiative on evaluation of market issues related to deployment of Gen IV systems, especially considering the increasing capacity of renewable electricity production and the importance of capital cost reductions for the future competitiveness of nuclear power. The EMWG will continue to engage with the Senior Industry Advisory Panel (SIAP) on market issues for Gen IV systems, and will monitor work being done elsewhere on integration of renewable and nuclear energy systems to inform its evaluation study, which resulted in the publication in 2018 of a first position paper.

The paper identified the economic aspects of deployment of Gen IV systems in an integrated grid with significant renewable energy resources, requiring flexible operation and load-following capabilities. This work is expected to continue in collaboration with SSCs in order to understand more specifically the merits and R&D challenges of each of the six GIF systems to meet the flexibility needs of low-carbon energy systems. It would also address new market opportunities, looking at both load-following capabilities and new energy outputs.

Looking ahead, the EMWG's focus will move in two directions. Firstly, it will work on a more detailed identification of cost uncertainties, opportunities for cost reduction, and assessment of series effects from first of a kind (FOAK) to n^{th} of a kind (NOAK). This may imply revisiting the guidelines for cost estimation if necessary. Second would be a shift from plant-level economics to the system-level economic considerations that will impact the sustainability of nuclear generation and future deployment of advanced Gen IV reactors. As mentioned, it has been recognised that next-generation reactors will have to be flexible to respond to the variability and uncertainty of renewable power generation. Therefore, non-electricity applications of next-generation reactors will play an important role in the economic viability of future nuclear plants.

For these two research streams, the EMWG will also actively collaborate with SIAP in identifying the challenges that affect market deployment of Gen IV systems, to inform R&D needs for both cost reductions and flexibility required to create favourable market conditions for the deployment of Gen IV systems.

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4.2. Risk and safety

The primary objective of the Risk and Safety Working Group (RSWG) is to promote a harmonised approach to safety, risk and regulatory issues in the development of Gen IV systems. The early work of the RSWG focused largely on identification of high-level safety goals, articulation of a cohesive safety philosophy, and discussion of design principles, attributes and characteristics that may help to ensure optimal safety of Gen IV systems. Subsequently, the group developed an Integrated Safety Assessment Methodology (ISAM) with the goal of providing a practical and flexible tool that allows for a graded approach throughout the design process to the analysis of risk and safety issues.

More recent work has focused on the consolidation of the methodology with the development of the Guidance Document for ISAM (GDI) [1] and the publication of risk and safety white papers on the application of ISAM in close collaboration with the six reactor SSCs/provisional SSCs (pSSCs) [2]. The GDI provides a more detailed description of, and justification for, the integration of the different ISAM tools, and it includes practical guidelines for its application. The risk and safety white papers present the application of ISAM for the six Gen IV systems in order to demonstrate its validity and feasibility. In addition, the white papers may include practical feedback and recommendations with regard to further improvements of ISAM.

From the perspective of assisting the deployment phase of Gen IV systems, the RSWG is also working with the GIF task force on the development of safety design criteria (SDC) and guidelines (SDG) for SFR systems. With several RSWG representatives also being members of the task force, the working group provides feedback on the consolidation of the SFR SDC Phase 1 report, focusing on the SDC.

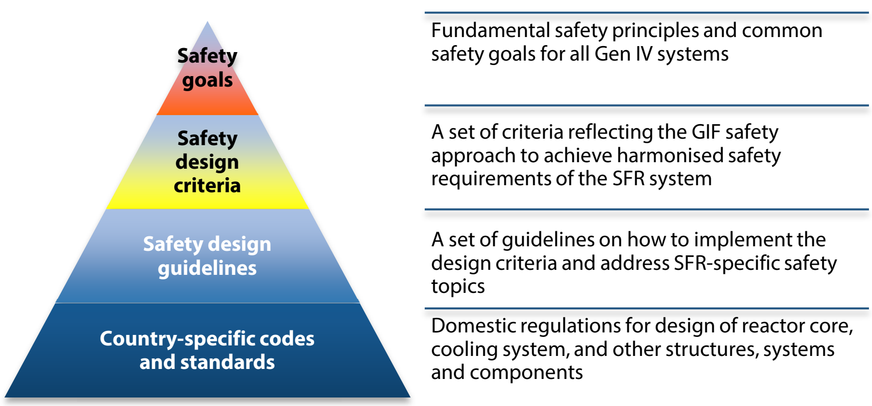
The task force was established in 2011 to develop SDC and SDG for SFRs. The main objective was to prepare reference criteria and guidance for the design of structures, systems and components to achieve the safety goals of the Gen IV SFR system design tracks. As shown in Figure 13, the SDC task force mission was to bridge the gap between the high-level safety fundamentals and the more detailed national codes and standards by developing the reference criteria and guidelines.

In 2013, the SDC report was completed and distributed to various international organisations and national regulators for review and feedback [3]. Based on comments received during the following two-year period, the report underwent a significant revision reflecting the feedback received from the IAEA, the Nuclear Regulatory Commission (United States), the Institute for Radiological Protection and Nuclear Safety (France), and the National Nuclear Safety Administration (China).

In parallel with the SDC report update, the task force has also initiated work on the development of SDG. The guidelines provide recommendations on how to comply with the SDC and present examples of good practices to help the designers to achieve a high level of safety in specific topical areas. The first SDG report on "Guidelines on Safety Approach and Design Conditions of Generation IV SFR systems" addresses the reliance on inherent/passive safety features and the design measures for prevention and mitigation of severe accidents. It was completed in 2016 and distributed to OECD/NEA's Working Group on the Safety of Advanced Reactors (WGSAR) and the IAEA for review and feedback [4]. The second report on "Safety Design Guidelines of the Key Structures, Systems and Components" is an ongoing effort intended to address the neutronic, thermal, hydraulic, mechanical, chemical and irradiation considerations that are important for the safe design of the reactor core, coolant systems and containment systems for Gen IV SFRs.

After completion of the second SDG report, the remaining core task force mission will be to synthesise and capture the feedback from external review of the reference SDC and SDG, especially to incorporate the regulatory perspectives from GSAR members to reduce the licensing risks. Implementation of the SDC and SDG by designers is also expected to stimulate safety-related R&D activities for Gen IV SFR systems so as to bridge the gap between the data needed for safety demonstration and the current technology and knowledge base. A long-term mission will also be integrating the knowledge and experience gained from the ongoing design, construction and operation of Gen IV SFR systems, based on the application of the SDC and SDG, including new knowledge from emerging trends and issues of regulatory importance.

Figure 13: **Hierarchy of safety standards**



Recently, the LFR and GFR pSSCs have started to develop their respective SDC with the support of the RSWG. In doing so, and with the objective of meeting their individual system safety and reliability goals, and conforming to the GIF safety approach basis, the LFR and GFR pSSCs are relying on the experience gained from the development of the SDC for SFR.

The current activities of the RSWG are in strong collaboration with the six SSCs/pSSCs. The focus of these activities is twofold: firstly, the finalisation of white papers on the application of ISAM, and, secondly, the preparation of the safety assessment reports for the six systems (Table 5). The safety assessment reports identify and review the main safety advantages and challenges of the systems and R&D needs on safety-related issues, with the aim of providing a snapshot of the major safety concerns [5]. The outlook for the RSWG activities is reflected in the plan to consolidate these separate risk and safety white papers, and separate system assessment reports, into two, respective, comprehensive synthesis reports covering all six systems. The white papers synthesis report has the objective of identifying and promoting a common and consistent risk-informed approach to safety in the design of Gen IV systems, including adequate feedback for further ISAM improvement. The system assessment synthesis report will highlight the strengths and weaknesses of the individual Gen IV systems, and propose necessary cross-cutting safety-related R&D.

Table 5: **Status risk and safety deliverables**

	White Paper on ISAM Implement.	System Safety Assessment	Safety Design Criteria/Guidelines
SFR	Completed	Completed	SDC-Completed SDG-Ongoing
VHTR	Completed	Completed	IAEA-CRP for HTGR SDC as starting point
LFR	Completed	Pending SSC update	SDC pending SSC update
SCWR	Completed	Completed	SDC development recently initiated
GFR	Completed	Pending SSC update per RSWG review	SDC pending RSWG review
MSR	Pending SSC update	Planned by SSC	RSWG proposed a new “MSR safety approach” task force

Note: CRP = co-ordinated research project.

To address the lessons learnt from the accident at TEPCO’s Fukushima Daiichi nuclear power plant, the RSWG is working to update the 2008 basis safety document to revise how those lessons can best shape our approach to assessing and ensuring the safety of Gen IV systems and to anticipate the challenges for Gen IV systems during extreme external hazards and common cause failures. The group will endeavour to develop a full-scope application of ISAM, with probabilistic safety assessment (PSA) fully integrated in the evaluation of defence in depth, and associated with the development of best practices for the use of concepts such as practical elimination or the definition of “severe accident” for the six GIF systems.

Looking ahead, future work of the RSWG will help support the Gen IV systems by strengthening the defence indepth and robustness of the safety demonstration, to further advance them towards the demonstration phase. In the advanced stages of design maturity, ISAM can be used to assist designers in measuring the level of safety and risk associated with a given design, providing a clear and exhaustive representation of the safety architecture as an essential input for assessment against safety objectives or licensing criteria. This application of ISAM will in particular be useful for decision makers and regulators in assessing how much progress GIF has achieved towards the deployment stage of a Gen IV system. Additionally, ISAM can help identify remaining impediments to this next stage and what needs to be done to overcome them.

The RSWG will also engage in the elaboration of the SDC and SDG for individual Gen IV systems, as well ensure their consistency across systems. The RSWG will continue to interact with the SFR SDC task force to assist in the completion of the phase two SDC activity to finalise the development of the SDG for SFR. In addition to the LFR and GFR SDC reviews, further steps will include engaging in an ongoing IAEA co-ordinated research project on the development of SDC for the VHTR system.

As more interaction with national/international organisations (GSAR) is expected in the coming years (e.g. for the review of SDC/SDG), the RSWG will continue to support the relations between the GIF community and these organisations.

Another important aspect is the interaction of the RSWG with the Proliferation Resistance and Physical Protection Working Group (PRPPWG) to ensure a mutual understanding of safety/security priorities and their implementation in PRPP and RSWG evaluation methodologies. After the last joint RSWG/PRPPWG meeting in November 2015, a common activity was launched at the interface between safety and security. Similarly, it also seems critical to strengthen collaboration with the EMWG to assess and quantify the economic implications of the different safety choices identified in the design process.

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4.3. Proliferation resistance and physical protection

The GIF *Technology Roadmap* defined proliferation resistance and physical protection (PR&PP) as one of the four goals to advance nuclear energy into its next, “fourth” generation. It recommended the development of a methodology to define measures for PR&PP and to evaluate them for the six nuclear energy systems. Accordingly, GIF formed a PR&PP Working Group (PRPPWG) to develop a methodology and define the parameters against which PR&PP would be measured.

The current version of this methodology, Revision 6, became available on the GIF website in 2011 [1]. The methodology was developed with the aid of a series of limited and focused studies. The studies were performed using an example fast neutron reactor, nuclear energy system, called the Example Sodium Fast Reactor, consisting of four sodium-cooled fast reactors of medium size, co-located with an on-site dry fuel storage facility and a pyrochemical used-fuel reprocessing facility. The report on this work is also available on the GIF website [2].

Towards the end of the past decade, the PRPPWG and representatives of the GIF SSCs for each of the six GIF design concepts began a joint project on the PR&PP aspects of the six designs.

As a result of a series of workshops and associated interactions, white papers on the PR&PP state of play for each of the six design concepts were developed jointly by the SSCs and the PRPPWG. A compendium report that includes these six white papers and a discussion of cross-cutting topics was prepared and also became available on the GIF website in 2011 [3].

As part of efforts to familiarise GIF system researchers and programme policy makers with the PR&PP methodology, the PRPPWG has conducted periodic workshops for reactor designers and students. Useful, mutual information exchanges occurred during these workshops, which helped to further focus the methodological approach. The presentations and material prepared for the most recent workshops (Berkeley-US, 2015, Jeju-RoK, 2016) are openly available on the GIF website. Moreover, the PRPPWG has supported special training on its methodology as a part of larger GIF initiatives to provide training on all of the Gen IV methodologies, most recently in December 2017.

PRPPWG members prepared papers and participated as panellists at several international meetings and conferences over the past several years to describe the methodology and its relevance to non-proliferation and security of advanced nuclear energy systems. There were also applications of the methodology outside of the GIF purview by PRPPWG and other practitioners. In 2012, an entire special issue of the *Nuclear Technology* scientific journal prominently featured peer reviewed papers on the methodology and its applications [4].

The PRPPWG also prepared a set of frequently asked questions (FAQs) for those less familiar with the objectives, scope and capabilities of the PR&PP methodology and applications, and is available on the GIF website. Additionally, an extensive bibliography of papers and reports on the PR&PP subject area has been produced and regularly maintained on the GIF website [5].

The PRPPWG has co-ordinated closely with the IAEA since its inception; there has always been an IAEA representative in the PRPPWG who has contributed to the work and direction of the group. Considerable interaction has taken place between GIF and the IAEA's INPRO programme on methodology development, beginning with a comparison of the respective proliferation resistance methodologies, with the aim of understanding how prospective users could benefit from each other's application of the approaches, or from joint application. The IAEA has promoted an effort to consider international safeguards in early design phases according to the safeguards-by-design concept, and the resulting technical reports refer to the GIF PR&PP methodology. The exchange of information with the IAEA will continue to be essential in the future.

The GIF PR&PP evaluation methodology was motivated by the need to have an approach to the assessment of new nuclear energy design concepts envisioned within the GIF programme.

The methodology has been developed to assist GIF designers from the early phases of design to incorporate PR&PP concepts and assess performance against the GIF PR&PP goal. Proliferation resistance depends on the intrinsic design features and on the application of an extrinsic safeguard approach. In a similar way, physical protection robustness is the result of design features and of the application of a physical protection system. The methodology now enjoys wide international consensus and has been used in applications beyond the initial purpose, including applications outside of GIF, and lessons learnt from those applications can also serve to improve the methodology within the GIF arena.

It is essential to the GIF goal of PR&PP robustness that PR&PP concepts can be incorporated into the evolution of all six Gen IV systems. To support this goal, in 2011 the PRPPWG completed an initial joint study with the SSCs on the status of PR&PP implementation across the six designs. Because major progress has occurred in all of the designs since then, the PRPPWG is now working with the SSCs to update these earlier studies. The PRPPWG conducted a workshop with SSCs in 2017 to plan for future mutual activities, and will work to develop and issue updated white papers.

Looking ahead to the next decade, most Gen IV systems will reach the performance phase, and one or more will enter their demonstration phase. In the performance phase, a safeguards strategy and a physical protection strategy will have to be identified, with an estimate of the related cost for extrinsic features. In the demonstration phase, the full approaches will have to be developed.

The primary emphasis of the PRPPWG activity will be to support such progress by developing a sustained and structured interaction between the SSCs and PR&PP experts. The PRPPWG will concentrate its future R&D activities on five broad goals, here substantiated with an indication of possible aspects that might have to be investigated:

To capture the salient features of the design concepts that impact their PR&PP performance. As the design progresses, the room for major design modifications dictated by PR&PP requirements will shrink considerably. The PRPPWG will liaise with the SSCs to support their need to take the PR&PP of the system's design into account. With time, the analysis might shift its focus from design modification suggestions to highlighting the safeguard challenges that the designs entail, so that the safeguards community can be better informed on future safeguards R&D needs. This might require the development of a different or complementary approach to the analysis of the GIF design. Collaboration with the IAEA will be essential.

To facilitate PR&PP cross-cutting studies of relevance for several of the Gen IV systems. While every GIF design concept has unique design characteristics and peculiarities, there are aspects that are common to two or more design concepts. It is important to be able to address them in an effective, efficient and consistent way. Areas where ad hoc methodological R&D might be needed could include the analysis of a) the Gen IV systems' fuels with potential similarities, and b) the common nuclear fuel cycle front-end and back-end steps of the fuel cycles with which they will operate.

To identify insights for enhancing PR&PP characteristics of future nuclear energy systems. The PRPPWG foresees the in-depth application of the "PR&PP-by-design" concept to at least one of the SSC designs. This would be done co-operatively between the PRPPWG and the particular designers.

The SSCs expressed the wish that PR&PP design guidelines might be developed to help system designers incorporate PR&PP features into the design from the early design stages. Collaboration with the IAEA will also be essential in this respect.

PR&PP aspects are intertwined with many other aspects related to the other three GIF goals, in which all of the design concepts will have to excel. An area where important synergies could be exploited is the interface between safety and security. Together with the RSWG, a potential R&D area is to further investigate this interface and propose methods to correctly address it.

To foster the implementation of a PR&PP culture into the earliest phases of design. To meet this challenge, it is important to make sure that PR&PP is addressed in the right way at the right stage of each of the system's conception, design and construction. Not all PR&PP aspects are to be taken into account at the same moment or within the same design stage, and as the design matures the focus will have to move from one set of aspects to another. There might be a need to further investigate what aspects are to be addressed at which design stage, and how to do it in order to maximise effectiveness and synergy with the rest of the design activities.

To keep cognisance of, and to benefit from, PR&PP activities outside GIF. There are several international initiatives outside GIF in PR&PP areas. It is important that the PRPPWG continues to maintain cognisance and, where possible, interaction with these activities and, when needed, include and adapt relevant findings and methods to make them useful for GIF designers.

The PRPPWG expects that all these activities will lead to a refinement of the PR&PP evaluation methodology and its application. This will streamline and focus the approach to PR&PP aspects to address issues of interest to GIF, and thus enhance decision making in the GIF programme.

Last but not least, PRPPWG will continue to seek opportunities to work on cross-cutting issues within GIF that will enable sound and robust designs.

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Chapter 5. Generation IV International Forum initiatives in support of R&D on Generation IV systems

5.1. Education and Training Task Force

In 2015, the Generation IV International Forum (GIF) established the Education and Training Task Force (ETTF) to respond to the challenge of maintaining a well-educated advanced reactor systems workforce and to meet the projected growth in this field. The GIF-ETTF serves as a platform to enhance open education and training as well as communication and networking in support of GIF. The principal objective is the promotion of education and training by a) identifying and advertising current training courses, b) identifying and engaging in collaboration with other international education and training (E&T) organisations, c) developing a webinar series dedicated to Generation IV (Gen IV) systems and related cross-cutting topics, and advertising these at the national and international level, and d) creating and maintaining a modern social media platform (such as LinkedIn www.linkedin.com/groups/8416234) to exchange information and ideas on Gen IV R&D topics, as well as related GIF E&T activities.

The development of webinars is intended to stimulate the interest of, and to inform, not only young scientists but also managers, key decision makers and the general public on the advantages of innovative reactors and the related key R&D topics. These free, live and interactive webinars are offered by GIF as part of an initiative to fill the expanding need for nuclear engineers in the workforce. Each presentation is recorded and archived as an online resource, contributing to ensuring that there are future generations of nuclear engineers specialised in advanced reactor systems. The series of webinars that has been held is presented in Table 6, and can be viewed at www.gen-4.org.

Other webinars planned for the future, beyond the first series, concern the following subjects:

- economics of the nuclear fuel cycle;
- the ASTRID sodium-cooled fast reactor (SFR);
- design, safety features and progress of the High-Temperature Reactor–Pebble bed Module (HTR-PM);
- Gen IV materials and their challenges;
- Russian BN 600 and BN 800 reactors;
- the safety of Gen IV reactors;
- the Russian MOSART fast-spectrum molten salt reactor concept;
- proliferation resistance of Gen IV reactor systems.

By exploiting modern internet technologies, GIF will continue reaching out to a broad audience, raising the interest and strengthening the knowledge of participants in topics related to advanced reactor systems and advanced nuclear fuel cycles. In addition to opening the classroom to everyone in the world, the webinars offer earlier opportunities for interdisciplinary networking and E&T collaboration. Collaborative synergy has been developed with other international education and training organisations, such as the European Nuclear Education Network (ENEN), to promote information exchange on various training courses, academic education, opportunities for jobs and scholarships, and educational materials.

Table 6: **GIF webinar series (September 2016 to March 2019)**

Atoms for Peace – The Next Generation **Dr John Kelly**	Closing the Fuel Cycle **Prof. Myung Seung Yang**	Introduction to Nuclear Reactor Design **Dr Claude Renault**
Sodium-Cooled Fast Reactors (SFR) **Dr Bob Hill**	Very High Temperature Reactors (VHTR) **Dr Carl Sink**	Gas-Cooled Fast Reactor (GFR) **Dr Alfredo Vasile**
Supercritical Water Reactors (SCWR) **Dr Laurence Leung**	Fluoride-Cooled High-Temperature Reactors (FHR) **Prof. Per Peterson**	Molten Salt Reactors (MSR) **Prof. Elsa Merle**
Lead Fast Reactor (LFR) **Prof. Craig Smith**	General Considerations on Thorium as a Nuclear Fuel **Dr Franco Michel-Sendis**	Metallic fuels for SFRs **Dr Steven Hayes**
Energy Conversion **Dr Richard Stainsby**	Estimating Costs of Gen IV Systems **Dr Goeffrey Rothwell**	Phénix and Superphénix Feedback Experience **Mr Joël Guidez**
Sustainability of Relevant Framework for addressing Gen IV Nuclear Fuel Systems **Dr Christophe Poinssot**	Design, Safety Features and Progress of the HTR-PM **Prof. Dr Yujie Dong**	Materials Challenges for Generation IV Reactors **Dr Stu Maloy**
SCK•CEN's R&D on MYRRHA **Prof. Hamid Aït Abderrahim**	Proliferation Resistance of Gen IV Systems **Dr Robert Bari**	Molten Salt Actinide Recycler and Transforming System with and Without Th-U support: MOSART **Dr Victor Ignatiev**
Astrid, Lessons Learned **Mr Gilles Rodriguez**	Advanced Lead Fast Reactor European Demonstrator, ALFRED Project **Dr Alessandro Alemberti**	Russia BN 600 and BN 800 **Mr Ilya Pakhomov**
Scientific and Technical Problems of Closed Nuclear Fuel Cycle in Two-Component Nuclear Energetics **Dr Alexander Orlov**	Safety of Gen IV Reactors **Dr Luca Ammirabile**	The ALLEGRO Experimental Gas-Cooled Fast Reactor Project **Dr Ladislav Belovsky**

After completion of the successful GIF webinar series at the beginning of 2019, several initiatives are envisaged. The GIF webinar series targeted non-scientists or non-nuclear scientists, including policy makers, and more generally the civil society, but the content of some webinars has definitely addressed topics that could be understood by a scientific audience. The creation of a massive open online course (MOOC) will enable non-scientific audiences or college students around the world to take university courses online. The MOOC will be offered for a period of a month or so and will be free of charge. Short videos will be offered weekly to develop familiarity with nuclear energy or deepen existing knowledge on the subject, and will include the place of nuclear energy in a future economy, or innovation in nuclear energy. Since YouTube has the ability to reach a very wide general audience and is the most attractive media format for younger people, setting up a YouTube channel is also under consideration, where videos could be watched and where people could also subscribe to watch future materials.

A website has recently been created (www.gen-4.org/gif/jcms/c_97306/education-and-training), and will allow more efficient advertising of international courses and allow lectures to be shared with/posted to GIF member countries. This is a good tool for co-ordination of E&T events (courses, workshops, summer schools, etc.) dedicated to Gen IV topics between GIF and other organisations (Euratom, the bodies listed in the paragraph below, national initiatives in member states, etc.).

Collaboration on E&T has started with the European Nuclear Education Network (ENEN). Collaboration will also be extended to other international organisations, including the African Network for Education in Science and Technology (AFRA-NEST), the Asian Network for Education in Nuclear Technology (ANENT), the Latin American Network for Education in Nuclear Technology (LANENT), the International Atomic Energy Agency (IAEA) and the American Nuclear Society (ANS), to promote, manage and preserve nuclear knowledge, and to ensure the continued availability of a talented and qualified workforce for the sustainability of nuclear technologies.

The organisation of a PhD Gen IV event, within the frame of ENEN PhD Event, is also being considered. The awarded researchers would be granted attendance at an international conference with the support of the ENEN Association and GIF.

GIF may organise an international framework for the co-ordination of projects from students and postdoctoral researchers on the development of an advanced nuclear reactor.

Finally, a book summarising the webinars presented over two years with questions and answers will be compiled.

5.2. Regulatory challenges

Over the last several years, GIF has been engaged in several activities related to safety and the regulatory framework for Gen IV nuclear energy systems. The goals of these efforts were to harmonise safety requirements among GIF members and then engage with the regulatory community to lay the foundation for future licensing of Gen IV systems. The efforts included promoting the external review of the safety design criteria (SDC) and safety design guidelines (SDG) for the sodium-cooled fast reactor (SFR), leading the engagement with the NEA Ad hoc Group on the Safety of Advanced Reactors (GSAR),[1] extending the SDC/SDG development to other Gen IV systems, engaging with System Steering Committees (SSCs) on safety research priorities, and keeping the Policy Group informed of these safety research priorities.

A GIF task force developed SDC and SDG for the SFR. The SFR SDC report was distributed for external review to national regulators and international organisations, such as the Multinational Design Evaluation Programme, the OECD/NEA and the IAEA. Several workshops were held at the IAEA with reactor designers, regulators and safety experts to review the SDC/SDG for SFR.

GIF representatives attended GSAR meetings in 2015 and 2016. At the September 2016 meeting, a significant part of the agenda was devoted to a “deep dive” into the safety aspects of SFR. Future GSAR meetings are anticipated to include “deep dives” into other GIF systems. The LFR preliminary SSC has an initiative to develop SDC/SDG for the LFR. For the high-temperature gas-cooled reactor (HTGR), GIF is working with the IAEA using the Coordinated Research Project approach to develop SDC. Progress on these efforts will be reviewed by the Risk and Safety Working Group (RSWG) to ensure consistency with the work that has been done on the SFR.

Looking to the future, GIF expects to continue its work on safety and regulatory frameworks. Engagement with regulators and technical support organisations will also continue, and the regulators are expected to begin providing guidance to Gen IV system developers on regulatory requirements in the not-too-distant future. For example, the US Nuclear Regulatory Commission recently issued a draft Regulatory Guide on the General Design Criteria for non-water-cooled reactors. This Regulatory Guide was published as Reg. Guide 1.232 Rev.0 in April 2018. Continuing this dialogue between GIF and the regulators will benefit not only GIF system developers, but also the regulators and their technical support organisations.

5.3. Market opportunities

Almost two decades have passed since the establishment of GIF in 2000. During this period, various energy reforms, affecting energy sources, policies and markets, have been implemented worldwide. There is no doubt, even at this time, that nuclear power is playing an important role in many countries as a power source that realises a stable supply of energy while curbing greenhouse gas emissions and thus mitigating anthropogenic impact on the planet’s climate.

1. In December 2014 at the Nuclear Energy Agency (OECD/NEA), GIF presented the SDC for SFR to a meeting of the OECD/NEA Committee on Nuclear Regulatory Activities (CNRA) and Committee on the Safety of Nuclear Installations (CSNI). The outcome was the initiation of the NEA GSAR ad hoc group, which has been tasked with co-ordinating international discussions on regulatory topics for Gen IV reactors.

Under such circumstances, the need for Gen IV nuclear systems with a high level of sustainability, safety and reliability, economic viability, proliferation resistance and physical protection is increasing. Accordingly, in recent years not only national governments, but also private companies have embarked on the development of such systems.

To allow the Gen IV systems to play an important role in future energy markets, it is essential to identify challenges, including environmental, to satisfying the various market needs, and to find solutions to such challenges. Certain challenges may differ between countries, but there will also be common challenges. Paving the way towards finding solutions based on multilateral co-operation will thus be an important role for GIF in the next decade. In collaboration with the Senior Industry Advisory Panel (SIAP), the Economics Modelling Working Group (EMWG) and others, the Vice-Chair for Market Issues has started an initiative addressing these issues.

The SIAP put forward two important recommendations: a) identify the attributes of Gen IV systems most attractive to industry (vendor/utility); and b) investigate market conditions and timelines for commercialisation of Gen IV reactors. Accordingly, the scope of the initiative led by the Vice-Chair for Market Issues is as follows:

- to develop a better understanding of the drivers, opportunities and constraints related to the market environment, to provide information on appropriate ways to carry out GIF activities;
- to work closely with the SIAP, SSC chairs and related task forces in carrying out its work, and provide recommendations regarding the role and value of Gen IV systems in future market environments.

The activities of the initiative will be in the form of surveys, economic evaluations, analyses of marketing issues and development of end-use options. Consideration will also be given to the development of deployment scenarios for Gen IV systems and documents on corresponding utility/end-user requirements.

Phase 1: Conduct survey of key points on market issues

The following preliminary issues were identified based on discussions with the SIAP and EMWG: a) national and international market drivers; b) market-related opportunities (e.g. small modular reactors [SMRs], integration of renewables, non-electric applications to replace fossil fuel-based heat production); c) market-related constraints; d) analysis of the key issues in political decision making with regard to the energy mix and the role of advanced reactors in each country (e.g. international agreement on the 2°C climate change scenario, energy security).

Phase 2: Build evaluation indexes to explain Gen IV systems' attractiveness in terms of the market drivers

Three aspects of the evaluation index were retained: a) evaluate the economics of Gen IV reactors, taking into account other merits based on sustainability; b) increase the opportunity for advanced nuclear reactors; and c) reduce the constraints linked with the deployment of advanced nuclear reactors.

Phase 3: Understand and value the attributes of Gen IV systems

Understand and value the attributes of Gen IV systems for stakeholders, addressing the market-related issues, funding of Gen IV systems and risk.

The GIF activities on market issues have just began. In a discussion of the Phase 1 survey, policy issues were identified as one of the key points. A detailed analysis on this point is under way. It is expected that such discussion and understanding of the attractive attributes of Gen IV systems will contribute to the development of Gen IV systems in the coming decades and also to their faster deployment in worldwide markets.

5.4. Senior Industrial Advisory Panel

The SIAP, in support of the GIF Vice-Chair for Market Issues, and in interaction with the EMWG, has suggested priority R&D areas to foster the chances of Gen IV systems complying with (future) market conditions. Using a broad reading of this notion of attractiveness to the "market", the SIAP started from the concept of social sustainability in its wider understanding: environment, economics, reliability of energy supply. From there, it proposed more precise areas for targeted R&D.

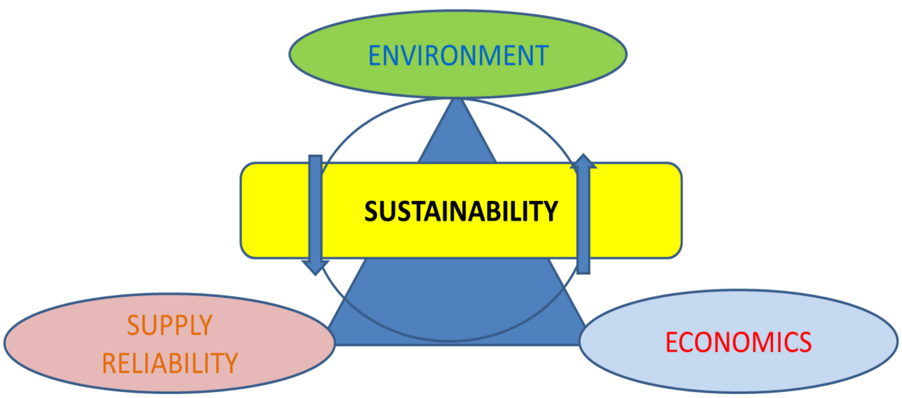
From general concepts…

The overall energy framework has evolved since the formation of GIF in 2000. A number of countries are now operating under liberalised electricity market conditions or are in transition towards such market operation. In addition, many countries are committing to decarbonise their energy mix.

The SIAP agreed on three main attributes necessary for Gen IV systems to compete in the "market": to be economic, to be publicly accepted, and to be able to be integrated in the energy mix. The time perspective is a readiness for commercial fleet deployment by around 2045 (for the first systems). Industry is expecting to have viable "options" available in this time frame, and timely R&D and further industrial-type demonstration phases should make this possible.

The attributes indicated above may be put in perspective and embedded into a wider concept of social sustainability (Figure 14) represented by the triangle of three pillars of energy policy: environment, supply reliability and economics (in whatever order). Sustainability sits inside the triangle, and results from the combination and balancing of the pros and cons of each form of energy relative to the three pillars. Challenges (issues/concerns/stakes) can be derived for these three pillars, and R&D objectives and further programmes might be defined to tackle them.

Figure 14: **Energy policy triangle**



Public acceptance should be the result of the combination and balancing of the respective answers to the challenges associated with the three pillars. It is subjective, depending much on the quality (i.e. transparency to the maximum possible extent) of the information provided, on the credibility of the information conveyer, and on the readiness to be informed, but it is also much influenced by the relative importance (weight) given by individuals to the three pillars.

… towards more concrete actions

The aforementioned three attributes that, from the industry perspective, would make Gen IV systems attractive are interrelated.[2] Indeed, public acceptance relates to safety, security and waste management (i.e. closed fuel cycle), which have a cost. Public acceptance is also connected to the affordability of energy and so with the economics of the energy system. Economics is wider than the cost aspect in liberalised energy markets, where the driver is the price decoupled from the costs (generation, grid and system costs). This points to the need for future nuclear systems to be properly integrated in a global "economic/affordable" energy mix, considering a global life cycle analysis and the environmental footprint of the diverse sources of energy.

From these three attributes, the SIAP suggests the following "challenges/opportunities" for Gen IV systems seeking commercial fleet deployment in 2045:

1. **Guarantee a very low full-lifecycle environmental footprint** (CO_2, SO_x, NO_x, water and land usage and pollution) during normal functioning of the system (plant and associated fuel cycle). For example, by minimising land impacts, cooling requirements (water reliance) and waste generation during operation and decommissioning.
2. **Exclude severe accident/core melt or ensure no off-site radioactive release** in case of severe accident/core melt, through reactor concept-dependent prevention measures, e.g. low power density fuel, accident-tolerant fuel and systems, high core thermal inertia, resistance to black out, smaller power ratings (SMRs), etc. and through mitigation measures, e.g. in-vessel/ex-vessel corium cooling, in-containment management, etc.
3. **Reduce the amount and lifetime of the ultimate high-level radioactive waste**, by developing, demonstrating and quantifying improvements in high-level waste management and by addressing the potential for partitioning and transmutation of transuranic elements. Diverse routes to be investigated include multi-recycling in a fleet of fast neutron reactors, or with dedicated transuranic burners.
4. **Ensure Gen IV nuclear systems are sufficiently flexible**, at minimal cost, to be integrated in electricity systems with increasing shares of variable/intermittent renewable energy sources (RES) (for example 50%), using diverse possible options: load following, remote control, modularity (SMRs), cogeneration and hybrid systems.
5. **Reduce the costs** of investment (overnight capital cost), reduce and master the duration of construction (financing cost), optimise the costs of licensing, operation and maintenance (O&M), the fuel cycle and waste management, and even optimise the decommissioning costs as early as at the design stage in order to be competitive in the market with other sources of energy.

The second and third challenges/opportunities are the two key concerns of the public with mention of the word "nuclear." Gen IV systems need to demonstrate real progress on these two areas. In addition, the "residual nuclear accident risk" needs to be put into perspective in comparison to other accident risks (energy and beyond). It may become more of a communication issue than a technical issue, but it deserves attention.

The fourth challenge/opportunity is the main change compared to the start of GIF in 2000 when the question was focused on Gen IV versus Gen III, while still assuming reactors would operate in baseload. The figure 50% variable/intermittent RES might be considered as challenging but realistic. An assumption may need to be taken on the contribution of CCS/U (carbon capture and storage/usage) and large-scale energy storage capacity. In addition, applications beyond pure electricity production have to be considered.

The fifth challenge/opportunity focuses on the cost aspects of Gen IV systems. The unknowns and uncertainties in electricity (and possible future energy) market design and

2. In view of commercial fleet deployment starting in 2045, which would mean FOAK (first of a kind) between 2037 and 2040, and "pre-FOAK", or demonstrators, between 2030 and 2035.

operation make it difficult to go beyond the pure cost dimension. The maximum therefore has to be done to reduce all elements of Gen IV costs, including the cost of licensing in each country.

Evaluating the wider aspects of competitiveness, in comparison with the cost of RES and other low-carbon dispatchable sources such as CCS/U and large-scale storage in a full-cost approach, would nevertheless be useful. It would require making necessary assumptions linked to the evolution of the market design and operation, which have, in particular, an impact on the system costs.[3]

For each of the challenges/opportunities, it may be relevant to further define areas for specific R&D/innovation actions, which should then be integrated into Gen IV system activities, seeking to minimise related costs by streamlining, increasing efficiency, and expanding co-operation.

5.5. Research infrastructure needs: From R&D to demonstration and deployment

Today's research infrastructure includes major scientific equipment, scientific collections, structured information and ICT-based infrastructure. They are single-sited or distributed throughout several countries. GIF member countries are faced with a wide spectrum of issues related to infrastructure, many of which are globally unique and regionally distributed. Many stakeholders are involved, from ministries to researchers and industry, with an underlying and growing use of e-infrastructure. They present opportunities for, and yet difficulties in, interactions between basic research and industry. Public and private funding appears always to be lacking, and single countries do not have the critical mass or the dimensions to implement large research infrastructure. There is a real need to co-operate on a broad international level.

Substantial research, development and demonstration (RD&D) of systems' conceptual/detailed design and analysis are needed. Refurbishment and/or construction of research infrastructure and facilities are increasingly complex and costly.

An opportunity exists, by identifying the latest R&D needs and the mapping of infrastructure, to plan for the shared use of existing facilities and to undertake the development of others. The most important priorities are in the areas of fuel cycle, fuel and material irradiation, reactor safety, dedicated loops, mock-ups and test facilities, advanced simulation and validation tools, transnational access to infrastructures, and the E&T and knowledge management of scientists and engineers.

GIF members strongly support a co-ordinated revitalisation of nuclear RD&D infrastructure worldwide, to a level that would once again quickly move forward a new generation of reactors.

5.6. Advanced materials engineering and manufacturing

Reactor structures and components in Gen IV reactors are required to operate under more onerous and exacting conditions than light water reactors. Reactor operating temperatures are higher for all systems, while greater cumulative lifetime neutron fluences and comparatively hostile corrosion environments are also common in typical Gen IV designs.

Operating Gen IV reactors will most probably require the successful use of both traditional structural nuclear materials and improved material designs resulting from recent advances in materials science. Furthermore, they are likely to utilise modern advanced manufacturing techniques, where they can reduce cost or speed up construction time. However, most nuclear design codes use design-by-rule philosophies that typically dictate that only qualified materials can be used. Qualification of new materials or new manufacturing processes can be a long and difficult process, and the long lead times involved produce an effective and consequent barrier to market entry of new or optimised materials and processes at an industrial scale.

[3] NEA (2019), *The Costs of Decarbonisation: System Costs with High Levels of Nuclear and Renewables*, OECD Publishing, Paris.

These issues can generate perceived complications surrounding the risks and benefits of intellectual property in new materials development that directly mitigate against investment in industrial-scale processes.

It is clear, however, that a new material or manufacturing process is unlikely to be used in a safety-critical application in the nuclear industry without fully characterised material behaviour and published and qualified design data.

Collectively, these issues present a barrier to market entry for Gen IV reactors and the development of materials and manufacturing solutions to benefit the six Gen IV reactor systems.

These considerations have led the GIF Policy Group to launch an investigation among GIF country research institutions and nuclear companies to assess the level of interest in cross-cutting activities in support of advanced materials and innovative manufacturing development to a high technology readiness level. The assessment will address the merits and difficulties of collaboration on this topic, develop a priority list of R&D areas and respective activities, and produce recommendations on the value of a possible GIF task force to cover this cross-cutting area.

Chapter 6. Conclusions and key messages

Nuclear energy is a safe, reliable and long-term source of low-carbon energy, and the Generation IV International Forum (GIF) considers it an essential part of any future sustainable energy mix.

Generation IV (Gen IV) concepts feature extended capabilities beyond those of light water reactors (LWRs) and complement existing and evolutionary Gen III/III+ reactors – to be deployed up to the end of the century – by providing additional options and applications such as:

- optimisation of resource utilisation;
- multi-recycling of fissile materials/used fuel and reduction the footprint of geological repositories for high-level waste;
- low-carbon heat supply for cogeneration and high-temperature industrial applications (e.g. process steam, synthetic fuels, hydrogen production);
- enhanced integration of nuclear and other low-carbon sources.

The Gen IV systems were selected on the basis of four, main criteria: sustainability, economics, safety and reliability, and proliferation resistance and physical protection (PR&PP). These systems are facing new challenges in the current and future contexts (e.g. deregulated markets, competition and symbiosis with intermittent renewable energy sources, novel security threats). Flexibility in the form of manoeuvrability, versatility of energy products or progressive deployment (e.g. small modular reactors [SMRs]) are key for facilitating the integration of Gen IV nuclear systems into a generating fleet that includes a large share of intermittent renewable energy sources.

Additional challenges create new R&D needs that GIF is progressively addressing in its framework of multinational collaboration, in particular by researching and designing reactor systems that will displace fossil fuels in the most efficient manner, not only for power generation but also for house heating, process heat supply to industry and carbon-neutral synthetic hydrocarbon production for transport fuel. By maximising the benefits of nuclear energy for a low-carbon future, GIF activities are encouraging worldwide acceptance of nuclear energy.

Initially, GIF activities of non-R&D experts focused on defining and implementing methodologies for refining and quantifying GIF systems' assessment criteria for economics, safety and reliability and PR&PP. These activities have led to a common understanding of the lessons learnt from the nuclear accident at the Fukushima Daiichi nuclear power plant by reinforcing the defence-in-depth approach against external events and promoting the robustness of the safety demonstration.

Current R&D work focuses on enhancing the safety characteristics of Gen IV systems, so as to exclude radioactive releases to the environment in case of accidents, eliminate therefore the need for emergency measures and minimise the impact on populations. GIF is reaching out to and involving regulatory bodies, transmission system operators and international agencies (e.g. the Nuclear Energy Agency [NEA] Working Group on the Safety of Advanced Reactors [WGSAR] and the International Atomic Energy Agency [IAEA]) in its work on safety objectives, safety design criteria (SDC) and safety design guidelines (SDG).

GIF, non-R&D activities have also successfully extended into the field of education and training (E&T), with about 20 webinars targeting students and academia. More recently, efforts to strengthen cross-cutting R&D within the GIF collaboration framework have resulted in additional initiatives aimed at shaping common views on topics related to nuclear market challenges and opportunities, and to research infrastructure use and planning. These new activities, as well as those related to safety and regulatory issues, are conducted under the responsibility of GIF Vice-Chairs with the active involvement of the System Steering Committees (SSCs), provisional System Steering Committees (pSSCs), the Senior Industry Advisory Panel (SIAP) and the modelling working groups (the Economics Modelling Working Group [EMWG] and Risk, and the Safety Working Group [RSWG] in particular).

These new, GIF activities aim to take full advantage of the expertise gathered within GIF, anticipating a framework – along with its challenges and opportunities – in which Gen IV systems will most likely be deployed, while specifying the respective R&D needs, so as to ensure that the performance of GIF systems is adapted to the realities of the framework. These new activities contribute to extending GIF outreach beyond the partners in R&D, industry, regulatory bodies, the IAEA, the OECD/NEA and academia.

Safety design criteria (SDC) and safety design guidelines (SDG), first developed for the sodium-cooled fast reactor (SFR) systems, are being extended to other systems. White papers on the safety of Gen IV systems have been published and will be regularly updated, along with the GIF Safety Basis Report. The Integrated Safety Assessment Methodology (ISAM) is being developed as a technology-neutral toolkit to evaluate the safety characteristics of all Gen IV systems. White papers on the PR&PP aspects of all Gen IV systems have been prepared and published in a report that is openly available. These white papers will be updated as needed. The PR&PP evaluation methodology is available for use by designers for more detailed assessment of Gen IV systems.

GIF is also updating its vision, i.e. challenges and goals, and the respective R&D needs for the next 10-15 years, while accounting for both new build (e.g. BN-800, Prototype Fast Breeder Reactor [PFBR], High-Temperature Reactor–Pebble bed Module [HTR-PM]), and near-term projects (e.g. BN-1200, BREST-OD-300, MOSART). The key R&D challenges in the coming 10-15 years have been identified for the six GIF systems as follows:

For the sodium-cooled fast reactor (SFR):

- advanced non-minor actinide-bearing, minor actinide-bearing, and high-burnup fuel evaluation, optimisation and demonstration (a cross-cutting challenge for all fast neutron spectrum systems);
- development of innovative in-service inspection and repair (ISIR) technologies;
- advanced energy conversion systems;
- development of leak-before-break (LBB) assessment procedures and instrumentation;
- development of steam generators, including investigations of sodium-water reactions and development of advanced inspection technologies for a Rankine-type steam generator;
- improved economics;
- validation of passive decay heat removal.

For the very high temperature reactor (VHTR):

- completion of fuel testing and qualification capability (including fabrication, quality assurance, irradiation, safety testing and post-irradiation examination [PIE]);
- qualification of graphite, hardening of graphite against air/water ingress, management of graphite waste;
- coupling technology and related components;
- establishment of design codes and standards for new materials and components;
- advanced manufacturing methods (cross-cutting challenge);
- cost reduction;

- licensing and siting;
- system integration with other energy carriers in hybrid energy systems;
- follow-up of HTR-PM demonstration tests, engaging and enhancing information exchange with start-ups, private investors, new national programmes;
- High-Temperature Engineering Test Reactor (HTTR) safety demonstration tests and coupling to hydrogen production plant.

For the lead-cooled fast reactor (LFR):

- phenomenology of the lead-water and lead-steam interactions;
- prevention and mitigation of sloshing;
- new corrosion-resistant materials (including surface modifications);
- operation and maintenance;
- fuel and fuel reprocessing (nitride, minor actinide-bearing and high burnup fuels);
- advanced modelling and simulation;
- design code and standards;
- severe accidents;
- progress towards the deployment of a Gen IV LFR (BREST-OD-300).

For the gas-cooled fast reactor (GFR):

- finalising the design and initiating the licensing process of the GFR experimental reactor ALLEGRO;
- qualification of the mixed oxide fuel adapted to the specific operating conditions of the ALLEGRO start-up core;
- development of dense fuel elements (i.e. material characterisation under normal and accidental conditions for fresh and irradiated fuel, qualification, and fabrication capacities) capable of withstanding very high temperature transients (i.e. carbide fuel with composite SiC and fibre-reinforced SiC clad);
- validation studies (codes and data): the need for specific experiments addressing innovative ceramic materials, as well as unique, GFR-specific, abnormal operating conditions, such as depressurisation and steam ingress;
- air and helium tests on subassembly mock-ups under the representative temperature and pressure conditions necessary to assess the heat transfer and pressure drop uncertainties for the specific GFR design;
- large-scale air and helium tests to demonstrate that the passive decay heat removal (DHR) function will be required for the ALLEGRO licensing process;
- GFR-specific components (e.g. blowers and turbo machines, thermal barriers) development and qualification.

For the supercritical water reactor (SCWR):

- development of cladding materials to withstand the high-pressure and high-temperature environment;
- establishment of a chemistry-control strategy to minimise water-radiolysis effect and activation-product transport;
- optimisation of the fuel assembly geometry and configuration to further enhance economic and safety characteristics;

- mitigation of these challenges by lowering the operating temperature of the coolant in SCWRs to reduce the peak cladding and fuel temperatures.

For the molten salt reactor (MSR):

- development of salt and material combinations (characterisation and qualification);
- development of integrated (physics and fuel chemistry) reactor performance modelling and safety assessment capabilities;
- demonstration of the MSR safety characteristics at laboratory level and beyond;
- establishment of a salt reactor infrastructure and economy that includes affordable and practical systems for the production, processing, transport and storage of radioactive salt constituents;
- development of the MSR licensing and safeguards framework;
- progress towards MSR demonstration (MOSART).

The key challenges for the Methodology Working Groups and for other GIF cross-cutting activities in the coming 10-15 years were also identified as follows.

For economic modelling:

- improve through further engagement with SSCs the cost estimates of Gen IV design, including a better understanding of cost uncertainties, opportunities for cost reduction, and assessments of the series effect from first-of-a-kind (FOAK) to the n^{th}-of-a-kind (NOAK) plants (learning curve through industrial deployment);
- investigate with the help of SSCs the merits and R&D challenges of each of the six Gen IV systems to meet flexibility needs of low-carbon energy systems and address new market opportunities looking at both load-following capabilities and new energy outputs.

For risk and safety:

- ensure that the systems' designs comply with defence-in-depth principles;
- provide the means to improve the robustness of the safety demonstration;
- develop a genuine risk-informed safety assessment approach with fully integrated probabilistic safety assessments in the evaluation of defence in depth.

For proliferation resistance and physical protection (PR&PP):

- adapt the PR&PP-by-design approach, integrating intrinsic and extrinsic features to achieve effective and efficient realisation of PR&PP robustness in synergy with other GIF goals;
- sustain close interaction with each of the SSCs in order to ensure that PR&PP-by-design is an essential feature of the six Gen IV systems.

For education and training:

- enhance communication on Gen IV systems through E&T initiatives, for example webinars, massive open online courses (MOOCs), specific events and activities for students and young researchers.

For advanced manufacturing and material engineering:

- identify and develop mechanisms whereby cross-cutting collaborative activities lead to the faster qualification of advanced manufacturing and materials engineering innovations, and thus reduce impediments to the deployment of Gen IV systems.

Some Gen IV systems will enter the demonstration phase in the next decade. To support these systems, GIF will ensure:

- best use of available experimental R&D infrastructures;

- support for the co-ordination of national programmes among GIF countries to avoid unnecessary duplication of facilities and ensure availability of both key experimental infrastructure and human resources;
- best use of (multi-scale/multi-physics) simulation, as well as verification, validation and qualification (VVQ) tools as a complement to experimental programmes designed to support the demonstration phase;
- best use of digital and product life management (PLM) tools to support the licensing phase and reduce time to market of innovative designs;
- support technical and methodological innovation to reduce the costs of investment (overnight capital cost), shorten and master the duration of construction (financing cost), and optimise the licensing costs, the operational costs (O&M), the fuel cycle and waste management costs and decommissioning costs (compared to existing nuclear power plants) at the design stage, in order to be competitive in the market;
- promote synergy between non-proliferation, physical protection and a robust safety design, thus strengthening a safety culture that aims for optimal integration of safety, security and safeguards for advanced fuel cycles and reactor concepts in an effort to achieve greater public awareness and acceptance.

In areas where common research and technology development needs have been identified (e.g. safety design methodologies, DHR systems, advanced fuels and materials, advanced manufacturing, modelling and simulation, and VVQ tools), cross-cutting R&D activities will be pursued collaboratively to address the above needs and advance the technology.

It is important to attract and educate a young generation of nuclear energy scientists and engineers. It is also essential to unleash and make the best use of the creativity of the next generation by promoting and offering strong motivations and expectations (smart society, smart energy mix), new ideas (advanced manufacturing, disruptive innovations, etc.), new ways of thinking, working and collaborating (the spirit of "start-up" companies), as well as skills coming from other fields (applied mathematics, multi-criteria optimisation, augmented reality, digital economy, etc.).

GIF will continue to engage with regulatory authorities and technical support organisations with the long-term goal of reaching a harmonisation of requirements and a better understanding of licensing application and costs.

List of abbreviations and acronyms

AFA	Alumina-forming austenitic
AFRA-NEST	African Network for Education in Science and Technology
ANENT	Asian Network for Education in Nuclear Technology
ANL	Argonne National Laboratory (United States)
ANS	American Nuclear Society
CCS/U	Carbon capture and storage/usage
CEA	Alternative Energy and Atomic Energy Commission (France)
CER	Ceramic
CFD	Computational fluid dynamics
CRP	Co-ordinated research project
DHC	Delayed hydride cracking
DHR	Decay heat removal
DR	Demonstration reactor
EERA–JPNM	Joint Programme on Nuclear Materials of the European Energy Research Alliance
ELFR	European Lead Fast Reactor
EMWG	Economics Modelling Working Group (GIF)
ENEN	European Nuclear Education Network
ESFR	European Sodium Fast Reactor
ESNII	European Sustainable Nuclear Industrial Initiative
ETTF	Education and Training Task Force (GIF)
E&T	Education and training
FA	Fuel assembly
FHR	Fluoride salt-cooled high-temperature reactor
FOAK	First of a kind
GACID	Global Actinide Cycle International Demonstration (GIF project)
Gen IV	Generation IV
GDI	Guidance Document for ISAM
GFR	Gas-cooled fast reactor
GIF	Generation IV International Forum
GSAR	Group on the Safety of Advanced Reactors (NEA)
GTHTR	Gas Turbine High-Temperature Reactor
HFIR	High Flux Isotope Reactor
HLM	Heavy liquid metals

HTGR	High-temperature gas-cooled reactor
HTR-PM	High-temperature reactor–pebble bed module
HTSE	High-temperature steam electrolysis
HTTR	High-temperature engineering test reactor
HX	Heat exchangers
IAEA	International Atomic Energy Agency
ICT	Information and communications technology
IEA	International Energy Agency
INL	Idaho National Laboratory (United States)
IRSN	Institute for Radiological Protection and Nuclear Safety
ISAM	Integrated Safety Assessment Methodology
ISIR	In-service inspection and repair
JRCMSD	Joint Research Centre Molten Salt Database (European Union)
JSFR	Japan Sodium Fast Reactor
KAERI	Korean Atomic Energy Research Institute
LANENT	Latin American Network for Education in Nuclear Technology
LBB	Leak before break
LBE	Lead-bismuth eutectic
LEU	Low-enriched uranium
LEU-UOX	Low-enriched uranium, uranium dioxide
LFR	Lead-cooled fast reactor
LWR	Light water reactor
MA	Minor actinides
MCC	Mining and Chemical Combine (Russian facility)
MDEP	Multinational Design Evaluation Programme
MOOC	Massive open online course
MOSART	Molten Salt Actinide Recycler and Transmuter (Russia)
MOX	Mixed oxide
MSFR	Molten salt fast reactor
MSR	Molten salt reactor
NNSA	National Nuclear Safety Administration (China)
NOAK	N^{th} of a kind
NRC	Nuclear Regulatory Commission (United States)
ODS	Oxide dispersion strengthened
NEA	Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory (United States)
O&M	Operation and maintenance
PIE	Post-irradiation examination
PLM	Product life management
PFBR	Prototype Fast Breeder Reactor

PSA	Probabilistic safety assessment
pSSC	Provision System Steering Committee (GIF)
PRPPWG	Proliferation Resistance and Physical Protection Working Group (GIF)
PR&PP	Proliferation resistance and physical protection
RD&D	Research, development and demonstration
RES	Renewable energy sources
RFECT	Remote field eddy current testing
RHP	Reactor high pressure
RLP	Reactor low pressure
RSWG	Risk and Safety Working Group (GIF)
R&D	Research and development
SC-HTGR	Steam cycle high-temperature gas-cooled reactor
sCO2	Supercritical carbon dioxide
SCC	Stress-corrosion cracking
SCWR	Supercritical water-cooled reactor
SDC	Safety design criteria
SDG	Safety design guidelines
SFR	Sodium-cooled fast reactor
SG	Steam generator
SIAP	Senior Industry Advisory Panel (GIF)
SiCf/SiC	Silicon carbide fibre-reinforced/silicon carbide
SiC/PyC	Silicon carbide/pyrolytic carbon
SMR	Small modular reactor
SSC	System Steering Committee (GIF)
SSTAR	Small secure transportable autonomous reactor
TFM	Transient fission matrix
THTR	Thorium Hochtemperatur Reaktor
TMSR	Thorium molten salt reactor
TRISO	Tristructural isotropic
VHTR	Very high temperature reactor
VVQ	Verification, validation and qualification

Units of measure

$GtCO_2$	Gigatonnes of carbon dioxide
GWd/tHM	Gigawatt days per tonne of heavy metal
kW	Kilowatt
kWe	Kilowatt electrical
MPa	Megapascal
MW	Megawatt

MWe	Megawatt electrical
MWth	Megawatt thermal
MW/m^3	Megawatts per cubic metre

Elements and compounds

Am	Americium
AmF_3	Trifluoroamericium
Be	Beryllium
BeF_4	Tetrafluoroberyllate
CeF_3	Trifluorocerium
Cm	Curium
Co	Cobalt
CO_2	Carbon dioxide
F	Fluorine
He	Helium
I-S	Iodine sulphur
KF	Potassium fluoride
Li	Lithium
LiF	Lithium fluoride
NaF	Sodium fluoride
Nb	Niobium
Np	Neptunium
PuC	Plutonium carbide
PuF_3	Plutonium fluoride
SiC	Silicon carbide
Th	Thorium
ThF_4	Thorium tetrafluoride
U	Uranium
UC	Uranium carbide
UCO	Uranium oxycarbide
UF_4	Uranium tetrafluoride
UO_2	Uranium dioxide
UPuC	Uranium plutonium carbide
ZrC	Zirconium carbide

This 2019 update to Generation IV International Forum (GIF) R&D Outlook for Generation IV Nuclear Energy Systems provides an overview of recent achievements in relation to multilateral research efforts on the six Gen IV systems reviewed in the 2009 report. More importantly, it addresses the remaining challenges and the R&D needed to overcome these challenges on the path to viability, performance demonstration, or in some cases, to the demonstration and deployment of innovative concepts. The Gen IV goals of improved sustainability, economics, safety and reliability, proliferation resistance and physical protection are reviewed in the context of today's transition towards low-carbon energy systems. Initiatives taken by GIF in the area of education and training, research infrastructures, development of safety approaches and criteria, economic modelling and analyses of future energy markets, as well as advanced manufacturing, all of which are designed to help support the deployment of the first Gen IV reactor prototypes, are also outlined in this report. In the context of rapidly evolving energy landscapes, the Generation IV International Forum is calling upon policy makers to accelerate the demonstration and deployment of innovative nuclear technologies so that nuclear energy can effectively contribute to a successful transition.

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