Technology Roadmap Update for Generation IV Nuclear Energy Systems

GEN IV International Forum[SM]

Preparing Today for Tomorrow's Energy Needs

Technology Roadmap Update for Generation IV Nuclear Energy Systems

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Foreword

The first decade of the Generation IV International Forum (GIF) has been marked by opposing trends with, on the one hand, a worldwide renewed interest in nuclear energy as an element of the solution to global warming and as a means of delivering power to both emerging and developed countries; and, on the other hand, by hesitations about the future of nuclear development in the aftermath of the Fukushima Daiichi nuclear power plant accident.

Given the changing circumstances after ten years of collaborative work, the GIF Policy Group decided to update the original 2002 *Technology Roadmap*. The main objective of this update is to focus on the most relevant developments of the six GEN IV systems originally selected (no new system has been considered) and provide, to the extent possible, a realistic high-level technical progress report.

The goal of this report is to assess the current technology status of each system, using the general classification of viability, performance, demonstration or commercialisation phases, and to identify the key remaining R&D challenges as well as ways of overcoming them through the GIF international collaboration framework.

For each nuclear system, the roadmap outlines the accomplishments to date, presents the current status of each system internationally, and provides the main R&D objectives along with the milestones anticipated in the next decade.

Developing the technologies and associated system designs to the point of commercialisation for each of the six systems identified in the original roadmap would have required a multi-year, multi-billion dollar international commitment. This was not the case and, as a result, the degree of technical progress of the different systems over the past decade is not uniform, having depended to a large extent on national priorities and efforts within GIF member countries. A number of the participating countries invested significant resources in the development of the sodium-cooled fast reactor (SFR) and very-high-temperature reactor (VHTR), in large part due to the considerable historical effort associated with these technologies. More limited resources were invested in the remaining four concepts (supercritical-water-cooled reactor [SCWR], lead-cooled fast reactor [LFR], gas-cooled fast reactor [GFR] and molten salt reactor [MSR]).

Such a situation should not be interpreted as a technical assessment of the desirability of any given concept over another or as a down-selection among the concepts. Each of the six concepts still retains the necessary characteristics (economics, safety and reliability, proliferation resistance, physical protection and sustainability) to be considered a Generation IV system.

I would like to thank the members of the team that worked on this document: David Petti, Wenquan Shen, Alexander Tuzov and Martin Zimmermann, as well as Alexey Lokhov and Jean-Claude Bouchter, who provided technical secretariat support to the activity. I would also like to thank the expert group members and the system steering committees and working group chairs for their essential contributions, as well as the Policy Group members for their general remarks and guidance.

Christophe Behar
GIF Vice-Chair

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Executive summary

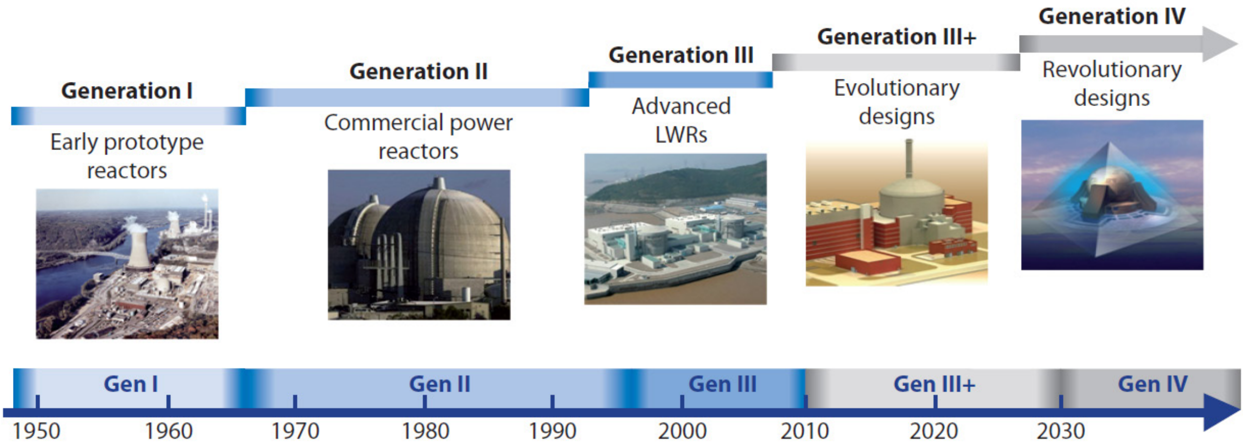
Nuclear power plants, which produce low-carbon electricity at stable and competitive costs, constitute an element of the solution to global warming and a means of delivering power to emerging and developed countries. Further development of nuclear technology is needed to meet future energy demand.

The Generation IV International Forum (GIF) was created in January 2000 by 9 countries, and today has 13 members,[1] all of which are signatories of the founding document, the GIF Charter.[2]

GIF defined in its *Technology Roadmap*[3] four goal areas to advance nuclear energy into its next, "fourth" generation (see Figure ES.1):

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

Figure ES.1: Generations of nuclear power: Time ranges correspond to the design and the first deployments of different generations of reactors



The *Technology Roadmap* also defined and planned the necessary R&D to achieve these goals and allow for the deployment of Generation IV energy systems after 2030. Generation IV nuclear energy systems include the nuclear reactor and its energy conversion systems, as well as the necessary fuel cycle technologies.

1. Argentina, Brazil, Canada, China, Euratom, France, Japan, the Republic of Korea, the Russian Federation, South Africa, Switzerland, the United Kingdom and the United States.
2. The Charter was officially established in July 2001.
3. The full report, *A Technology Roadmap for Generation IV Nuclear Energy Systems*, GIF-002-00, 2002 is available at: www.gen-4.org.

Closing of the nuclear fuel cycle is an important component for achieving the sustainability goal. It is based on the reprocessing and partitioning of spent nuclear fuel and the management of each fraction with the best possible strategy. Fissile material, for example, can be recovered from the spent fuel and used to make new fuel. At present, almost 95% of the spent fuel from light water reactors can be reused in the form of reprocessed uranium and MOX fuel.

With advanced fuel cycles using fast-spectrum reactors and extensive recycling, it may be possible to breed fissile fuel from fertile material, and thus produce equal or more fissile material than the reactor consumes. This would also significantly reduce the footprint of deep geological repositories for the disposal of ultimate waste. The advanced separation technologies for Generation IV systems are being designed to avoid the separation of sensitive materials, and they include other features to enhance proliferation resistance and incorporate effective safeguards.

The *Technology Roadmap* established an understanding of the ability of various reactors to be combined in so-called symbiotic fuel cycles, for example, through combinations of thermal reactors and fast reactors to accommodate transition periods. This was one of the primary motivations for having a portfolio of Generation IV systems rather than a single system in the original *Technology Roadmap,* since various combinations of a few systems in the portfolio would provide a symbiotic system worldwide.

In 2002, GIF selected six systems from nearly 100 concepts as Generation IV technologies:

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

The *Technology Roadmap Update* has confirmed the choice of these six systems.

Timelines and research needs were developed for each system, categorised in three successive phases:

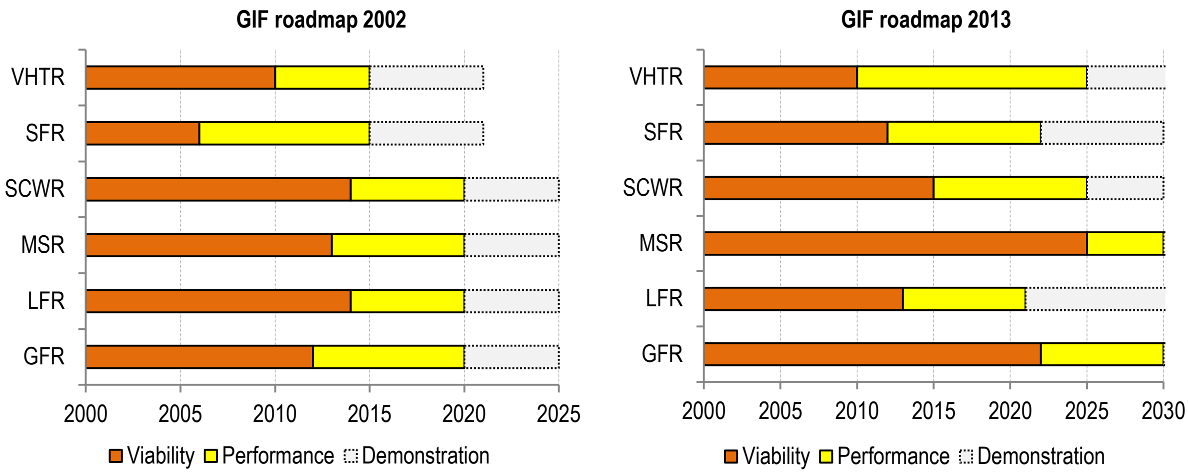
- the *viability phase,* when basic concepts are tested under relevant conditions and all potential technical show-stoppers are identified and resolved;
- the *performance phase,* when engineering-scale processes, phenomena and materials capabilities are verified and optimised under prototypical conditions;
- the *demonstration phase,* when detailed design is completed and licensing, construction and operation of the system are carried out, with the aim of bringing it to the commercial deployment stage.

The main objective of this *Technology Roadmap Update* is to focus on the most relevant developments of the six Generation IV systems selected, providing a high-level report that summarises the achievements of the past ten years and defines R&D goals for the next decade.

The original and updated timelines for each GIF system are summarised in Figure ES.2, and the main milestones for each system and methodology working group are provided in Table ES.1.

The development of technologies and associated system designs to the point of commercialisation for each of the six systems, as identified in the original *Technology Roadmap*, would have required considerable investment and international commitment. Since the "starting point" and R&D funding of the different Generation IV systems were not equivalent, the degree of technical progress over the past decade has not been uniform for all systems. A number of participating countries devoted significant resources to the development of the SFR and VHTR, for example, in large part due to the considerable historical effort associated with these technologies. More limited resources were dedicated to the other systems.

Figure ES.2: System development timelines as defined in the original 2002 Roadmap (left) and in the 2013 update[4]



The Fukushima Daiichi nuclear power plant accident has emphasised the importance of designing nuclear systems with the highest levels of safety. Lessons learnt from the accident will benefit the current operating fleet, as well as future nuclear systems, including Generation IV systems. The accident demonstrated in particular the need for reliable residual heat removal over long periods as well as the necessity to exclude significant off-site releases in case of a severe accident. For the Generation IV systems, an additional set of questions has to be analysed in detail and compared to the work on advanced light water reactors. These relate, in particular, to:

- the use of non-water coolants in most Generation IV designs;
- higher operational temperatures;
- higher reactor power density;
- in some cases, the close location or integration of fuel-cycle or chemical facilities.

In the coming years, GIF will work on demonstrating the capability of Generation IV systems to achieve the highest level of safety, taking into account the lessons learnt from the Fukushima Daiichi accident.

4. These timelines are indicative and may change, for example, if structural materials, fuel or other important components are not validated at the planned dates.

Table ES.1: Key objectives for the next 10 years

GIF SYSTEM	
GFR	• Reference concept of 2 400 MW_{th} reactor capable of breakeven breeding. • Improving the design for the safe management of loss-of-coolant accidents including depressurisation, and a robust removal of decay heat without external power supply. • Advancing suitable nuclear fuel technologies with out-of-pile and irradiation experiments. • Building experimental facilities for qualifying the main components and systems. • Design studies for a small experimental reactor (e.g. ALLEGRO).
LFR	• Prototypes expected after 2020: Pb-Bi-cooled SVBR-100, BREST-300 in Russia. • Proceeding with detailed design and licensing activities. • Preliminary analyses of accidental transients including earthquakes and in-vessel steam generator pipe ruptures. • Main R&D efforts will be concentrated on: – materials corrosion and development of a lead chemistry management system; – core instrumentation; – fuel handling technology and operation; – advanced modelling and simulation; – fuel development (MOX for first core, then MA-bearing fuels); and possibly nitride fuel for lead-cooled reactors (BREST); – actinide management (fuel reprocessing and manufacturing). – ISI&R (techniques for opaque medium, seismic impact).
MSR	• A baseline concept: the molten salt fast reactor (MSFR). • Commonalities with other systems using molten salts (FHR, heat transfer systems). • Further R&D on liquid salt physical chemistry and technology, especially on corrosion, safety-related issues and treatment of used salt.
SFR	• Three baseline concepts (pool, loop and modular configurations). • Several sodium-cooled reactors operational or under construction (e.g. in China, India, Japan and Russia). • Develop advanced national SFR demonstrators for near-term deployment (France, Japan and Russia); proceed with respective national projects in China, Korea and India. • In the coming years, the main R&D efforts will be concentrated on: – safety and operation (improving core inherent safety and I&C, prevention and mitigation of sodium fires, prevention and mitigation of severe accidents with large energy releases, ultimate heat sink, ISI&R); – consolidation of common safety design criteria; – advanced fuel development (advanced reactor fuels, MA-bearing fuels); – component design and balance of plant (advanced cycles for energy conversion, innovative component design); – used fuel handling schemes and technologies; – system integration and assessment; – implementation of innovative options; – economic evaluations, operation optimisation.
SCWR	• Two baseline concepts (pressure-vessel-based and pressure-tube-based). • R&D over the next decade will include: – advancing conceptual designs of baseline concepts and associated safety analyses; – more realistic testing of materials to allow final selection and qualification of candidate alloys for all key components; – out-of-pile fuel assembly testing; – qualification of computational tools; – first integral component tests and start of design studies for a prototype; – in-pile tests of a small scale fuel assembly in a nuclear reactor. • Definition of a SCWR prototype (size, design features) for decisions to be taken in the coming years.

Table ES.1: Key objectives for the next 10 years *(cont'd)*

VHTR	• In the near future, the main focus will be on VHTR with core outlet temperatures of 700-950°C. • Further R&D on materials and fuels should enable higher temperatures up to above 1 000°C and a fuel burnup of 150-200 GWd/tHM. • Development of further approaches to set up high-temperature process heat consortia for end-users interested in prototypical demonstrations. • Development of the interface with industrial heat users – intermediate heat exchanger, ducts, valves and associated heat transfer fluid: – Advancing H2 production methods in terms of feasibility and commercial viability to better determine process heat requirements for this application. – Regarding nuclear safety: ▪ Verify the effectiveness and reliability of the passive heat removal system. ▪ Confirm fuel resistance to extreme temperatures (~1 800°C) through testing. ▪ Proceed with the safety analyses of coupled nuclear processes for industrial sites using process heat.
GIF METHODOLOGY WORKING GROUPS	
Economic Modeling Working Group (EMWG)	• Over the next two to three years, the EMWG will release a new version of the G4ECONS cost estimating code with advanced capabilities. • The *Cost Estimating Guidelines* will be reviewed after the *User Guide* updates: – Over the next 10 years, the EMWG will continue to monitor the progress of Generation IV systems economic analyses and further improve the methodology consistent with these designs.
Proliferation Resistance and Physical Protection Working Group (PRPPWG)	• As new and innovative designs for nuclear energy systems are developed through GIF (and other possible fora), the PR&PP methodology approach will be essential to incorporate good design principles for proliferation resistance and physical protection into these new designs. • Enable safeguards by design: Robust safeguards are essential to the PR&PP characteristics of all of the emerging GIF designs. • Assist GIF system developers in introducing PR&PP concepts into their design work.
Risk and Safety Working Group (RSWG)	• In 2008, the RSWG published the *Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems* – a consensus regarding some of the safety-related attributes and characteristics that should be reflected in Generation IV systems: – Future work: Provision for application of the integrated safety assessment methodology (ISAM) in the development of Generation IV systems. – A number of detailed analyses and "lessons learnt" investigations will be performed, especially as related to the Fukushima Daiichi accident.

Chapter 1. Introduction

Generation IV International Forum

The Generation IV International Forum (GIF) was created in January 2000 by 9 countries and today has 13 members, all of, which are signatories of its founding document, the *GIF Charter*. Argentina, Brazil, Canada, France, Japan, the Republic of Korea, South Africa, the United Kingdom and the United States signed the GIF Charter in July 2001. It was subsequently signed by Switzerland in 2002, Euratom[1] in 2003, and the People's Republic of China and Russian Federation in 2006.

GIF considers that nuclear energy is needed to meet future energy demand, and that international collaboration is required to advance nuclear energy into its next "fourth" generation of systems, deployable after 2030 (see Table 1). GIF defined four goal areas in its original *Technology Roadmap:* [2]

- sustainability;
- safety and reliability;
- economics;
- proliferation resistance and physical protection.

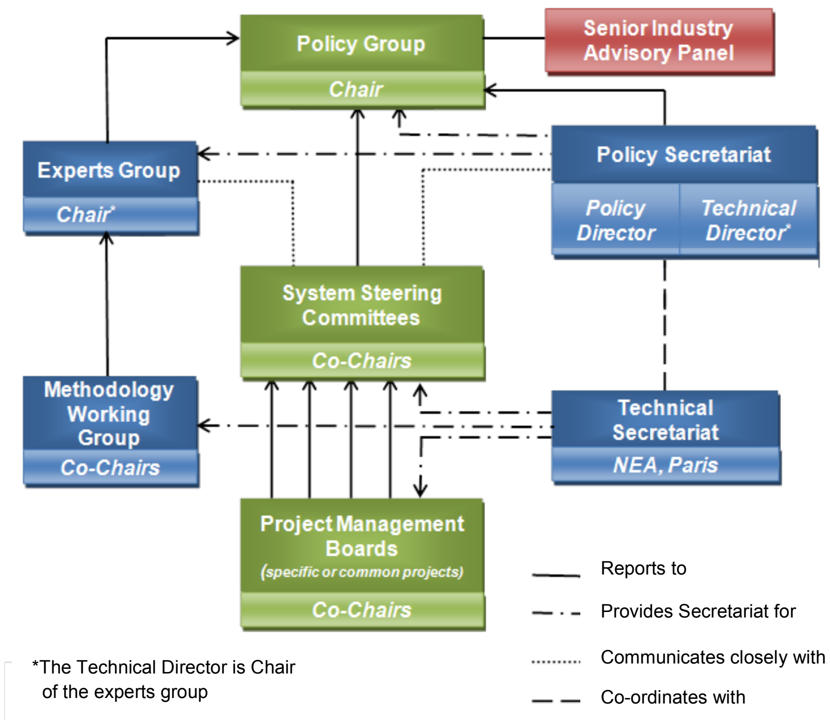
Table 1: Generations of nuclear power

<table>
<tr><th></th><th>Generation I</th><th>Generation II</th><th>Generation III</th><th>Generation III+</th><th>Generation IV</th></tr>
<tr><td>Period of deployment</td><td>1950-1960</td><td>1970-1990</td><td>1990-2000</td><td>After 2000</td><td>After 2030</td></tr>
<tr><td>Examples</td><td>Shippingport, Dresden, Magnox</td><td>PWR and VVER, BWR, CANDU</td><td colspan="2">ABWR, AES-92
AP1000, EPR</td><td></td></tr>
<tr><td rowspan="2">Comments</td><td>Early prototype reactors</td><td>Large commercial power plants that are still operating today</td><td>Advanced LWR</td><td>Evolutionary designs offering improved economic and safety features</td><td>– Life-cycle economic advantage
– Enhanced safety
– Minimal waste
– Proliferation resistant</td></tr>
<tr><td></td><td></td><td colspan="2">One indication of being part of Generation III/III+ is certification by EUR (in Europe) or EPRI/URD (in the United States) utilities requirements[3]</td><td></td></tr>
</table>

1. The European Atomic Energy Community (Euratom) is the implementing organisation for development of nuclear energy within the European Union.
2. *A Technology Roadmap for Generation IV Nuclear Energy Systems*, GIF-002-00, 2002. Available at: www.gen-4.org.
3. In the late 1980s, utilities from the United States, Europe and Asia united their efforts in preparing a set of requirements for advanced light water reactors. In 1990, the first edition of the advanced light water reactors (ALWRs) utility requirements document (URD) was issued by the

The GIF Charter provides a general framework for GIF activities and outlines its organisational structure. Figure 1-1 gives a schematic representation of the GIF governance structure and indicates the relationship among different GIF bodies.

Figure 1-1: GIF governance structure



Generation IV goals

The following goals were defined in the original GIF Charter:

Sustainability

- Generate energy sustainably and promote long-term availability of nuclear fuel.
- Minimise nuclear waste and reduce the long term stewardship burden.

Safety and reliability

- Excel in safety and reliability.
- Have a very low likelihood and degree of reactor core damage.
- Eliminate the need for offsite emergency response.

Electric Power Research Institute (EPRI) in the United States. In 1991, five European utilities considered that a more open specification would be needed to cover a wider range of designs, and thus the European Utilities' Requirements (EUR) organisation was created. The EUR covers a broad range of conditions for a nuclear power plant to operate efficiently and safely. These include plant layout and specifications, systems, materials, components, probabilistic safety assessment methodology and availability assessment. Although a reactor still requires regulatory design approval in each country, EUR compliance indicates that a reactor design meets a list of requirements set by the utilities for the next generation of light water reactors (LWRs). Plants certified as complying with EUR include very different designs: AP1000, AES-92 (with VVER-1000/V-392), EPR, ABWR, KERENA and BWR 90.

Economics

- Have a life cycle cost advantage over other energy sources.
- Have a level of financial risk comparable to other energy projects.

Proliferation resistance and physical protection

- Be a very unattractive route for diversion or theft of weapon-usable materials, and provide increased physical protection against acts of terrorism.

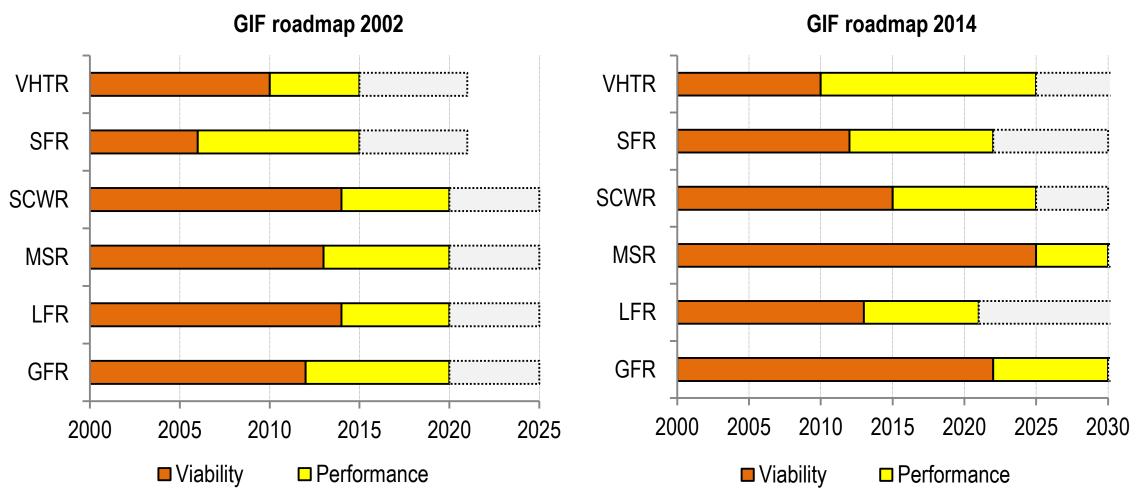
The Technology Roadmap

The 2002 Roadmap

The *Technology Roadmap* (2002), defined and planned the necessary R&D and associated timelines to achieve these goals and allow deployment of Generation IV energy systems after 2030 (see Figure1-2). This roadmapping exercise was a two-year effort by more than 100 international experts to select the most promising nuclear systems. In 2002, GIF selected the six systems listed below, from nearly 100 concepts, as Generation IV systems:

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

Figure 1-2: System development timelines as defined in the original Roadmap in 2002 (left) and in the 2014 update[4]



4. These timelines are indicative and may change, for example, if structural materials, fuel or other important components are not validated at the planned dates.

Figure 1-2: System development timelines as defined in the original Roadmap in 2002 (left) and in the 2014 update[4] *(cont'd)*

Viability phase	Performance phase	Demonstration phase
Basic concepts, technologies and processes are tested under relevant conditions, with all potential technical *show-stoppers* identified and resolved.	Engineering-scale processes, phenomena and materials capabilities are verified and optimised under prototypical conditions.	Assuming the successful completion of viability and performance R&D, a demonstration phase of at least 10 years is anticipated for each system, requiring funding of several billion U.S. dollars. This phase involves the licensing, construction and operation of a prototype or demonstration system in partnership with industry and perhaps other countries. The detailed design will be completed and licensing of the system will be performed during this phase.
Viability phase endpoints	**Performance phase endpoints**	
• Pre-conceptual design of the entire system, with nominal interface requirements between subsystems and established pathways for disposal of all waste streams. • Basic fuel cycle and energy conversion (if applicable) process flow sheets established through testing at appropriate scale. • Cost analysis based on pre-conceptual design. • Simplified PRA for the system. • Definition of analytical tools. • Pre-conceptual design and analysis of safety features. • Simplified preliminary environmental impact statement for the system. • Preliminary safeguards and physical protection strategy. • Consultation(s) with regulatory agency on safety approach and framework issues.	• Conceptual design of the entire system, sufficient for procurement specifications for construction of a prototype or demonstration plant, and with validated acceptability of disposal of all waste streams. • Processes validated at scale sufficient for demonstration plant. • Detailed cost evaluation for the system. • PRA for the system. • Validation of analytical tools. • Demonstration of safety features through testing, analysis or relevant experience. Environmental impact statement for the system. • Safeguards and physical protection strategy for the system, including cost estimate for extrinsic features. • Pre-application meeting(s) with regulatory agency.	

System Arrangements have been established for four systems (SFR, VHTR, SCWR and GFR) and Memoranda of Understanding (MOU) were signed for each of the remaining systems (LFR and MSR). The status of these arrangements and MOUs as of January 2014 is shown in Figure 1-3.

Figure 1-3: Status of the GIF System Arrangements and Memoranda of Understanding
(as of 1 January 2014)

System	CA	CN	EU	FR	JP	KR	RU	CH	US	ZA
SFR		✓	✓	✓	✓	✓	✓		✓	
VHTR		✓	✓	✓	✓	✓		✓	✓	
SCWR	✓		✓		✓		✓			
GFR			✓	✓	✓			✓		
LFR			P		P		P			
MSR			P	P			P			

✓= Signatory to the System Arrangement; P = signatory to the Memorandum of Understanding; Argentina, Brazil, and the United Kingdom are inactive.

Objectives of the technology roadmap update

The main objective of this *Technology Roadmap Update* is to formulate a high-level report summarising the achievement of the past ten years and defining the R&D steps for the next decade, with more details for the coming three to five years. Worthwhile and challenging goals, as well as activities and projects to be accomplished in next decade through GIF, are explored. In this context, the following questions are considered in the update:

- Are there new/modified technical issues to be addressed?
- Are there any new concepts to be considered for R&D collaboration within GIF and should GIF continue working on all six systems originally considered?
- What is the impact of the Fukushima Daiichi accident on Gen IV goals and safety targets?
- What are the requirements for future R&D and prototype/technology demonstration needs in the near term (~10 years), including economic assessments?

Lessons learnt from the Fukushima Daiichi accident relevant to the work of GIF

The Fukushima Daiichi nuclear power plant (NPP) accident, which occurred on 11 March 2011, resulted from the massive Great East Japan earthquake (magnitude 9 on the Richter scale, the largest ever recorded in Japan) and the ensuing tsunami (estimated at more than 14 metres) that hit the Fukushima Daiichi nuclear power plant. This caused wide-scale flooding of the site with the subsequent failure of the emergency diesel generators and the pumps that provided cooling water from the ultimate heat sink (the Pacific Ocean).

All of the safety systems that relied on electrical power to meet their function of protecting the fuel in the cores at units 1, 2 and 3 failed. The systems that did not rely on electrical power were available for a short time following the accident; however, they also eventually failed. When cooling was lost to the cores at units 1, 2 and 3, significant fuel damage occurred. Core melting is estimated to have begun at unit 1 several hours after the tsunami struck the site; cooling was lost at unit 3 on 13 March and at unit 2 on 14 March.

Time will be needed to collect the data from the three damaged reactors at Fukushima Daiichi and fully analyse the accident and its consequences. Nuclear regulators across the world, as well as international organisations – mainly the IAEA and the NEA – are working to draw lessons from the accident.

These lessons concern: the capability of the nuclear power plant to respond to extreme natural and man-made events and combinations thereof; consequential loss of safety systems, associated with long-term loss of electrical supplies and the ultimate heat sink, and severe accident management systems; and loss of core and spent fuel pool cooling and containment integrity. Such lessons learnt will be applicable to the current operating fleet, as well as to new reactor designs and fuel cycle facilities.

The Fukushima Daiichi accident demonstrated the need for reliable residual heat removal over long periods as well as the necessity to exclude significant off-site releases in the case of a severe accident. For the Generation IV systems, there is a set of additional questions that have to be analysed in detail, as compared to the issues to be addressed for advanced light water reactors. These relate in particular to:

- the use of non-water coolants in most Generation IV designs;
- higher operational temperatures;
- higher reactor power density;
- in some cases, close location or integration of fuel-cycle or chemical facilities.

The capability of Generation IV systems to achieve the safety goals must be demonstrated. Any Generation IV nuclear system will be licensed only if it fulfils the stringent requirements summarised in the Generation IV safety and reliability goals. The implementation of lessons learnt from the Fukushima Daiichi accident is also considered in this report.

Chapter 2. Ten-year objectives for the most promising systems

Gas-cooled fast reactor (GFR)

The GFR system is a high-temperature helium-cooled fast-spectrum reactor with a closed fuel cycle. It combines the advantages of fast-spectrum systems for long-term sustainability of uranium resources and waste minimisation (through fuel multiple reprocessing and fission of long-lived actinides), with those of high-temperature systems (high thermal cycle efficiency and industrial use of the generated heat, similar to VHTR).

The advantages of the gas coolant are that it is chemically inert (allowing high-temperature operation without corrosion and coolant radio-toxicity) and single phase (eliminating boiling), and it has low neutron moderation (the void coefficient of reactivity is small).

However, there are some technological challenges associated with the use of gas coolant without the graphite that is common in the HTR system. Its low thermal inertia leads to rapid heat-up of the core following loss of forced cooling. Since the power density is high in the GFR, the HTR-type "conduction cool-down" will not work for the removal of the decay heat, and other solutions must be considered. Also, the gas-coolant density is too low to achieve enough natural convection to cool the core, and the power requirements for the blower are important at low pressure. Lastly, additional consideration will need to be given to the effects of the fast neutron dose on the reactor pressure vessel in the absence of core moderation (the graphite moderator provides protection for HTR systems).

GAS-COOLED FAST REACTOR (GFR) IN THE NEXT DECADE

- Reference concept of 2 400 MW_{th} reactor capable of breakeven breeding.
- Improving the design for the safe management of loss-of-coolant accidents including depressurisation, and a robust removal of decay heat without external power supply.
- Advancing suitable nuclear fuel technologies with out-of-pile and irradiation experiments.
- Building experimental facilities for qualifying the main components and systems.
- Design studies for a small experimental reactor (e.g. ALLEGRO).

The reference design for GFR is currently based around 2 400 MW_{th}, since the 600 MW_{th} reactor presented in the original roadmap could not meet the breakeven-breeding requirement. The 600 MW_{th} is still considered as an option for a gas-cooled small modular reactor (SMR) that does not need to be a breakeven-breeder.

The direct power conversion cycle chosen as a reference in the original roadmap is no longer considered the only option. It was originally assumed that the HTR community would develop this technology in projects such as PBMR in South Africa and GT-MHR in the United States and Russia. Today in the United States, a commercial entity is developing the conceptual design of a small GFR and its associated technologies. Some near-term thermal HTR projects have moved away from the direct cycle concept, favouring the indirect cycle because of its lower technological risk and higher flexibility with respect to the choice of working fluid for the turbine. Therefore, the reference concept is an indirect cycle with helium on the primary circuit, a Brayton cycle on the secondary circuit and a steam cycle on the tertiary circuit (see Figure 2-1).

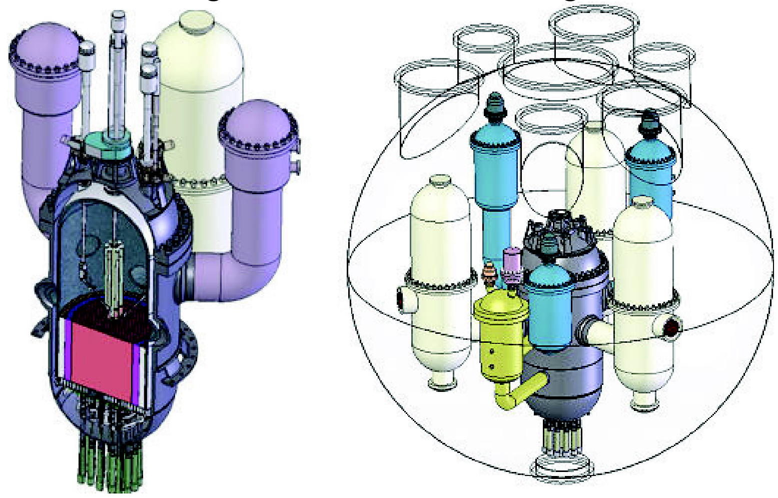
Major accomplishments in the last decade

The system arrangement was signed at the end of 2006 by Euratom, France, Japan and Switzerland. It is to be noted that, while France has been very active in the development of the GFR concept, in particular conceptual design, safety assessment and fuel development in the previous years, in 2010 French research priorities were re-focused on sodium-cooled fast reactors, which led to less effort being invested in the GFR system. Furthermore, the Fukushima Daiichi accident in 2011 has shifted priorities away from GFR in Japan, and to a lesser extent in Switzerland.

In addition to their national programmes, France and Switzerland are very active members within Euratom, with a number of organisations in France and PSI in Switzerland being members of the GoFastR project (Euratom FP7), which provided, up to 2013, the main contribution from Euratom to the GIF GFR system development.

Two projects were discussed at the origin of the SA, dealing with conceptual design and safety (CD&S), and fuel and core materials (FCM). The CD&S project arrangement was signed in 2009 by Euratom, France and Switzerland, and is effective since 17 December 2009. The FCM project arrangement remains unsigned and the participants have agreed to continue their collaboration on an informal basis.

Figure 2-1: GFR reference design



GFR – reactor, decay heat loops, main heat exchangers and fuel handling equipment

GFR – spherical guard vessel

R&D objectives

Experimental reactor

The original need remains for a small experimental reactor to be available within the next 10-20 years. The ALLEGRO experimental reactor project currently being undertaken by a consortium of four countries (the Czech Republic, Hungary, Poland and the Slovak Republic) fulfils this requirement. ALLEGRO will be the first fast spectrum gas-cooled reactor to be constructed and will be the test bed to develop and qualify the high-temperature, high-power density fuel that is required for a commercial-scale high-temperature GFR. This fuel qualification will be carried out at full scale, in the correct coolant at representative temperature and with the correct neutron spectrum and flux.

Fuel development and small-scale irradiation

Development of an acceptable fuel system is a key viability issue for the GFR system. It is necessary to develop a cladding material that meets the core specifications in terms of length, diameter, surface roughness, apparent ductility, level of leak tightness (including the potential need of a metallic liner on the clad), compatibility with helium coolant (plus impurities), and the anticipated irradiation conditions. The needs include fabrication capacities and material characterisation under normal and accidental conditions for fresh and irradiated fuel.

The target criteria are:

- clad temperature of 1 000°C, during normal operation;
- no fission product release for a clad temperature of 1 600°C during a few hours;
- maintaining the core-cooling capability up to a clad temperature of 2 000°C.

Out-of-pile experimental facilities for qualification of the main systems

In terms of neutronics and zero-power reactor needs, 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, including: slightly different spectral conditions, innovative materials and various ceramic materials (UC, PuC, SiC, ZrC, Zr_3Si_2), and unique abnormal conditions (depressurisation, steam ingress).

For core thermal hydraulics, air and then helium tests on sub-assembly mock-ups are necessary to assess heat transfer and pressure drop uncertainties of the specific GFR technology selected.

A large scale demonstration of the passive DHR function will be required (air and then helium tests) by the licensing process of ALLEGRO for assessing the system transient behaviour.

The development and qualification of components include:

- **Specific blowers and turbomachines.** These are needed to cope with a wide range of pressure operations (from 70 to 1 bar) with rotating parts that retain their leak tightness.
- **Thermal barriers**. During normal operation, the GFR metallic structures are protected from the hot (850°C) helium flow by thermal barriers. These thermal barriers must continue to be effective during transients, typically up to 1 250°C for 1 hour; withstand helium velocities of about 60 $m.s^{1}$ and depressurisation rates in the range of 2 $MPa.s^{-1}$. GFR-specific solutions must be developed and qualified in relevant facilities.
- **Valves and check-valves**. The safety demonstration of the GFR relies on continuous core cooling by gas circulation, either through normal loops or dedicated DHR loops for which it is necessary to isolate the main loops and open the DHR loops with a high degree of reliability. Valves and check-valves are therefore critical components of the GFR. Qualification tests of candidate technologies for these components are needed and must be performed using a dedicated helium loop.
- **Instrumentation**. The development of instrumentation that can survive under GFR conditions is one of the main issues of gas-cooled reactors. In particular, the main

safety issue concerns the temperature measurement at the core outlet, in order to be able to detect hot spots on the fuel cladding or fuel assembly plugging. The primary development objective is the reduction of measurement uncertainties and also the development of innovative measurement methods, using if possible the helium transparency. The instrumentation R&D programme includes core temperature measurement, monitoring of structural temperatures, and optical viewing during fuel handling and maintenance phases.

The GFR also requires a dense fuel element that can withstand very high temperature transients, due to the lack of thermal inertia of the system. Ceramic or refractory metal clad should be selected, developed and qualified. The development programme requires material properties measurements, selection of different materials, their arrangement and their interaction, out-of and in-pile tests up to qualification, and demonstration tests.

Safety objectives

The need to ensure robust decay heat removal (DHR) without external power input, even in depressurised conditions, is now regarded as a requirement. Previous concepts used electrical (battery) driven blowers to handle depressurised DHR. Although the DRH system has no diesel power units that would need protection from potential flooding, integrity of the electrical infrastructure following an extreme event is still required.

Work is required on two fronts; first to reduce the likelihood of full depressurisation and second, to increase the autonomy of the DHR system through the use of self-powered systems. While these self-powered systems cannot be considered passive, they do not require any external power input.

Finally, the strategy to deal with severe accidents is to be established.

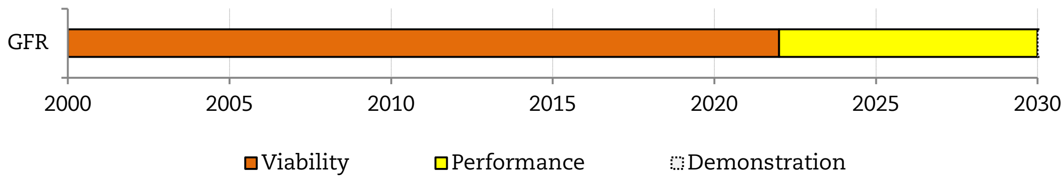
Milestones

Progress has been made within the GIF GFR Conceptual Design and Safety Project Arrangement with a focus on safety aspects. GFR fuel development is critical for this reactor system, and requires further international collaboration. Nevertheless, a new initiative (ALLEGRO demonstrator) has been launched recently.

In the next 10-20 years, the following steps in GFR development are expected:

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

Timelines for the GFR



Lead-cooled fast reactor (LFR)

LFRs are Pb or Pb-Bi-alloy-cooled reactors operating at atmospheric pressure and at high temperature because of the very high boiling point of the coolant (up to 1 743°C). The core is characterised by a fast-neutron spectrum due to the scattering properties of lead.

Pb and Pb-Bi coolants are chemically inert and possess several attractive properties:

- There is no exothermic reaction between lead and water or air.

- The high boiling point of lead eliminates the risk of core voiding due to coolant boiling.
- The high density of the coolant contributes to fuel dispersion instead of compaction in case of core destruction.
- The high heat of vaporisation and high thermal capacity of lead provide significant thermal inertia in case of loss-of-heat-sink.
- Lead shields gamma-rays and retains iodine and caesium at temperatures up to 600°C, thereby reducing the source term in case of release of volatile fission products from the fuel.
- The low neutron moderation of lead allows greater spacing between fuel pins, leading to low core pressure drop and reduced risk of flow blockage.
- The simple coolant flow path and low core pressure drop allow natural convection cooling in the primary system for shutdown heat removal.

LEAD-COOLED FAST REACTOR (LFR) IN THE NEXT DECADE

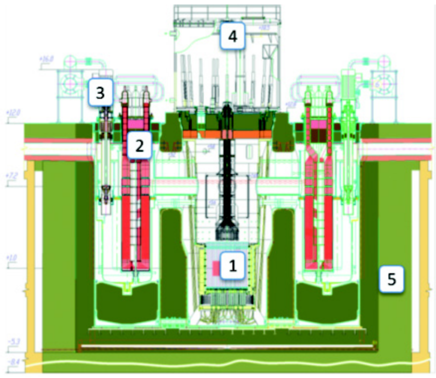
- Prototypes expected after 2020: Pb-Bi-cooled SVBR-100, BREST-300 in Russia.
- Proceeding with detailed design and licensing activities.
- Preliminary analyses of accidental transients including earthquake and in-vessel steam generator pipe ruptures.
- Main R&D efforts will be concentrated on:
  - materials corrosion and development of a lead chemistry management system;
  - core instrumentation;
  - fuel handling technology and operation;
  - advanced modelling and simulation;
  - fuel development (MOX for first core, then MA-bearing fuels); and possibly nitride fuel for lead-cooled reactors (BREST);
  - actinide management (fuel reprocessing and manufacturing);
  - ISI&R (techniques for opaque medium, seismic impact).

However, several drawbacks must be overcome, including the need for coolant chemical (oxygen) control for prevention of lead erosion-corrosion effects[1] on structural steels at high temperatures and flow rates, and seismic/structural issues because of the weight of the coolant. The opacity of lead, in combination with its high melting temperature, presents challenges related to inspection and monitoring of reactor in-core components as well as fuel handling. In particular, in the case of reactor system cooled by pure Pb, the high melting temperature of lead (327°C) requires that the primary coolant system be maintained at temperatures adequately high to prevent the solidification of the lead coolant.

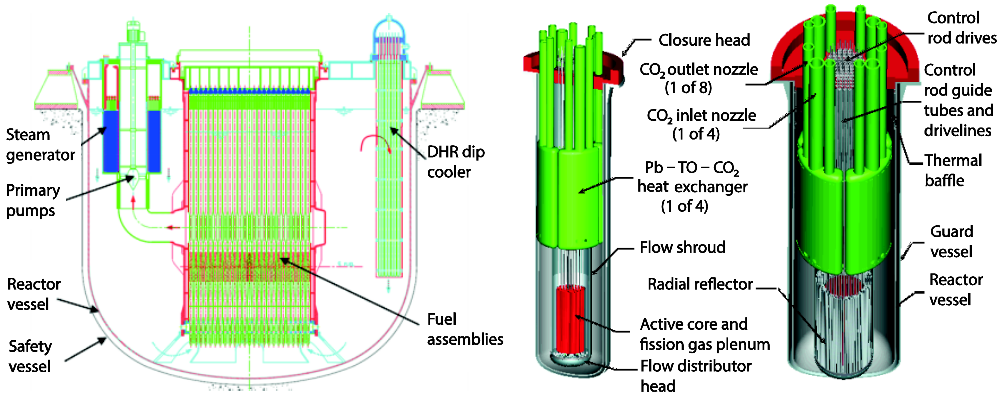
1. The most resistant materials are refractory metals and chromium steels. Oxide films formed on the steel surface tend to protect against such effects. This phenomenon can be used to slow down the corrosion of structural materials. However, there is a considerable challenge in maintaining the required oxygen content in the coolant to provide stability of the protective iron oxide film on the steel surface, and, at the same time, to avoid a surplus of PbO in the coolant that could result in circuit plugging. The oxygen content in the coolant can be regulated by either bubbling gas mixtures through the lead or by passing the molten lead through a lead oxide filling. (see IAEA No. NP-T-1.6, available at: www-pub.iaea.org/MTCD/Publications/PDF/P1567_web.pdf).

Although Pb-Bi reactors have been operated successfully in some of the Russian submarine programmes, this experience cannot be easily extrapolated to the LFR since the propulsion reactors were small, operated at low capacity factors, featured an epithermal (not fast) neutron spectrum and operated at significantly lower temperatures[2] than those anticipated in Gen-IV lead-cooled fast reactors. An additional issue with the lead-bismuth cooled reactors is related to the accumulation of volatile Polonium-210 which is a strong alpha emitter[3]. In the Russian Federation, techniques to trap and remove ^{210}Po have been developed.

Figure 2-2: LFR reference designs: BREST-OD-300 (top), ELFR (left) and SSTAR (right)



1 Core; 2 Steam generator; 3 Pump; 4 Refueling machine; 5 Reactor vault



The LFR systems identified by GIF include a wide range of plant ratings from the small to intermediate and large size. Important synergies exist among the different systems so that a co-ordination of the efforts carried out by participating countries will be one of the key points of LFR development. The options considered are: a small transportable system of 10-100 MWe size (Small Secure Transportable Autonomous Reactor or SSTAR – United States) that features a very long core life; a system of intermediate size (BREST 300 – Russia); and a larger system rated at about 600 MWe (European Lead Fast Reactor or ELFR – Euratom), intended for grid-connected power generation (see Figure 2-2).

2. Although the inlet temperatures of the submarine reactors were substantially lower than current designs for both LBE-cooled and lead-cooled reactors, these current designs have temperature cycles that are similar.
3. ^{210}Po is generated from ^{209}Bi under irradiation and has a half-life of about 138 days.

The typical configuration of LFRs is a pool type configuration without an intermediate heat exchanger system. Because of the chemical inertness of the coolant, the secondary side system (delivering high pressure superheated water) can be interfaced directly with the primary side using steam generators immersed in the pool. The expected secondary cycle efficiency of LFR systems is above 42%.

The LFR system features a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. A full actinide recycle fuel cycle with central or regional fuel cycle facilities is envisioned. The LFR system is well positioned to fulfil the four goals of GIF, primarily because of the coolant inertness (and corresponding simplified plant design) and the use of a closed fuel cycle. Proliferation resistance and physical protection goals are achieved by using MA-bearing MOX fuel. The safety goal is intended to be achieved by taking advantage of inherent characteristics of the coolant such as its chemical inertness as well as thermodynamic and neutron diffusion properties that permit the use of passive safety systems.

Major accomplishments in the last decade

The co-operation on LFR within GIF was initiated in October 2004, and periodic meetings of the Provisional System Steering Committee (PSSC) were held from March 2005 with participation of representatives from Euratom, Japan, the United States, and experts from the Republic of Korea. The original PSSC prepared a draft system research plan (SRP) which was finalised in October 2010.

In 2009, the GIF Policy Group decided to set up a memorandum of understanding (MOU) for both the LFR and MSR systems, which would provide a more flexible structure for R&D co-operation on those systems in the mid-term. In November 2010, the MOU for collaboration on the LFR system was signed by the signatories of the Joint Research Centre (JRC), for Euratom, and of the Centre for Research into Innovative Nuclear Energy Systems (CRINES) of the Tokyo Institute of Technology, for Japan. In July 2011, the MOU was signed by ROSATOM for the Russian Federation.

The MOU members have decided to accept China, the Republic of Korea and the United States as observers of the PSSC activities. They defined the set of GIF reference LFR systems as outlined above. The members also recognised that a thorough review of the system research plan conceptual framework was necessary to address changes in the status of LFR development. This work, initiated in 2012, is currently ongoing.

One of the main outcomes of the GIF-LFR activities up to 2012 is the subdivision of the research activities into four main areas of interest as follows:

- system integration and assessment (SIA) project;
- system and component design project;
- fuel development project;
- lead technology and materials project.

In Japan, two basic design concepts have been developed: a small LFR called LSPR and a direct contact PBWFR. In parallel, accelerator driven system (ADS) activities have been performed. At present, the experimental activities are concentrated on basic research related to thermal-hydraulics, materials corrosion, oxygen sensor and oxygen control.

The Russian Federation is carrying out design activities for the BREST–300, expected to be in operation after 2020. In parallel activities are carried out also on SVBR–100, a LBE (lead-bismuth eutectic) cooled reactor, based on the previous experience developed for naval propulsion systems.

In Europe, significant activities included projects aimed at the conceptual design of an industrial-size plant, the ELFR, the conceptual design of a 300 MW_{th} demonstrator called ALFRED (Advanced Lead Fast Reactor European Demonstrator), and the activities on

MYRRHA (an accelerator driven lead-bismuth cooled system) designed by SCK-CEN in Belgium.

In the United States, only limited development of the SSTAR has been carried out. However, private investors are considering possible modifications of this design to shorten its implementation phase, and there is some industrial interest in promoting a LFR concept.

In China, the Chinese Academy of Sciences (CAS) started in 2011 a new effort to develop an ADS. The China LEAd-based Reactor (CLEAR) was selected as the reference reactor. The CLEAR development plan includes three phases, the first being a 10 MW_{th} LBE-cooled research reactor (CLEAR-I), with both critical and sub-critical modes of operation, expected to be built before 2020.

In the Republic of Korea, R&D activities on LFR are on-going since 1996. HELIOS, one of the largest LBE test loops, has been operated in both forced and natural circulation conditions of PEACER (Proliferation resistant, Environment friendly, Accident tolerant, Continual, Economical Reactor). The results were published in the framework of the OECD/NEA Task Force on Benchmarking of Thermal-hydraulic Loop Models for Lead-alloy-cooled Advanced Nuclear Energy Systems (LACANES). Advanced corrosion-resistant materials have been developed and tested in both static and dynamic conditions. Small modular reactor designs have been developed to explore their potential as distributed power/heat sources.

R&D objectives

In the near future, the size of testing facilities is expected to increase to reach the dimension needed to test components very close to the full reactor scale. The Russian Federation is looking towards the construction and operation of the LBE-cooled SVBR-100 by 2018 and the lead-cooled BREST-300 by 2020.[4] The BREST plant design includes an integrated facility for fuel reprocessing. The European strategy is focused on the design of ALFRED, a 300 MW_{th} demonstrator.

During the next decade the main R&D efforts will be dedicated to material research, lead corrosion, innovative fuel developments, and design of innovative systems and components. The following main points need further investigation:

- material corrosion;
- core instrumentation;
- fuel handling technology and operation;
- advanced modelling and simulation;
- fuel development: MOX for first core, then MA-bearing fuels;
- actinide management: fuel reprocessing and fabrication;
- ISI&R techniques for opaque medium;
- seismic impact.

With regard to material corrosion, the main strategy is presently based on oxygen control and/or coating/aluminisation (especially for fuel cladding). Oxygen control needs an assessment of oxygen distribution for very large pools. In parallel, the need for lead chemistry control has to be investigated while qualification under irradiation of the proposed solution must be carried out. Material studies will have to address also the

4. The first criticalities of the SVBR-100 and BREST OD-300-100 reactors are expected in December 2018 and December 2020, respectively.

combined effects of the aggressive corrosion environment and the high radiation dose to which core internal materials will be subjected. Special materials for high velocity application (i.e. mechanical primary pumps) will continue to be investigated.

Safety objectives

The GIF safety objectives are addressed for the LFR by exploiting the favourable characteristics of lead coolant in terms of safety, for example high boiling point, chemical inertness and high density. Lead properties allow for reduction of the primary side pressure losses with two main advantages:

- The primary side strong natural circulation permits considerable grace time to cope with the unprotected loss of flow transient.
- Natural circulation is used to introduce fully passive safety systems for removal of the residual heat (both primary and DHR loops working in natural circulation).

DHR systems, based on the concept of active actuation (using locally stored energy) and passive operation, provide the LFR with a very high safety potential over long periods with no requirements for operator actions. This feature of the LFR system has been exploited from the beginning in the design activities and it is a significant part of the LFR strategy for facing extreme accident events. Passive safety systems have also been used to control the reactivity in emergency conditions. Fully passive, redundant and diverse shutdown systems have been designed.

Several other advantages of the lead coolant in terms of safety can be summarised as follows:

- The tendency of lead to retain bulk fission products, thereby reducing the source term to containment, potentially reduces the requirements for an emergency evacuation plan.
- The containment system design pressure can be limited by optimising the water inventory in the secondary system.
- A pool-type LFR with a guard vessel would not suffer loss of primary coolant, even in the event of failure of the reactor vessel. The core would always remain covered and, by design provision, the natural circulation flow path would be maintained.
- No hydrogen generation is expected in a LFR system because of the chemical inertness of the coolant.

The main activities to be performed in the near future, as far as safety is concerned, are related to experimental activities for the demonstration of LFR safety system functionality and performance. Although safety system capabilities have been assessed through numerical simulations, and separate effects tests have been performed, it is expected that licensing authorities will require integral testing at appropriate scale to assess the behaviour of the systems to be licensed. Other experimental testing should be planned to confirm, for example, the expected tendency for fuel dispersion instead of compaction in case of cladding failure. Specific activities are currently planned for steam generator tube rupture tests at small scale with extension to larger scale in the future.

Milestones

The areas for LFR R&D include:

- completion of designs;
- testing of special materials for use in lead environments (corrosion issues);
- fuel studies, including recycling;
- special studies addressing seismic, sloshing and LBE dust/slag formation issues;
- evaluation of long-term radioactive residues from fuel and system activation;

- technology pilot plant/demonstration activities.

As far as GIF activities are concerned, the milestones for the next ten years can be summarised as follows:

- creation of an international lead technology community on the basis of the present GIF MOU;
- preparation of the LFR SA for signature as well as a detailed schedule and scope of activities for the signature of PAs.

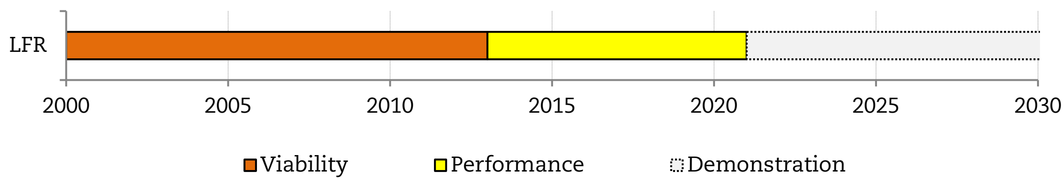
The LFR PSSC is working at present on the following short-term activities:

- overall system research plan for LFR development;
- white paper on safety, based on application of the integrated safety assessment methodology (ISAM) with support from the RSWG;
- safety design criteria for Generation IV lead fast reactor in collaboration with the RSWG.

Regarding the technology aspects of the LFR, detailed designs of LFR-type systems will be completed in the coming years, including further material research and solutions for lead corrosion.

- Lead-cooled (BREST-OD-300) and lead-bismuth-cooled (SVBR-100) reactors are expected after 2020 in the Russian Federation. According to Rosatom's project implementation plan, the first criticality of BREST OD-300-100 and SVBR-100 reactors are expected in December 2020 and December 2018, respectively. All R&D required for the licensing of the above facilities will be accomplished to meet these timelines. Provisions for funds for the construction of both plants have been made in the Russian Federal Target Programme. The LFR demonstration phase is thus expected to begin in 2021 provided that the authorisation to operate the plants is obtained.
- In Europe, the planning of ALFRED strongly depends on the availability of funds.

Timelines for the LFR



Molten salt reactor (MSR)

MSRs can be divided into two subclasses. In the first subclass, fissile material is dissolved in the molten fluoride salt. In the second subclass, the molten fluoride salt serves as the coolant of a coated particle fuelled core similar to that employed in VHTRs. In order to distinguish reactor types, the solid fuel variant is typically referred to as a fluoride salt-cooled high-temperature reactor (FHR).

Between 1950 and 1976, a large MSR development programme was conducted in the United States. Two test reactors were successfully operated: the Aircraft Reactor Experiment (ARE) and the Molten Salt Reactor Experiment (MSRE). A preliminary design of a 1 000 MWe reactor, the Molten Salt Breeder Reactor (MSBR) based on the $^{232}Th/^{233}U$ cycle was completed, and a design was partially developed for a demonstration reactor. These programmes created the basis of the thermal neutron MSR technology. While the concept of an FHR has its origin in the 1970s with the advent of TRISO fuel, FHR development emerged from dormancy only some ten years ago.

MOLTEN SALT REACTOR (MSR) IN THE NEXT DECADE

- A baseline concept: the molten salt fast reactor (MSFR).
- Commonalities with other systems using molten salts (FHR, heat transfer systems).
- Further R&D on liquid salt physical chemistry and technology, especially on corrosion, safety-related issues and treatment of used salts.

In the beginning, MSRs were mainly considered as thermal-neutron-spectrum graphite-moderated reactors. Since 2005, liquid-fuelled MSR R&D has focused on fast-spectrum MSR (MSFR) options combining the generic assets of fast neutron reactors (extended resource utilisation, waste minimisation) with those related to molten salt fluorides as both fluid fuel and coolant (low pressure and high boiling temperature, optical transparency). In addition, because MSFRs exhibit large negative temperature and void reactivity coefficients, MSFR systems have been recognised to have favourable features making them a potential long-term alternative to solid-fuelled fast-neutron systems. However, mastering the technically challenging technology will require concerted, long-term international R&D efforts, namely:

- studying the salt chemical and thermodynamic properties, including with transuranic elements;
- development of efficient techniques for gas extraction from the coolant;
- system design: development of advanced neutronic and thermal-hydraulic coupling models;
- analysis of salt interactions with air or water in case of a severe accident;
- analysis of the accident scenarios (e.g. heat exchanger loss);
- salt reprocessing: lanthanide and actinide reductive extraction tests.

FHRs may offer large-scale power generation while maintaining full passive safety. They can support both high-efficiency electricity generation and high-temperature industrial process heat production. However, while much of the R&D for MSFR is relevant for FHRs, additional developments are required before FHRs can be considered for deployment, in particular on:

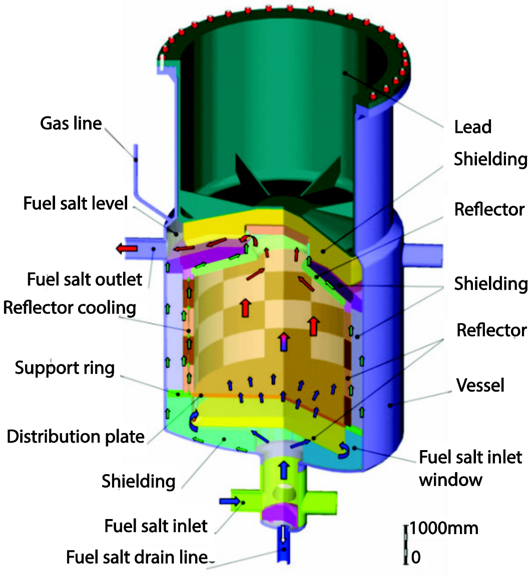
- continuous fiber ceramic composites;
- FHR specific fuel elements and assemblies;
- tritium release prevention technologies;
- redox control technologies.

Major accomplishments in the last decade

The decision to set up a PSSC for the MSR, with the participation of Euratom, France, the Russian Federation and the United States, was taken by the GIF Policy Group in May 2004. France and the JRC, on behalf of Euratom, signed a MOU on 6 October 2010. The United States and the Russian Federation remained observers (the Russian Federation signed the

MOU in late 2013). In mid-2012, China, who participated for the first time in a PSSC meeting, declared its willingness to become a permanent observer.

Figure 2-3: Conceptual MOSART design (left) and MSFR (right)



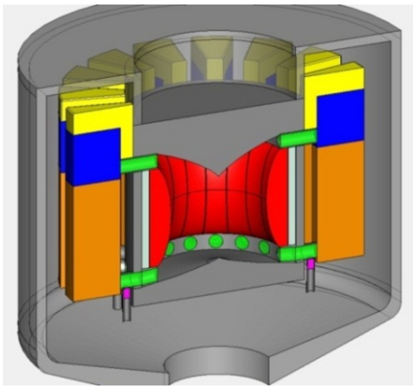


Figure 2-3 shows the conceptual designs of two MSR systems that are under consideration, the molten salt actinide recycler and transmuter (MOSART) and the molten salt fast reactor (MSFR).

In February 2010, white papers on the MOSART and MSFR were presented to the PR&PP working group. In April 2010, MSR safety concerns, issues and benefits were presented to the RSWG.

In 2010, two MSR PSSC meetings were held, the first one in Paris, France, in March and a second one at Oak Ridge National Laboratory (ORNL), Oak Ridge, TN, United States, in September. The ORNL meeting was coupled with a FHR workshop organised by the laboratory and provided the opportunity for PSSC members to have an overview of the efforts and plans of the United States to develop this type of molten salt reactor. The United States is performing limited development of FHR technologies and concepts in its national laboratory system and through university research. At the end of the PSSC meeting, a session was organised and opened to observers from institutions not involved in GIF MSR activities. During this last session, progress and future plans in MSR R&D programmes were presented.

At the 2012 PSSC meeting, China gave an overview of its FHR programme to develop a 2 MW_{th} FHR test reactor through the Chinese Academy of Sciences at the Shanghai Institute of Applied Physics, with planned initial criticality before the beginning of 2016.

Partners of the MSR PSSC are involved in the Euratom-funded ISTC-3749 project, which started in February 2009 and includes as members France (CNRS), Germany (KIT), the Czech Republic (NRI), the United States (ORNL), Euratom (JRC-ITU) and the IAEA.

In 2011, a European project called EVOL (Evaluation and Viability of Liquid Fuel Fast Reactor Systems) started, in parallel with a complementary Russian project named MARS (minor actinide recycling in molten salt) involving Russian research organisations (RIAR, KI, VNIITF and IHTE). The common objective of these two projects was to propose a conceptual design for the best MSFR system configuration by 2012, based on results from physical, chemical and material studies for the reactor core, the reprocessing unit and waste conditioning. It is intended to strengthen the demonstration that the MSFR system can satisfy the goals of Generation IV in terms of sustainability (Th breeder), non-proliferation (integrated fuel cycle, multi-recycling of actinides), resource savings (closed Th/U fuel cycle, no uranium enrichment), safety (no reactivity reserve, strongly negative feedback coefficient) and waste management (actinide burner).

R&D objectives

The renewal and diversification of interests in molten salts led the MSR PSSC in 2008 to shift the R&D aims and objectives promoted in the original Generation IV Roadmap, issued in 2002, to include different applications for fuel and coolant salts.

Following the GIF Experts Group recommendations, six liquid fuelled MSR projects have been proposed:

- materials and components;
- liquid salt chemistry and properties;
- fuel and fuel cycle;
- system design and operation;
- safety and safety system;
- system integration and assessment.

Since that time, the MSFR system operating with a thorium fuel cycle is considered as a baseline concept. Although its potential has been assessed previously, specific technological challenges remain and the safety approach has yet to be established. International collaboration on FHRs is relatively new and a common set of FHR focused projects has yet to be developed. Creation of a consensus set of collaborative FHR development projects is a near-term objective of the MSR PSSC.

Liquid-fuelled MSR viability assessment should address essential R&D issues, in particular in the following areas:

- **Physical-chemical behaviour** of fuel salts and notably coupling between neutronic, thermal-hydraulic and chemistry (including fission products and tritium creation–extraction and more generally reprocessing aspects). The development and qualification of appropriate simulation tools to study normal and incidental situations have been initiated within the EVOL project; the whole task should be completed at the end of the decade.

- **Compatibility of salts with structural materials for fuel and coolant circuits**, as well as fuel processing material development. This topic is directly linked to instrumentation and control of liquid salt because the corrosion of the structural materials by the molten salt is strongly dependent on the redox potential of the salt. Two kinds of studies will be necessary: a) academic lab-scale studies aiming to improve the basic knowledge on the available nickel-based materials family, for example Hastelloy N and new materials (Ni, W, Cr material) which are under development; and b) integrated corrosion studies in loops or demonstrator facilities aiming to test the same materials in realistic thermal and hydraulic conditions. These studies will be spread over time depending upon the means and the availability of the experimental testing loops, and more importantly, the identification of reactor operating temperature in order to establish experimental test parameters.

- **Instrumentation and control of liquid salt**: In-situ measurements need to be developed at a lab-scale in order to control the redox potential of the salt (and limit the corrosion) and to measure the composition of the salt containing fissile and fertile elements.

- **On-site fuel processing**: The different steps of the process (except fluorination for which there is a large experimental feedback from research work carried out in the United States) are being tested in CNRS laboratories (France). The main differences in the reprocessing proposed by CNRS as compared to MSBR treatment are related to the electrochemical steps. These methods have to be examined from both thermodynamic and kinetic points of view. The techniques will also have to be studied from the technological and engineering points of view, as the system moves from the lab-scale to the industrial scale.

- **Liquid salt technologies**: In addition to obvious synergies between liquid MSR and FHR and there is a potential interest in the use of molten salts as intermediate fluid for heat transport in other Generation IV systems (SFR, LFR, VHTR). Liquid salts offer potential advantages regarding, in particular, high volumetric heat capacity, reduced equipment size, absence of chemical exothermal reactions, intermediate loop and power cycle coolants.

Regarding the long-term development of the MSR project (after 2025 or 2030), data will be needed to support a decision on further developments of the molten salt reactor system. Aiming directly for a demonstrator will not be possible for such an innovative system. Therefore, a step-by-step process made up of three main phases will be undertaken.

- **Inactive-salt testing loop**: This pilot plant will aim to improve skills in handling large quantities of molten salt: salt preparation and management, chemical control, accidental leak and freezing management, validation of hydrodynamic models, process instrumentation, heat exchanger and pump testing, gaseous and volatile fission product and particle separation. This loop will also be used for material testing.

- **Active demonstrator without induced fission**: This phase would probably need three different demonstrators to take into account the specificity of molten salt containing heavy metal: a small demonstrator aiming to study corrosion under thermal gradient; another small demonstrator to demonstrate the feasibility of chemical potential control of the molten salt fuel; and a third larger one dedicated to studying the hydrodynamics of the fuel loop for safety demonstration.

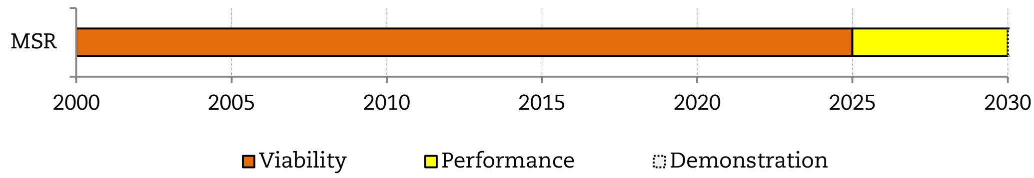
- **Active demonstrator with induced fission:** Two reactors will be needed. The first one, named MONO, would be a full-scale unit with limited power (100 MW_{th}), representative of a single loop of the larger reactor, a 1 000 MW_{th} reactor named DEMO, which would have 16 circulation loops. DEMO is necessary to establish the basis for obtaining the approval of safety authorities: demonstrate the control of the reactor; and test the management of the active salt (drain-out, stop) with its volatile and fission products. It will allow testing of all the structural materials under real conditions. DEMO will have a lower specific power than the commercial reactor and the lower maximum temperatures of its structures will allow the use of already available nickel-based materials. DEMO will have only bubbling treatment on-line, but it will produce representative salt samples to test the off-line chemical reprocessing.

Milestones

The MSR PSSC re-evaluated the liquid fuelled MSR milestones mentioned in the original GIF *Technology Roadmap*, owing to the peculiar and more innovative position of liquid fuelled MSRs among other Generation IV systems, which has led to the milestones summarised in the table below. The development of the FHR is in the initial phase, and therefore, no consensus set of FHR milestones has been developed yet.

Period	Comment
Up to 2011	Scoping and screening phase
2012-2025	Viability phase including the following main topics: • Management and salt control (2012-2014); • Confirmation of bubbling efficiency (2014-2015); • Heat exchanger viability (2015-2017); • Validation of reprocessing flow sheets at laboratory scale; • Definition of safety analysis methodology and specification of accident scenarios.
After 2025-2030	Adequate data will be available to support a decision on the further development of molten salt reactor system. Performance phase: the performance of all developed systems will be evaluated.

Timelines for the MSR



Sodium-cooled fast reactor (SFR)

The SFR uses liquid sodium as the reactor coolant, allowing a low-pressure coolant system and high-power-density operation with low coolant volume fraction in the core.

Because of advantageous thermo-physical properties of sodium (namely high boiling point, heat of vaporisation, heat capacity and thermal conductivity) there is a significant thermal inertia in the primary coolant. A large margin to coolant boiling is achieved by design, and is an important safety feature of the SFR.

While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water and requires a sealed coolant system. Further development of passive safety approaches and validation of their performance are key research objectives in the coming years.

Much of the basic technology for the SFR has been established in former fast reactor programmes, and was further confirmed by the Phénix end-of-life tests in France, the lifetime extension of BN-600 in Russia, the restart and success of core confirmation tests of Monju in Japan and the start-up of an experimental fast reactor in China. France, Japan and Russia are designing new SFR demonstration units for near-term deployment; China, the Republic of Korea and India are also proceeding with their national SFR projects.

Plant size options under consideration by GIF range from small, 50 to 300 MWe, modular reactors, to larger plants, up to 1 500 MWe. The outlet temperature is 500-550°C for the options, which allows using the materials that were developed and proven in prior fast reactor programmes.

SODIUM-COOLED FAST REACTOR (SFR) IN THE NEXT DECADE

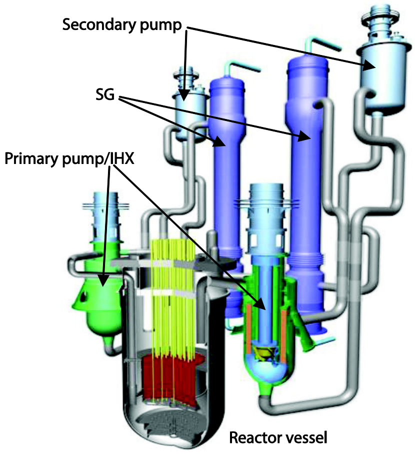
- Three baseline concepts (pool, loop and modular configurations).
- Several sodium-cooled reactors operational or under construction (e.g. in China, India, Japan and Russia).
- Develop advanced national SFR demonstrators for near-term deployment (France, Japan and Russia); proceed with respective national projects in China, Korea and India.
- In the coming years, the main R&D efforts will be concentrated on:
  - safety and operation (improving core inherent safety and I&C, prevention and mitigation of sodium fires, prevention and mitigation of severe accidents with large energy releases, ultimate heat sink, ISI&R);
  - consolidation of common safety design criteria;
  - advanced fuel development (advanced reactor fuels, MA-bearing fuels);
  - component design and balance of plant (advanced cycles for energy conversion, innovative component design);
  - used fuel handling schemes and technologies;
  - system integration and assessment;
  - implementation of innovative options;
  - economic evaluations, operation optimisation.

Currently there are three options for SFR configurations: pool, loop and modular. Therefore, the current GIF SFR system research plan (SRP) includes (see Figure 2-4):

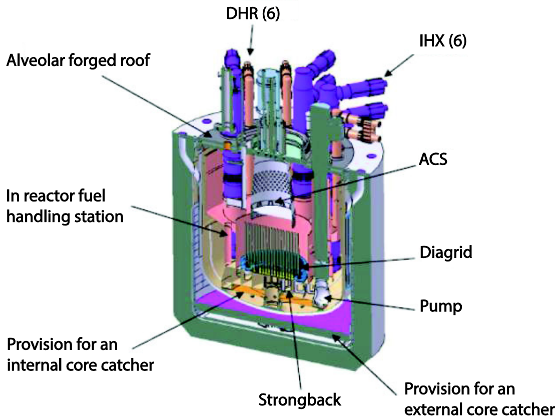
- A large size (600 to 1 500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially MA-bearing fuels, supported by a fuel cycle with advanced aqueous processing at a central location serving a number of reactors.
- An intermediate-to-large size (300 to 1 500 MWe) pool-type reactor with oxide or metal fuel.
- A small size (50 to 150 MWe) modular type reactor with metal-alloy fuel (uranium-plutonium-MA-zirconium), supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor.

The SFR closed fuel cycle enables regeneration of fissile fuel and facilitates management of minor actinides. However, this requires that recycle fuels would be developed and qualified for use. The two primary fuel recycling technology options are 1) advanced aqueous and 2) pyrometallurgical processing. A variety of fuel options are being considered, with mixed oxide the lead candidate for advanced aqueous recycle and mixed metal alloy the lead candidate for pyrometallurgical processing.

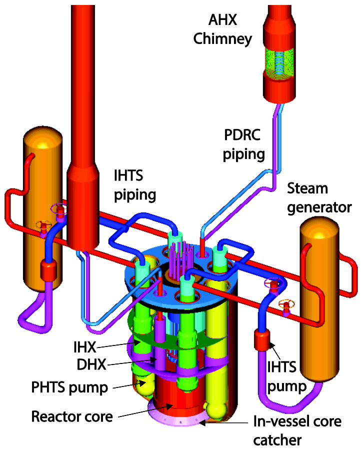
Figure 2-4: Four design tracks of the GIF SFR (as of 2011)



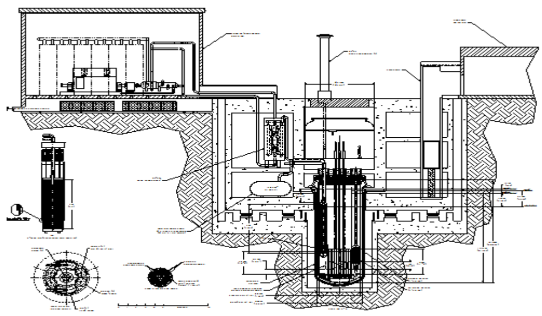
Large loop



Large loop



Intermediate-to-Large loop



Small modular

Important safety features of the Generation-IV SFR system 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 and alternative fluids are considered for the power conversion system to achieve high performance in terms of thermal efficiency, safety and reliability.

The SFR is an attractive energy source for nations that desire to make the best use of limited nuclear fuel resources and manage nuclear waste by closing the fuel cycle. Its fast neutron spectrum enables full actinide recycling and greatly extends uranium resources compared to thermal reactors. The SFR technology is more mature than other fast reactor technologies and thus is deployable in the very near-term for actinide management. With innovations to reduce capital cost, the SFR also aims to be economically competitive in future electricity markets.

Major accomplishments in the last decade

A significant amount of R&D has been performed by the SFR signatories, in particular through their respective national SFR projects.

The most recent activities related to **safety and operation** include the evaluation of various codes, modelling of early molten fuel discharge, multi-dimensional calculation, and sodium void reactivity effect. The performance of components of decay heat removal systems, in-service inspection methodology and some issues of radioactive elements transportation were also investigated, and some experimental studies of sodium boiling, fuel pin failure modes and analysis of metal fuel pin disruption tests were performed using data from the Transient Reactor Test Facility (Idaho National Laboratory) as well as from other experiments.

The **development of advanced fuels and actinide management** has attracted considerable attention in participating countries. Based on technical evaluation using the available knowledge on fast reactor fuels (as well as specific tests currently being conducted on MA-bearing fuels), it appears that both oxide and metal fuels could be straightforward options to meet the performance goals. Fuel investigations now include not only the homogeneous but also the heterogeneous route for MA transmutation. In 2012, preparation and implementation of irradiation tests as well as post-irradiation examinations have continued for MA-bearing oxide, metal, nitride and carbide fuels. Development work on MA-bearing fuel fabrication processes and the simplified pelletising method in hot cell by remote operation has been performed. The promising candidates for the core materials are ferritic/martensitic and ODS steels. Fabrication and characterisation of ferritic/martensitic cladding tubes have continued. Cladding liner material has been developed to mitigate fuel-to-cladding chemical interaction. A fuel pin with ODS cladding is being prepared for irradiation in Joyo.

The recent theoretical and experimental R&D results from the **component design and balance-of-plant** project (CDBOP) showed the viability of the supercritical CO_2 Brayton energy conversion cycle, and allowed for the investigation of the performance of key components and processes in both steady-state and transient conditions. In addition, fundamental data on interactions of sodium with CO_2, as well as on corrosion of austenitic and ferritic steels by supercritical CO_2, have been obtained.

The issue of improving steam generators has recently been added (since 2011) to the CDBOP. Several technical solutions and improvements have been suggested, in particular double-walled steam generator tubes. Safety approaches to the monitoring of sodium/water reaction in the steam generators have been investigated. Specific instrumentation techniques (hydrogen-diffusion membrane, hydrogen-metre performance and passive acoustic detection systems) have also been developed.

Approaches for ultrasonic in-service inspection techniques have been developed and tested, for example scanning through multiple steel walls from outside the reactor vessel, and a new linear strip waveguide sensor technique for examination of the upper part of the core, as well as eddy-current testing of double-walled steam generator tubes. Data have been obtained that will enable the formulation of a leak-before-break methodology for 9Cr1Mo (Grade 91) ferritic steel for use in intermediate sodium circuit piping and components. Information has been shared on lessons learnt from refurbishments and repairs of the Joyo, Monju and Phénix SFRs.

R&D objectives

SFR R&D activities and SFR SA signatory plans have been organised into several projects. The scope and objectives of the R&D to be carried out in these projects are summarised in the sections below.

Safety and operation (SO) project

In the field of safety, experiments and analytical model developments are being performed to address both passive safety and severe accident prevention and mitigation. Options for safety system architecture are also being investigated. This R&D covers reactor operation, inspection, maintenance and technology testing campaigns in existing SFRs (Monju, BN-600 and CEFR). The R&D needs in inherent safety features, severe accident mitigation and development of safety analysis tools are listed below.

Inherent safety features:

- safety principles (reactivity feedback, core design goals, balanced safety approach);
- passive or self-actuated shutdown system;
- decay heat removal options (short and long term);
- reactor transient behaviour and testing experience;
- severe accident prevention.

Severe accident mitigation:

- experiments on fuel melting behaviour;
- specialised fuel assembly design for severe accident behaviour (e.g. sacrificial inner duct);
- core catcher options.

Safety analysis tools:

- validation and uncertainty quantification;
- severe accident modelling;
- probabilistic safety assessment techniques.

Accumulation of decommissioning experience will also be very important in demonstrating the possibility of achieving the economic goal.

Advanced fuel (AF) and Global Actinide Cycle International Demonstration (GACID) projects

Fuel-related research aims towards developing high burnup minor-actinide (neptunium, americium, curium) bearing fuels as well as claddings and wrappers capable of withstanding high neutron doses and temperatures. It includes research on remote fuel fabrication techniques for fuels that contain minor actinides and possibly traces of fission products, as well as performance under irradiation of fuels, claddings and wrappers. Candidates under consideration are: oxide, metal, nitride and carbide for fuels; alternate fast reactor fuel forms and targets for special applications (e.g. operation at higher temperature); and ferritic/martensitic & ODS steels for core materials.

Regarding waste reduction capability, some tests at experimental scale were performed to assess the feasibility of MA transmutation in reducing the quantity, toxicity and half-life of ultimate radioactive waste. Two concepts have to be explored in more detail, with special focus on americium transmutation:

- Transmutation in homogeneous mode, with up to a few per cent of americium in the core, allowing the demonstration of breakeven-breeding (the quantity of MA that is transmuted equals the quantity that is produced in the core).
- Transmutation in heterogeneous mode, with 10% to 20% of americium in the blankets.

Collaborative R&D activities have been initiated with the objective of demonstrating, on a significant scale, that fast reactors can indeed manage the actinide inventory to satisfy the Generation IV criteria of safety, economy, sustainability, proliferation resistance and physical protection. The project consists of: MA-bearing test-fuel fabrication, material properties measurements, irradiation behaviour modelling, and pin-scale irradiations in Joyo; licensing and fuel assembly irradiation tests in Monju; and post-irradiation examination, as well as transportation of MA raw materials and MA-bearing test fuels.

Component design and balance-of-plant (CDBOP) project

Research on component design and balance of plant has the objective of enhancing SFR system performance in order to reduce the capital cost per unit electrical power and the cost of electricity generation. Primary research and development activities include work on: advanced components and technologies to enhance the economic competitiveness of the plant; development of advanced in-service inspection instrumentation and repair methods using different approaches and technologies; research and development on advanced energy conversion systems (the supercritical CO_2 Brayton cycle and nitrogen gas Brayton cycle) to improve plant economics and eliminate sodium-water reactions; and innovation in advanced, high-reliability Rankine cycle steam generator designs and related instrumentation to enhance the robustness against sodium-water reaction and efficiency. In addition, the importance of the experience and lessons learnt from the operation and upgrading of SFRs is recognised and summarised.

System integration and assessment (SIA) project

The objectives of the SIA project are to: review and integrate the Generation-IV R&D results from a system design perspective in order to help define and refine requirements for the overall SFR concept R&D; review and integrate results from the R&D projects in order to ensure consistency; and periodically assess the system options and design tracks for conformance with the GIF technology goals and other SFR-specific requirements.

The SIA Project Arrangement has been completed and is expected to be signed by all SFR signatories in 2014. The project plan identifies several integration tasks:

- Identification of general classes of Generation-IV SFR system options and specific design tracks being pursued by the project members.
- Identification of contributions from interesting trade studies on the design and performance of SFR systems that could be valuable inputs to the project.
- Definition of a comprehensive list of Generation-IV SFR research and development needs to identify possible overall or synergy opportunities for the technical projects.

Operability and economy

In addition to the advances targeted in the above projects, significant advances in the operability and economic performance of SFR are achievable, thereby increasing plant availability up to current and future industry standards and lowering operating costs. These additional advances can be achieved by:

- reduced duration of fuel loading outage, through improvement of fuel handling systems, for example;
- increased fuel burnup and cycle length;
- Improved instrumentation for detection and localisation of sodium leaks;
- improved ISI&R capabilities, which play a key role in SFR operation (due to the opaqueness and elevated temperature of the sodium coolant), through advanced instrumentation (ultrasonic techniques, robotics);
- extended plant lifetime to 60 years, comparable to current Generation III/III+ reactors, through:
  - development and qualification of materials with enhanced resistance to ageing degradation;

– development of improved inspection and diagnostic capabilities for verifying fitness of materials and structures for continued service.

Codes and standards – such as the RCC-MRx code in Europe or the new ASME Section III, Division 5, which provides design and construction rules for mechanical components such as vessel, piping, and support structures (core excluded) – are key for reactor design and regulatory review and may require revisions to allow the use of advanced materials. In particular, some R&D to extend current codes and data (e.g. for highly irradiated materials) will be required in support of SFR design and licensing.

Other safety objectives

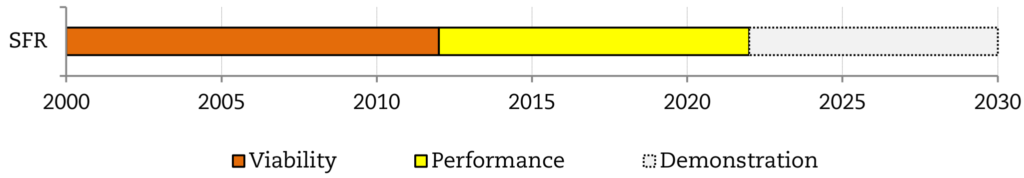
Additional R&D on safety issues highlighted by the Fukushima Daiichi accident is foreseen in the work plan for the coming years. A primary focus on the following issues is anticipated:

- robust and highly reliable systems for adequate cooling of safety-relevant components and structures;
- geometric stability of the SFR core in case of a strong earthquake and assurance of reliable performance of the control rods;
- seismic-resistant design of the spent fuel pools and fuel-handling devices;
- integrity of the primary circuit and its cooling;
- design features aimed at excluding the risk of flooding of the reactor building;
- effective options for dealing with severe accidents.

Milestones

Period	Comment
2012-2015	The expected start-up of new sodium-cooled reactors in the world (namely BN-800 in Russia in 2014) and completion of the design of ASTRID (France) will contribute to the advancement of R&D during the performance phase. **Safety**: • Apply safety design criteria (SDC) to the three baseline concepts. • Demonstrate enhancement of SFR safety assurance. **Advanced fuels**: • Preliminary evaluation of advanced fuels (identification of advanced fuel options among oxide, metal, carbide and nitride candidate fuel types for subsequent detailed evaluation). Preliminary selection of advanced fuel(s) for SFR at the end of 2015. • Evaluation of MA-bearing fuels (selection of advanced fuel concepts for subsequent evaluation of high-burnup performance, based on evaluation of their potential to provide effective utilisation and destruction of minor actinides in the SFR). **Component design and balance of plant**: • Development of advanced in-service inspection instrumentation and repair methods. • Development of advanced energy conversion systems for improving plant economics and eliminating sodium-water reactions. • Development of advanced high-reliability steam generators and related instrumentation. **System integration and assessment**: • Application of the EMWG methodology to various project options.
2015-2022	**Safety**: • Studies of innovative design and safety systems, in particular for severe accident. • Consolidation of the safety design criteria (SDC). **Advanced fuels**: • High-burnup fuel performance evaluation: Final selection by around 2020 of advanced fuel design(s) for subsequent demonstration.
After 2022	Advanced, Generation IV SFR are expected to enter the demonstration phase.

Timelines for the SFR



Supercritical-water-cooled reactor (SCWR)

SCWRs are high temperature, high-pressure, light water reactors that operate above the thermodynamic critical point of water (374°C, 22.1 MPa). The reactor core may have a thermal or a fast-neutron spectrum, depending on the core design. The concept may be based on current pressure-vessel or on pressure-tube reactors, and thus may use light water or heavy water as a moderator. Unlike current water-cooled reactors, the coolant will experience a significantly higher enthalpy rise in the core, which reduces the core mass flow for a given thermal power and increases the core outlet enthalpy to superheated conditions. For both pressure-vessel and pressure-tube designs, a once-through steam cycle has been envisaged, omitting any coolant recirculation inside the reactor. As in a boiling water reactor, the superheated steam will be supplied directly to the high pressure steam turbine and the feed water from the steam cycle will be supplied back to the core. Thus the SCWR concepts combine the design and operation experience gained from hundreds of water-cooled reactors with the experience from hundreds of fossil-fired power plants operated with supercritical water (SCW). In contrast to some of the other Generation IV nuclear systems, the SCWR can be developed step-by-step from current water-cooled reactors.

SUPERCRITICAL-WATER-COOLED REACTOR (SCWR) IN THE NEXT DECADE

- Two baseline concepts (pressure-vessel-based and pressure-tube-based).
- R&D over the next decade will include:
  - advancing conceptual designs of baseline concepts and associated safety analyses;
  - more realistic testing of materials to allow final selection and qualification of candidate alloys for all key components;
  - out-of-pile fuel assembly testing;
  - qualification of computational tools;
  - first integral component tests and start of design studies for a prototype;
  - in-pile tests of a small scale fuel assembly in a nuclear reactor.
- Definition of a SCWR prototype (size, design features) for decisions to be taken in the coming years.

SCWRs have unique features that offer many advantages as compared to state-of the-art water-cooled reactors:

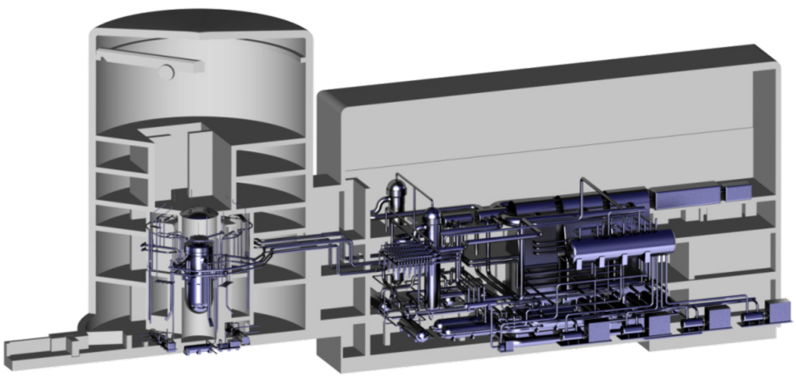
- Thermal efficiency can be increased to 44% or more, as compared to 34-36% for current reactors.
- Reactor coolant pumps are not required. The only pumps driving the coolant under normal operating conditions are the feed water pumps and the condensate extraction pumps.

- The steam generators used in pressurised water reactors and the steam separators and dryers used in boiling water reactors can be omitted since the coolant is superheated in the core.
- Containment, designed with pressure suppression pools and with emergency cooling and residual heat removal systems, can be significantly smaller than those of current water-cooled reactors.
- The higher steam enthalpy allows for a decrease in the size of the turbine system and thus a reduction in the capital costs of the conventional island.

These general features offer the potential of lower capital costs for the given electric power of the plant and of better fuel utilisation, offering a clear economic advantage compared with current water-cooled reactors.

However, there are several technological challenges associated with the development of the SCWR, in particular the need to validate transient heat transfer models (for describing the depressurisation from supercritical to sub-critical conditions), qualification of materials (namely advanced steels for cladding), and demonstration of the passive safety systems.

Figure 2-5: Conceptual design of the HPLWR, a supercritical-water-cooled reactor with a net electric power of 1 000 MW, developed in Europe



General challenges in SCWR development arise from the higher core outlet temperature and the higher enthalpy rise of the coolant in the core, relative to current water-cooled reactors. These challenges are:

- Non-uniformities of local power and coolant mass flow rate in the core may cause hot spots due to the larger enthalpy rise of the coolant. As with fossil-fired power plants, the problem may be overcome with multiple heat-up steps plus intermediate coolant mixing which, however, adds more complexity to the core design.
- The higher coolant temperature results in higher fuel cladding temperatures. Zirconium alloys can no longer be used for the fuel cladding and must be replaced by steels or other high temperature material. This impacts the fuel burnup and the peak cladding temperature.
- A boiling crisis in the core can be physically excluded as SCW is a single phase fluid. Nevertheless, the fuel cladding may still overheat if the design limits of heat flux or coolant mass flux are exceeded.
- Proven safety systems known from advanced boiling water reactors may be employed, but the safety strategy must be changed from control of coolant

inventory to control of coolant mass flow rate due to the absence of recirculation inside the reactor.

- The large density variation within the core could lead to instability and subsequently large neutronic variation and high fuel cladding temperature.
- Operation under SCW conditions introduces unique water chemistry challenges related to water radiolysis and corrosion product transport. A chemistry control strategy must be developed to define relevant material test conditions.
- If a fast neutron spectrum is envisaged, the requirement of a negative void reactivity coefficient in any core position limits the achievable positive breeding gain to negative values.

The following section summarises how well these challenges have been addressed within the last ten years.

Major accomplishments in the last decade

Pre-conceptual core design studies for a core outlet temperature higher than 500°C have been performed in Japan, assuming either a thermal or a fast neutron spectrum, as summarised by Oka et al.[5] Both options are based on a coolant heat-up in two steps with intermediate mixing below the core. Additional moderator for a thermal neutron spectrum is provided by feed water inside water rods. The fast-spectrum option uses zirconium-hydride (ZrH_2) layers to minimise hardening of the neutron spectrum in case of core voiding. A pre-conceptual design of safety systems for both options has been studied with transient analyses.

A pre-conceptual plant design with 1 700 MW net electric power based on a pressure-vessel-type reactor has been studied by Yamada et al.[6] and has been assessed with respect to efficiency, safety and cost. The study confirms the target net efficiency of 44% and estimates a cost reduction potential of 30% compared with current pressurised water reactors. Safety features are expected to be similar to advanced boiling water reactors.

A pre-conceptual design of a pressure-vessel-type reactor with a 500°C core outlet temperature and 1 000 MW electric power has been developed in Europe (see Figure 2-5), as summarised by Schulenberg and Starflinger.[7] The core design is based on coolant heat-up in three steps. Additional moderator for the thermal neutron spectrum is provided in water rods and in gaps between assembly boxes. The design of the nuclear island and balance of plant confirms results obtained in Japan, namely an efficiency improvement of up to 43.5% and a cost reduction potential of 20 to 30% compared with the latest boiling water reactors. Safety features, as defined by the stringent European Utility Requirements, are expected to be met.

Canada is developing a pressure-tube-type SCWR concept with a 625°C core outlet temperature and a pressure of 25 MPa.[8] The concept is designed to generate 1 200 MW electric power (a 300 MW concept is also being considered). It has a modular fuel channel configuration with separate coolant and moderator. A high-efficiency fuel channel is incorporated to house the fuel assembly. The heavy-water moderator is in direct contact

5. Oka Y., S. Koshizuka, Y. Ishiwatari, A. Yamaji (2010), *Super Light Water Reactors and Super Fast Reactors*, Springer, France.
6. Yamada K., S. Sakurai, Y. Asanuma, R. Hamazaki, Y. Ishiwatari, K. Kitoh (2011), *Overview of the Japanese SCWR concept developed under the GIF collaboration*, Proc. ISSCWR-5, Vancouver, Canada, March 13-16, 2011.
7. Schulenberg T., J. Starflinger (2012), *High performance light water reactor – design and analyses*, KIT Scientific Publishing.
8. Yetisir M., W. Diamond, L.K.H. Leung, D. Martin and R. Duffey (2011), *Conceptual Mechanical Design for A Pressure-Tube Type Supercritical Water-Cooled Reactor*, Proc. 5th International Symposium on Supercritical Water-cooled Reactors, Vancouver, Canada, March 13-17, 2011.

with the pressure tube and is contained inside a low-pressure calandria vessel. In addition to providing moderation during normal operation, it is designed to remove decay heat from the high-efficiency fuel channel during long-term cooling, using a passive moderator cooling system. A mixture of thorium oxide and plutonium is introduced as the reference fuel, which aligns with the GIF position paper on thorium fuel. The safety system design of the Canadian SCWR is similar to that of the ESBWR. However, the introduction of the passive moderator cooling system coupled with the high-efficiency fuel channel could reduce significantly the core damage frequency during postulated severe accidents such as large-break loss-of-coolant or station black-out events.

Pre-conceptual designs of three variants of pressure-vessel-type supercritical reactors with thermal, mixed and fast neutron spectrum have been developed in Russia[9,10], which joined the SCWR System Arrangement in 2011.

Outside of the GIF framework, two conceptual SCWR designs with thermal and mixed neutron spectrum cores have been established by research institutes in China under the framework of Chinese national R&D projects from 2007-2012, covering some basic research projects on materials and thermo hydraulics, core/fuel design, main system design (including the conventional part), safety systems design, reactor structure design and fuel assembly structure design. The related feasibility studies have also been completed, and show that the design concept has promising prospects in terms of overall performance, integration of design, component structure feasibility and manufacturability.

Prediction of heat transfer in supercritical water can be based on data from fossil-fired power plants as discussed by Pioro et al.[11] Computational tools for more complex geometries like fuel assemblies are available but still need to be validated with bundle experiments. System codes for transient safety analyses have been upgraded to include SCW, such as depressurisation transients to subcritical conditions. Flow stability in the core has been studied numerically. As in boiling water reactors, flow stability can be ensured using suitable inlet orifices in fuel assemblies.

A number of candidate cladding materials have been tested in capsules, autoclaves and recirculating loops up to 700°C at a pressure of 25 MPa. Stainless steels with more than 20% chromium are expected to have the required corrosion resistance up to a peak cladding temperature of 650°C. More work is needed to develop alloys suitable for use at the design peak cladding temperatures of 850°C for the Canadian SCWR concept. Further work is also needed to better identify the coolant conditions that lead to stress corrosion cracking. It has been shown that the creep resistance of existing alloys can be improved by adding small amounts of elements such as zirconium, as reported by Kaneda et al.[12] In the longer term, the experimental oxide dispersion-strengthened steel alloys offer an even higher potential, whereas nickel-based alloys that are being considered for use in ultrasupercritical fossil-fired plants are less favourable for use in SCWRs due to their high neutron absorption and associated swelling and embrittlement.

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9. Ryzhov S.B., V.A. Mokhov, M.P. Nikitenko, A.K. Podshibyakin, I.G. Schekin, A.N. Churkin, Advanced designs of VVER reactor plant, The 8th International Topical Meeting on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-8), October 10-14, 2010, Shanghai, China, Paper N8P0184.
10. Ryzhov S.B., P.L. Kirillov, et al. (2011), *Concept of a Single-Circuit RP with Vessel Type Supercritical Water-Cooled Reactor*, Proc. ISSCWR-5, Vancouver, Canada, March 13 16, 2011.
11. Pioro I.L., R.B. Duffey (2007), *Heat transfer and hydraulic resistance at supercritical pressures in power engineering applications*, ASME Press.
12. Kaneda J., S. Kasahara, F. Kano, N. Saito, T. Shikama, H. Matsui (2011), *Material development for supercritical water-cooled reactor*, Proc. ISSCWR-5, Vancouver, Canada, March 13 16, 2011.

Key water chemistry issues have been identified by Guzonas et al.[13]; predicting and controlling water radiolysis and transport of corrosion products (including fission products) remain the major R&D areas. In this regard, the operating experience using nuclear steam reheat at the Beloyarsk nuclear power plant in Russia is extremely valuable.

R&D objectives

Pre-conceptual designs of a pressure-vessel-type SCWR have been developed in Japan and Europe. The development of the pressure-tube-type SCWR is ongoing in Canada and is scheduled for completion in 2015. Striving to achieve high thermal efficiency, comparable to that of advanced fossil-fired power plants, would lead to higher peak cladding temperature for the Canadian concept. This requires additional research on new materials and novel solutions such as coatings or surface modification. All design concepts will be assessed with respect to the criteria of the *Technology Roadmap*. Outside of GIF, China will be focusing on completing the basic design of its SCWR prototype.

Thus far, R&D of SCWR designs and technologies have closely followed the *Technology Roadmap* developed in 2002. What should be the R&D focus within the next 5 to 10 years?

Having a better understanding of the expected design(s) of a SCWR, R&D within the **next five years** must include more realistic testing of materials, thermal hydraulics and core components. The SCWR roadmap for this time frame includes:

- Out-of pile tests of a small scale, electrically heated, SCWR fuel-assembly simulator, at relevant operating conditions. Suitable test facilities are available in China, Japan and Russia.
- Final selection and qualification of candidate alloys for all key components. This includes in-pile tests of cladding alloys at SCWR conditions, to study the effect of irradiation, water radiolysis and corrosion product deposition. As materials selection is being finalised, issues such as joining and manufacturability must be addressed.
- Qualification of computational tools for SCWR applications (such as coupled nuclear/thermal-hydraulic phenomena, fuel performance, and reactor-system analyses) would require a large amount of experimental data and the construction of integral test facilities.

For the **ten year perspective**, the first integral component tests are envisaged and the design of a prototype can be started. Over this time frame, the roadmap includes:

- Test of a small-scale fuel assembly in a nuclear reactor, which is considered to be the next mandatory step before a prototype SCWR can be built. This test has never been done at supercritical conditions.
- Decisions about a SCWR prototype – its size, design target and potential location – can be made in five years, once the pre-conceptual design phase has been completed. Based on these decisions, a joint international project for the basic design of a prototype could be considered within the next ten years, but the first out-of-pile component tests associated with this design would likely be beyond this time frame.

Safety objectives

Generation III nuclear power plants are already being built with design features that do not require any evacuation of the population around these plants, even under the

13. Guzonas D., F. Brosseau, P. Tremaine, J. Meesungnoen, J.-P. Jay-Gerin (2012), "Water chemistry in a supercritical water–cooled pressure tube reactor", *Nuclear Technology*, Vol. 179.

worst-case conditions of a core melt-down. As for the Generation IV nuclear systems, the SCWR will be licensed only if it fulfils at least these stringent requirements. More specifically, the Fukushima Daiichi accident demonstrated the need for passive residual heat removal over long periods and the SCWR should be designed accordingly.

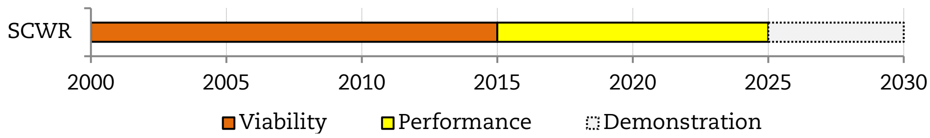
Milestones

The SCWR technology development is ongoing with a focus on the GIF objectives of improved safety, proliferation resistance, economics and sustainability. The SCWR R&D is progressing according to the 2009 System Research Plan with minor delays. For example, a fuel qualification test is being designed and licensed. Design and construction of a prototype or demonstration unit is planned to be included in the next SCWR SRP.

The key milestones for the next ten years are the following:

- up to 2015: Out-of pile, small-scale fuel assembly test; cladding material selection; qualification of computational tools; pre-conceptual design completion;
- 2017: Decision about a SCWR prototype;
- 2017 to 2022: In-pile, small-scale fuel assembly test.

Timelines for the SCWR



Very-high-temperature reactor (VHTR)

The VHTR is a next step in the evolutionary development of high-temperature gas-cooled reactors. It is a graphite-moderated, helium-cooled reactor with thermal neutron spectrum. It can supply nuclear heat and electricity over a range of core outlet temperatures between 700 and 950°C, and potentially more than 1 000°C in the future. The reactor core of the VHTR can be a prismatic-block type such as the Japanese HTTR, or a pebble-bed type such as the Chinese HTR-10. For electricity generation, a direct cycle with a helium gas turbine system directly placed in the primary coolant loop, or, at the lower end of the outlet temperature range, an indirect cycle with a steam generator and a conventional Rankine cycle can be used. For nuclear heat applications such as process heat for refineries, petrochemistry, metallurgy and hydrogen production, the heat application process is generally coupled with the reactor through an intermediate heat exchanger (IHX), the so-called indirect cycle. The VHTR can produce hydrogen by using thermochemical processes (such as the sulfur-iodine [S-I] process), combined thermochemical and electrolysis (such as the hybrid sulfur process), high temperature steam electrolysis (HTSE), or from heat, water and natural gas by applying the steam reformer technology. A reference VHTR system that produces hydrogen is shown in Figure 2-6.

A 600 MW_{th} VHTR dedicated to hydrogen production can yield over 2 million normal cubic metres per day. The VHTR can generate electricity with high efficiency, ~50% at 950°C, compared with 47% at 850°C. Co-generation of heat and power makes the VHTR an attractive heat source for large industrial complexes. Because of its excellent safety characteristics, the VHTR can be deployed in refineries and petrochemical industries to substitute large amounts of process heat at different temperatures, including hydrogen generation for upgrading heavy and sour crude oil.

The high degree of safety of the HTGR/VHTR that was demonstrated by AVR, THTR, Peach Bottom and Fort Saint Vrain reactors continues to be a strong motivation for coupling the system to industrial processes. Further demonstrations of the safety performance for both the prismatic and pebble bed concepts, at HTTR and HTR-10, emphasise the benefit of the strong negative temperature coefficient of reactivity, the high heat capacity of the graphite core, the large temperature increase margin, and the robustness of TRISO fuel in producing a reactor concept that does not need off-site power to survive multiple failures or severe natural events as occurred at the Fukushima Daiichi nuclear station.

Major accomplishments in the last decade

While the original approach for VHTR at the start of the Generation IV programme focused on very high outlet temperatures and hydrogen production, current market assessments have indicated that electricity production and industrial processes based on high temperature steam that require outlet temperatures of (700-850°C) already have a great potential for applications in the next decade and also reduce the technical risk associated with higher outlet temperatures. As a result, over the past decade, the focus of design studies has moved from higher outlet temperature designs such as GT-MHR and PBMR to lower outlet temperature designs such as HTR-PM in China and the NGNP in the United States.

VERY-HIGH-TEMPERATURE REACTOR (VHTR) IN THE NEXT DECADE

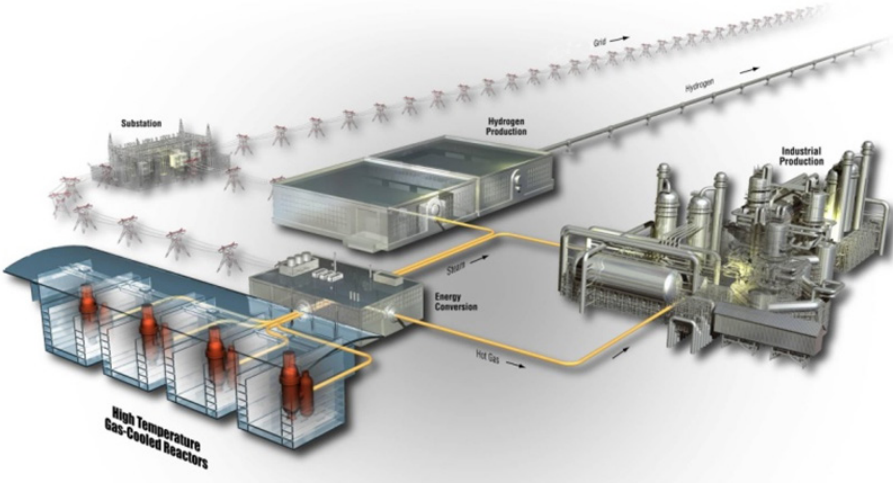
- In the near future, the main focus will be on VHTR with core outlet temperatures of 700-950°C.
- Further R&D on materials and fuels should enable higher temperatures up to above 1 000°C and a fuel burnup of 150-200 GWd/tHM.
- Development of further approaches to set up high-temperature process heat consortia for end-users interested in prototypical demonstrations.
- Development of the interface with industrial heat users – intermediate heat exchanger, ducts, values and associated heat transfer fluid:
  - Advancing H2 production methods in terms of feasibility and commercial viability to better determine process heat requirements for this application.
  - Regarding nuclear safety:
    - Verify the effectiveness and reliability of the passive heat removal system.
    - Confirm fuel resistance to extreme temperatures (~ 1 800°C) through testing.
    - Proceed with the safety analyses of coupled nuclear processes for industrial sites using process heat.

A variety of different configurations of the reactor and balance of plant have been examined across the world. Today, the direct Brayton power conversion cycle is being pursued less aggressively in favour of an indirect Rankine cycle because of its lower technological risk and increased flexibility in terms of working fluid and mission of the reactor (electricity, process heat, and co-generation).

In the near term, the work should focus on the needs of lower-temperature (from 700 to 950°C) demonstration projects. Future operation at higher temperatures (up to above 1 000°C) requires development of high temperature alloys, qualification of new graphite type and development of composite ceramic materials. Also, improvement and validation of computational fluid dynamics and system models through theoretical and experimental benchmarks would be needed.

The VHTR evolves from HTGR experience and extensive international databases that have supported its development. The basic technology for the VHTR has been well established in former HTGR plants, such as Dragon, Peach Bottom II, AVR, THTR, and Fort Saint Vrain and is being advanced in concepts such as the HTR-PM and NGNP. The ongoing 30 MW_{th} HTTR project in Japan is intended to demonstrate the feasibility of reaching outlet temperatures up to 950°C coupled to a heat utilisation process, and the HTR-10 in China has demonstrated the inherent safety performance with electricity production and co-generation at a power level of 10 MW_{th}. The former projects in Germany and the United States provide data relevant to VHTR development.

Figure 2-6: A 4-pack modular VHTR for process heat, hydrogen production and electricity generation



Demonstrating the viability of the VHTR core requires meeting a number of significant technical challenges. Fuels and materials must be developed that:

- permit an increase of the core-outlet temperatures from around 800°C to more than 1 000°C for the entire plant lifetime;
- permit the maximum fuel temperature under accident conditions to reach levels approaching 1 800°C;
- permit maximum fuel burnup of 150-200 GWd/tHM;
- avoid power peaking and temperature gradients in the core, as well as hot streaks in the coolant gas;
- limit structural degradation from air or water ingress.

Process-specific R&D gaps need to be filled to adapt the chemical process and the nuclear heat source to each other with regard to temperatures, power levels and operational pressures. Heating of chemical reactors by helium is a departure from current industrial practice and needs specific R&D and demonstration. The development of an intermediate heat exchanger, ducts, valves and associated heat transfer fluid is needed to deliver process heat to many of the chemical processes.

The viability of using nuclear process heat to produce hydrogen needs further study. Any contamination of the product will have to be avoided. Development of heat exchangers, coolant gas ducts and valves will be necessary for isolation of the nuclear island from the production facilities. This is especially the case for isotopes like tritium, which can easily permeate metallic barriers at high temperatures.

Over the past decade, *significant* advances have been made in the key technologies necessary to deploy a VHTR. Further R&D is needed, as discussed in the following section.

R&D objectives

Fuels and materials

Qualification of TRISO fuel: There is a need to develop and demonstrate the performance of TRISO fuel at high operating temperature (up to 1 250°C), high burnup (up to 200 GWd/tHM) and under off-normal conditions (1 600-1 800°C). In the United States and China, fabrication activities are demonstrating that UO_2 and UCO-TRISO fuel can be fabricated to the high quality/low defect levels necessary for the concept. Irradiation testing of spheres manufactured in China has demonstrated performance as good as, and in some cases better than, the historical German experience. Work in the United States has demonstrated that UCO-TRISO fuel is capable of burnups approaching 200 GWd/tHM at temperatures of ~1 250°C. Accident safety testing of this UCO-TRISO fuel has demonstrated a high degree of robustness for hundreds of hours at 1 600, 1 700 and 1 800°C. A United States fuel vendor has been established that is capable of producing either UO_2 or UCO-TRISO fuel in compact form. Work is planned to continue in both China and the United States to complete qualification of these fuels within the next decade.

ZrC coatings for TRISO fuel: Above a fuel operating temperature of 1 200°C, new coating materials such as ZrC and/or improved coating techniques have been considered. Use of ZrC in VHTRs enables an increase in power density and total power level with the same coolant outlet temperature. It displays greater resistance to chemical attack by the fission product palladium. Under accident conditions, historical data suggest that the ZrC-TRISO fuel may be more robust than traditional SiC-TRISO fuel. Unexplained anomalous historical results, the susceptibility of ZrC to oxidation, and recent data suggesting significant thermomechanical material property degradation under accident conditions, however, have limited the interest in pursuing ZrC among VHTR researchers. Both the historical and more recent fabrication data on ZrC indicates it is more difficult to fabricate than SiC. Furthermore, the outstanding behaviour of UCO-SiC-TRISO fuel may be sufficient to meet the high temperature irradiation requirements of the VHTR.

Carbon/Carbon (C/C) and SiC/SiC composite components: C/C and/or SiC/SiC composites may be needed for control rod sheaths, especially at the higher outlet temperatures anticipated in a prismatic-block-core VHTR, so that the control rods can be inserted into the high-temperature areas of the core. Promising ceramics such as fiber-reinforced ceramics, sintered alpha silicon carbide, oxide-composite ceramics, and other composite materials are also being developed for other industrial applications needing high-strength, high-temperature materials. Work continues around the world on C/C and SiC/SiC composites for a variety of nuclear applications. Irradiation stability of some SiC/SiC composites has been demonstrated up to 70 dpa. Novel fabrication routes, development of hermetic composites, irradiation testing and establishment of design rules to enable use in a nuclear system are the focus of R&D on composites over the next ten years.

Pressure vessel materials: Design efforts in the United States for VHTRs at higher outlet temperatures have indicated that engineering solutions will allow the use of traditional LWR pressure vessel steels (A508/A333) for the VHTR. Efforts in the United States have focused on developing the data needed to allow use of LWR pressure vessel steels in a VHTR environment. These data are being incorporated into the ASME Code for use in gas-cooled reactors.

Heat utilisation systems materials: Internal core structures and cooling systems, such as intermediate heat exchangers, hot gas ducts, process components and isolation valves, that are in contact with hot helium, can use current metallic materials up to a core outlet temperature of about 700 to 800°C. Efforts in the United States have focused on developing the data needed to extend Alloy 800H for use up to 850°C and Alloy 617 for use up to 950°C. Within the next four years, all of the data needed to codify these materials will be provided to the ASME. Recent tests have indicated tritium permeation will be less of an issue than originally thought. The low concentrations of hydrogen compounds in the VHTR cause less permeation than predicted by current theory.

Reactor systems

Core internals: Core internal structures containing the fuel pebbles or blocks are made of high-quality graphite. The performance of such graphite for core internals has been demonstrated in gas-cooled pilot and demonstration plants, but recent improvements in the manufacturing process of industrial graphite have shown improved oxidation resistance and better structural strength. Irradiation tests are needed to qualify components using advanced graphite or composites to the fast neutron fluence limits of the VHTR. Current irradiation testing in the United States and Europe is qualifying a number of current grades of nuclear graphite from the major graphite vendors. Irradiation and post-irradiation examination are underway. Preliminary data suggest the performance is acceptable. Full qualification of current grades of graphite is expected in the next decade.

Balance of plant

The VHTR balance of plant is determined by the specific application, which can be thermochemical processes, dedicated electricity production or co-generation. A variety of process heat applications (e.g. co-generation of steam and electricity, production of process heat for a chemical ethylene plant, coal-to-diesel fuel production, production of fertiliser) were studied as part of the NGNP programme in the United States to understand the technical needs of the specific process, to evaluate the potential of reactor and balance of plant configurations to meet the requirements, and to provide an economic assessment of using a VHTR in the specific application.

In these process heat applications, all components have to be developed and qualified for use at temperatures between 700 and 950°C, and more than 1 000°C in the future, in the VHTR environment. Failure mechanisms such as creep, fretting and ratcheting have to be studied in detail, precluded by design, and demonstrated in component tests. Specific components such as helical-tube steam generators, IHX, isolation valves, hot gas ducts with low heat loss, steam reformers, and process-related heat exchangers have to be developed for use in the modular VHTR, which mainly uses only one loop. This leads to much larger components than formerly developed and a new design approach is to modularise the component itself.

The steam cycle is technically ready. It combines the high efficiency of VHTR and maturity of steam turbines used in fossil-fired power plants. High quality steam for electricity production or co-generation is a low-risk and high-performance option for VHTR. Design, manufacturing, operation, and in-service inspection of helical-tube steam generators require more efforts and operation experience feedback. Supercritical steam can further increase the efficiency of the VHTR steam cycle.

The Brayton cycle option has good prospects for VHTR electricity production in the future. Some key components such as recuperator, helium turbine and IHX require R&D efforts beyond work done already in many countries. Some new concepts, such as the indirect supercritical CO_2 turbine cycle, provide more options.

Low pressures are necessary or preferable for many processes. Alternate coolants, such as molten salts, for the intermediate loop should be adapted where needed. Process-specific components will also need to be tested. Other applications will require different components such as helium-heated steam crackers, distiller columns and superheaters.

Over the past decade, different balance-of-plant configurations and different IHX designs have been evaluated. The most promising candidate IHX designs have been tested at small scale. A number of different coolants have been evaluated and the technical challenges with each identified. Some large-scale configuration testing is anticipated as part of the HTR-PM project in China, but for process heat applications it remains to be accomplished once a detailed conceptual design is established, given the cost of such testing.

Hydrogen production subsystem: Second-generation solid-oxide fuel cells are demonstrating much lower degradation than first generation cells. The technology has matured enough to support near-term commercial scaling and deployment by industry. However, the development and qualification of the hydrogen generation process subsystem is needed. Medium-scale demonstrations of both the high-temperature steam electrolysis (HTSE) and thermochemical SI processes were conducted. Also, HTSE testing at high pressure has been completed, a key step for its integration into an energy conversion system. Integrated testing of the SI system revealed some issues associated with the process interfaces. Work continues to bring that system to the maturity level required for integration into a power system. The hybrid copper-chloride thermo-chemical process and the hybrid sulfur process are also being researched by many member countries. Beyond experimental testing, significant analytical capabilities/methods were developed that can be used to evaluate hydrogen production in a VHTR plant (or other nuclear system).

Fuel cycle

Disposal of once-through fuel and graphite: The VHTR uses a once-through, LEU (<20% ^{235}U) fuel cycle. Like LWR spent fuel, VHTR spent fuel could be disposed of in a geologic repository or conditioned for optimum waste disposal. The current VHTR coated-particle fuel encapsulates the spent-fuel fission products in a form that is extremely resistant to leaching in a final repository. However, as removed from the reactor, the fuel includes large quantities of graphite, and research is required to define the optimum packaging form of spent VHTR fuels for long-term disposal. Also, radiation damage will require replacement of some graphite core components every four to ten years.

Recycling of LWR and VHTR spent fuels in a symbiotic fuel cycle can achieve significant reductions in waste quantities and radiotoxicity because of the VHTR's ability to accommodate a wide variety of mixtures of fissile and fertile materials without significant modification of the core design. This flexibility was demonstrated in the AVR test reactor in Germany and is a result of the ability of gas-cooled reactors to decouple the optimisation of the core cooling geometry from the neutronics.

For an actinide-burning alternative, specific Pu-based driver fuel and MA-bearing transmutation fuel MA would have to be developed. These fuels may benefit from the R&D on SiC and ZrC coatings mentioned above but will need more R&D than LEU fuel.

Analytical and experimental efforts over the past decade in Europe and the United States have evaluated both direct disposal and fuel recycling for VHTR materials. Routes to deconsolidate fuel pebbles or compacts have been established and head-end processes developed to work with either aqueous or pyroprocessing recycling. Preliminary work suggests that recycled graphite can be an acceptable feedstock for re-fabricated graphite. Experimental and analytical work on using a VHTR as an actinide burner continues in many countries.

Economics

Detailed economic studies have been performed in the United States for both electricity production and process heat applications. The results suggest that VHTRs are competitive with new LWRs for electricity production. For process heat and co-generation applications, the VHTR can be competitive with conventional combined-cycle gas turbine systems producing steam and electricity when the cost of natural gas is greater than $8/MMBtu. Carbon taxes may reduce this threshold. Currently, the cost of natural gas varies widely across the world. Thus, the economic viability of VHTR process heat and co-generation applications depends largely on the financial and regulatory climate in each country. The inherent safety features of VHTR may benefit its economic performance indirectly.

Safety objectives

Passive decay heat removal (DHR) systems have been designed to facilitate operation of the VHTR, with a final goal of simple operation and transparent safety concepts. Demonstration tests are planned to verify the system's passive characteristics and to show that its safety margins are sufficient.

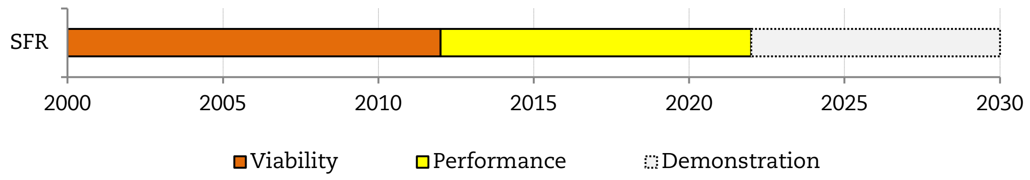
Analysis and demonstration of the inherent safety features of the VHTR are needed, and could potentially draw on development and demonstration of earlier gas-cooled reactors. Additional safety analysis is necessary with regard to nuclear process heat applications in an industrial environment. Design-basis and beyond-design-basis accident analyses for the VHTR will need to include phenomena such as chemical attack of graphitic core materials, typically by either air or water ingress. Adequacy of existing models will need to be assessed, and new models may need to be developed and validated.

Experimental demonstration and validation of key features are underway with large in-vessel and ex-vessel experiments in the United States. Over the next 5 years, these experiments are anticipated to evaluate depressurised conduction cooldown events in a VHTR and demonstrate the role of the reactor cavity cooling system in the passive safety response of the plant. HTTR in Japan has been subjected to a series of operational transients to provide extensive data on the response of a VHTR to upset conditions. These data will provide robust validation of system tools. Operation data and safety experiments on HTR-10 can be used for validation of pebble-bed reactor analysis tools.

Milestones

- The construction and operation of HTR-PM (start-up expected around 2015) will contribute positively to the advancement of R&D:
  - In the near future, the main focus will be on VHTR with outlet temperatures between 700 and 950°C.
  - Thermal hydraulic safety experiments (namely on depressurisation) in the coming years are needed for improvement and validation of computational fluid dynamics and system models through theoretical and experimental benchmarks.
  - Qualification of UCO-TRISO fuel (1 250°C; burnups of 150 to 200 GWd/tHM).
  - Qualification of new grades of graphite for VHTR use.
  - Qualification of Ni alloys for high temperatures (between 800 and 950°C).

Timelines for the VHTR



- Further R&D in the longer term with the objective of operation up to 1 000°C will include:
  - R&D on advanced materials (e.g. SiC/SiC composites) and advanced fuels (ZrC-TRISO) will allow temperatures from around 800°C to more than 1 000°C and fuel burnup of 150-200 GWd/tHM.
  - Development of the interface with industrial heat users – intermediate heat exchangers, ducts, valves and associated heat transfer fluids. Development of high temperature alloys, qualification of new graphite types and development of composite ceramic materials.

Chapter 3. Ten-year objectives for methodology working groups

Economic Modeling Working Group (EMWG)

The Economic Modeling Working Group (EMWG) was established to develop a methodology to assess the innovative nuclear systems against the GIF economic goals, namely:

- to have a life cycle cost advantage over other energy sources (i.e. to have a lower levelised unit cost of energy over their lifetime);
- to have a level of financial risk comparable to other energy projects (i.e. to involve similar total capital investment and capital at risk).

The innovative nuclear systems within Generation IV need unique tools for their economic assessment because their characteristics are less well-known than those of earlier nuclear power plants. An integrated economic model is necessary to compare various Generation IV technologies, as well as to answer optimal configuration questions, such as which fuel cycle is most suitable in different parts of the world and what are the optimal deployment ratios.

EMWG accomplishments

The EMWG developed a methodology that has been validated through sample calculations for both Generation III and Generation IV systems, and has been used by academia in several publications. The methodology consists of the following:

- *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems, Rev. 4* (GIF/EMWG/2007/004)[1];
- G4ECONS software package;
- User's manual for G4ECONS Version 2.0 (GIF/EMWG/2007/005).

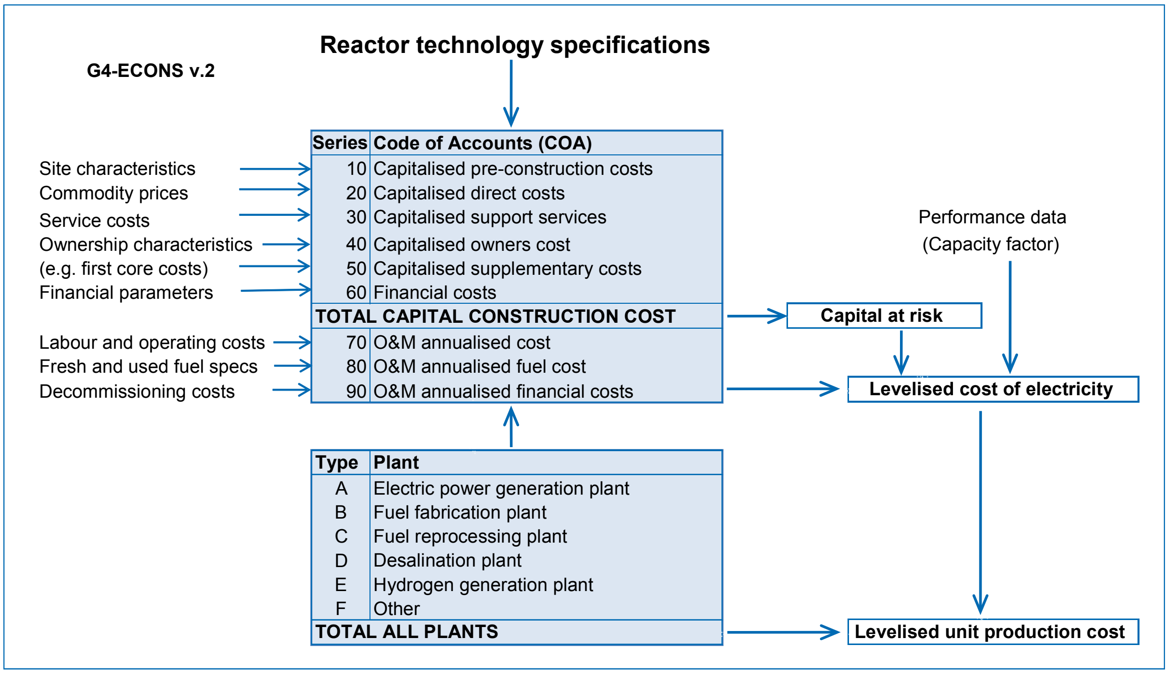
The cost estimating guidelines provide a uniform set of assumptions and a uniform code of accounts in developing cost estimates for advanced nuclear energy systems. It discusses the development of all relevant life cycle costs for Generation IV systems, including the planning, research, development, demonstration (including prototype), deployment, and commercial stages. These guidelines form the basis for the software model G4ECONS, a spreadsheet tool used to calculate the levelised unit cost of energy products, including heat and electricity, and, for applications such as hydrogen production or desalination, of non-electrical products (e.g. hydrogen or potable water). An additional module was developed to calculate the cost of fuel cycle services. The structure of G4ECONS is shown in Figure 3-1. The combination of software and guidelines facilitates the development of consistent and comprehensible cost estimates by the system development teams.

In September 2007, the EMWG, with the agreement of the GIF Experts and Policy Groups, released the methodology for public as well as GIF application. A CD containing the complete methodology is available from the OECD/NEA. To date, over 166 copies of the CD have been provided upon request from GIF entities, various IAEA working groups,

1. Available at www.gen-4.org/gif/jcms/c_9509/tools.

several universities and a number of consulting companies. Several papers demonstrating the implementation of the GIF cost estimating methodology were presented by EMWG members at the GLOBAL 2009 Conference, held in Paris, France, and the GIF symposia held in Paris in 2009 and in San Diego, United States, in 2012. Several members of the EMWG presented papers in various international meetings and published articles in scientific journals based on the outcomes from the work carried out within the group. The EMWG has also developed training presentations for the application of the methodology.

Figure 3-1: Generation IV Excel-based cost calculation of nuclear systems (G4ECONS)



Current activities

The EMWG is discussing the possibilities of collaboration with the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). The initial focus will be on comparison of economic analyses of Generation III systems obtained by the methodologies developed within GIF and INPRO.

Enhancement of the G4ECONS software has been suggested to better facilitate the analysis of heterogeneous fuel cycles that may be proposed for fast reactor systems and particularly for actinide-management applications. Several studies were begun to demonstrate an approach for estimating the cost of actinide-management services.

Applications of the GIF methodology by groups and institutions outside GIF were reviewed to gain feedback and experience that may be helpful to GIF in the future. Improvements in the methodology are under consideration based on the users' experience and progress in the development of Generation IV systems. Economic literature review is an ongoing task so the latest information may be made available to the GIF teams.

Future efforts

Over the next two to three years, the EMWG will release a new version of the G4ECONS code to allow better treatment of heterogeneous fuel cycles, provide capability for the treatment of cost uncertainty and improve the user-friendliness of the software. The proposed additions include cash-flow analysis for multi-unit construction at a single site. The new version will allow also sensitivity analyses with respect to uncertainties in unit cost elements. The additional information provided by these calculations could be used in more detailed analyses from both an investor and policy-maker perspective. This could increase the utility of G4ECONS and expand its potential user base.

An updated users guide will accompany the new code version. After completion of the code and users guide updates, the EMWG will revise the *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* document, including a review of the cost data aiming at updating the information. This new version of G4ECONS may be applied and tested on advanced Generation III/III+ LWRs and SMRs.

Over the next ten years, the EMWG will continue to monitor the development of Generation IV systems and designs and further improve the methodology accordingly. Monitoring of the methodology applications will continue, as well as training and assistance to the system steering committees (SSC) as they begin applying the methodology to their specific systems. The EMWG will continue to report to the Experts Group and Policy Group as requested, and explore new areas of co-operation through GIF/INPRO interface exchanges.

ECONOMIC MODELING WORKING GROUP (EMWG) IN THE NEXT DECADE

- Over the next two to three years, the EMWG will release a new version of the G4ECONS cost estimating code with advanced capabilities.
- The *Cost Estimating Guidelines* will be reviewed after the *User Guide* updates:
  - Over the next 10 years, the EMWG will continue to monitor the progress of Generation IV systems economic analyses and further improve the methodology consistent with these designs.

Proliferation Resistance and Physical Protection Working Group (PRPPWG)

GIF Technology Roadmap (2002)

When the *Technology Roadmap* was issued in 2002, it was envisioned that the R&D programme for PR&PP would be conducted in three areas: 1) safeguards and physical protection technology R&D for each GIF system, 2) formulation of PR&PP criteria and metrics, and 3) evaluation of the criteria and metrics. The PRPPWG was established in late 2002 with a charter that covered items 2 and 3. Specifically, the working group was charged with developing a methodology for the systematic evaluation of Generation IV energy systems with respect to proliferation resistance and physical protection. Overall, the methodology would enable comparative evaluation of the performance of different systems (or options for a given system) against the GIF PR&PP goals. The working group would also determine the measure (or measures) for expressing proliferation resistance and physical protection, and develop an evaluation approach that is comprehensive and quantitative, to the extent possible.

The PRPPWG was not given a specific mandate with respect to item 1 but its work has been clearly linked to safeguards and physical protection. As outlined in the 2002 *Technology Roadmap*, each GIF design would support R&D on, *inter alia*, material deployed, potential vulnerabilities, protective barriers, safeguards approaches, potential misuse, material protection and on control and accounting for each step in the fuel cycle. While each GIF design has not yet formally addressed all nine tasks given in the 2002 Roadmap for PR&PP R&D, there has been interaction between each of the system steering committees and the PRPPWG on the status of each design with regard to PR&PP R&D and a report prepared jointly by the PRPPWG and the SSCs was approved by the Policy Group in 2011 (see discussion below).

PRPPWG accomplishments

In a succession of revisions beginning in 2004, the PRPPWG has developed a methodology for PR&PP evaluation of all GIF systems. Measures and associated metrics were included in each revision. Consensus was achieved among all GIF participating countries and related organisations (IAEA, EU) and revision 6 of the methodology report was approved by GIF for open distribution[2] in 2011.

The methodology was developed and demonstrated by use of a hypothetical "example sodium fast reactor". Workshops with GIF designers and other stakeholders, to familiarise them with the methodology and to understand their needs for the design process, were held in the United States (2005), Italy (2006), Japan (2007, 2011), the Republic of Korea (2008) and Russia (2012). A joint meeting between the PRPPWG and the IAEA/INPRO was held at IAEA in October 2013.

PROLIFERATION RESISTANCE AND PHYSICAL PROTECTION WORKING GROUP (PRPPWG) IN THE NEXT DECADE

- As new and innovative designs for nuclear energy systems are developed through GIF (and other possible fora), the PR&PP methodology approach will be essential to incorporate good design principles for proliferation resistance and physical protection into these new designs.
- Enable safeguards by design: Robust safeguards are essential to the PR&PP characteristics of all of the emerging GIF designs.
- Assist GIF system developers in introducing PR&PP concepts into their design work.

Starting in 2007, the PRPPWG and the six SSCs conducted a series of workshops on the PR&PP characteristics of their respective designs and identified areas in which R&D is needed to further include such characteristics and features in each design. A common template was developed to collect, in a systematic way, GEN IV design concept information and PR&PP features and issues. This work culminated in reports on each of the six design concepts, written jointly by the PRPPWG and the respective SSC. An overall report, compiling the six contributions, was approved by GIF for open distribution in 2011. The intent is to generate preliminary information about the PR&PP merits of each system and to recommend directions for optimising their PR&PP performance.

The overall report captures the current salient features of the GIF system design concepts that impact their PR&PP performance. It identifies crosscutting studies to assess PR&PP design or operating features common to various GIF systems; and it suggests beneficial characteristics of future nuclear energy systems, beyond the nuclear island and power conversion system, that should be addressed in subsequent GIF activities.

The PRPPWG has co-ordinated closely with the IAEA since its inception and an IAEA representative on the group has been contributing to its work on a continuing basis. Moreover, there continues to be a close association between the PRPPWG and the IAEA/INPRO effort on proliferation resistance. A representative of the PRPPWG participates on a regular basis in the annual GIF/INPRO co-ordination meeting.

National programmes have adapted the PR&PP methodology to their specific needs and interests. In the United States, the methodology has been used to evaluate alternative spent fuel separations technologies. In Canada, there has been a safeguards-by-design application of the PR&PP methodology. The methodology is also being applied to provide PR consideration within a European R&D project on a sodium fast reactor. In

2. See www.gen-4.org/gif/jcms/c_9365/prpp for reports and relevant references.

Belgium, outside the GIF programme, there has been an application to the accelerator-driven MYRRHA project. In Japan, the PR&PP methodology has been used for non-proliferation study for the FaCT project.

The PRPPWG has co-ordinated its efforts with the GIF Risk and Safety Working Group and a joint meeting between the groups took place in Russia in October 2012, in conjunction with the 23rd meeting of the PRPPWG.

A summary of the work of the PRPPWG over the past decade appears in a special issue on PR&PP of the ANS journal *Nuclear Technology*, Volume 179, published in July 2012.

Current situation assessment

In the area of proliferation resistance, there has already been considerable interaction between GIF and INPRO, beginning with a comparison of the respective methodologies of the two organisations so as to understand how prospective users could benefit from each and from a joint application of the two approaches. Some members of GIF have participated in INPRO projects and other IAEA projects in nuclear energy and safeguards, which has provided a useful catalyst to further co-operation. Moreover, the annual meetings between GIF and INPRO have provided an excellent forum for information exchange and for defining future collaborative efforts. Work that has been carried out under the INPRO/PROSA project has been monitored for potential application in the GIF programme.

There are ongoing and planned efforts, both in national programmes and internationally by IAEA and Euratom, to promote and implement the concept of safeguards by design (SBD) in the nuclear facility design process. The IAEA has efforts underway on SBD and has issued general guidelines in 2013. Facility-specific guidance documents are expected to be published in 2013-14. As noted above, the IAEA also has the PROSA programme underway, which will be relevant to SBD and the PR&PP methodology. In this context, the interaction with designers has identified a need for simplified scoping PR evaluations.

There is an increased emphasis worldwide on the development and deployment of small modular reactors. Since some of the GIF designs are in the SMR category, it will be important to maintain cognizance of issues and developments as they are relevant to PR&PP aspects.

It will be important to maintain awareness of lessons learnt from the Fukushima Daiichi accident analyses for their potential relevance to PR&PP issues.

A committee of the US National Academies has studied how methodologies for proliferation risk assessment relate to the needs and questions of policy makers in this area. Their findings and recommendations were issued in June 2013 and are currently being evaluated by the sponsoring organisations in the United States. The PRPPWG will assess in due course the relevance of the study and of the US sponsors' responses to its recommendations on future work of the group.

Future PRPPWG activities

Working with SSCs on maturing their designs: As new and innovative designs are developed for nuclear energy systems through GIF (and other fora), the PR&PP methodology approach will be essential to incorporating good design principles for proliferation resistance and physical protection into new emerging and viable concepts.

If the GIF sponsors in the various participating countries wish to advance the utilisation of the PR&PP methodology in the design process, the next major step for joint activity between the SSCs and the PRPPWG should be to designate one or two concept designs for an in-depth pilot study. This would involve applying the PR&PP methodology to the development of a model of the design and would be a follow-on effort to the initial joint studies between the PRPPWG and the SSCs that have been described above. The model would be rather high-level and attempt to capture the broad features of the design in terms of expressing its robustness for PR&PP characteristics. The pilot study would include participation by nuclear energy system designers as specified by the SSCs and members of the PRPPWG who would bring modelling expertise to the collaboration. In addition, subject matter experts in safeguards and physical protection would be needed to provide specific context for the development of the models. This study could fit well within the scope of one of the GEN IV SIA projects.

In the longer term, when the results and insights from these pilot studies become available, other GIF design concepts would also engage in such model development with the assistance of the PRPPWG. The overall benefit would be to introduce PR&PP aspects early in the design process in order to cost-effectively provide for safeguards and security before the design has fully matured and thus to avoid costly retrofits. This would ultimately be a useful approach to minimising project risk for the emerging GIF concepts.

Enabling SBD: Robust safeguards are essential to the PR&PP characteristics of all of the emerging GIF designs. In conjunction with the PRPPWG effort with the SSCs, the PRPPWG will maintain cognizance of technology developments and good practices that would foster SBD in the GIF designs.

SMRs: To the extent that it is relevant to GIF designs, the PRPPWG will maintain awareness of developments in this area and enable the incorporation of robust PR&PP features in SMRs.

IAEA/INPRO: The PRPPWG will continue to co-ordinate with the IAEA in areas of mutual interest.

Continued interaction between the PRPPWG and the other GIF crosscutting groups: Co-ordination with the RSWG and with the EMWG should be pursued to assure effective implementation of the three methodology group approaches in GIF design activities.

Risk and Safety Working Group (RSWG)

Past accomplishments

In accordance with its terms of reference, the primary objective of the Risk and Safety Working Group is to promote a harmonised approach on safety, risk and regulatory issues in the development of Generation 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 Generation IV systems. In 2008, the RSWG published its thoughts and recommendations on these and related topics in a report entitled *Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems*. Within this document, the RSWG achieved a consensus regarding some of the safety-related attributes and characteristics that should be reflected in Generation IV nuclear systems.

Some of the major areas in which consensus has been reached include:

- a non-prescriptive cohesive safety philosophy applicable to all Generation IV systems;
- objectives and ways to meet the potential safety improvements;
- basic principles for an approach applicable to the design and the assessment of innovative systems including the ways to assess the adequacy of the defence-in-depth principle application and especially to address the treatment of severe plant conditions;

- role of passive features;
- role of the probabilistic safety assessment (PSA) and other existing analysis approaches, and the need for developing innovative indicators and tools.

The more recent work of the RSWG has turned to focus primarily on an integrated framework for assessing risk and safety issues for use throughout the Generation IV technology development cycle. In 2011, the RSWG published the second report entitled *An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems*. It is envisioned that the ISAM will be used in three principal ways:

- Throughout the concept development and design phases with insights derived from ISAM serving to actively drive the course of the design evolution. In this application, ISAM is used to develop a more detailed understanding of design vulnerabilities and resulting contributions to risk. Based on this detailed understanding of vulnerabilities, new safety provisions or design improvements can be identified, developed and implemented relatively early.
- Selected elements of the methodology will be applied at various points throughout the design evolution to yield an objective understanding of risk contributors, safety margins, effectiveness of safety-related design provisions, sources and impacts of uncertainties, and other safety-related issues that are important to decision makers.
- ISAM can be applied in the late stages of design maturity to measure the level of safety and risk associated with a given design relative to safety objectives or licensing criteria. In this way, ISAM will allow evaluation of a particular Generation IV concept or design relative to various potentially applicable safety metrics or "figures of merit". This *post facto* application of ISAM will be useful especially for decision makers and regulators who require objective measures of safety for licensing purposes or to support certain late-stage design selection decisions.

RISK AND SAFETY WORKING GROUP (RSWG) IN THE NEXT DECADE

- In 2008, the RSWG published the *Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems* – a consensus regarding some of the safety-related attributes and characteristics that should be reflected in Generation IV systems.
  - Future work: Provision for application of the integrated safety assessment methodology (ISAM) in the development of Generation IV systems.
  - A number of detailed analyses and "lessons learnt" investigations will be performed, especially as related to the Fukushima Daiichi accident.

It is specifically intended that this methodology be used neither to dictate design requirements or compliance with quantitative safety goals nor to constrain designers in any other way. The sole intent is to provide a methodology that yields useful insights into the nature of safety and risk of Generation IV systems, thereby allowing meaningful evaluations of Generation IV concepts for the attainment of the Generation IV safety objectives.

The integrated methodology consists of five distinct analytical tools and stages, which are structured around the last one, the probabilistic safety assessment. The tools/stages are the following:

- qualitative safety requirements/characteristic review;
- phenomena identification and ranking table;

- objective provision tree;
- deterministic and phenomenological analyses;
- probabilistic safety assessments.

Each tool will be used to answer specific kinds of safety-related questions to differing degrees of detail and at different stages of design maturity. By providing specific tools to examine relevant safety issues at different points in the design evolution, ISAM as a whole offers the flexibility to allow a graded approach to the analysis of technical issues of varying complexity and importance. The methodology is well integrated, as evidenced by the fact that results from each analysis tool support or relate to inputs or outputs of other tools. Although individual analytical tools can be selected for individual and exclusive use, the full value of the integrated methodology is derived from using each tool in an iterative fashion and in combination with the others throughout the development cycle.

Current activities and near-term future efforts

Late in 2010, responding to direction from the GIF Chair, a task force was formed to define and articulate safety design criteria (SDC) for SFR systems. The task force comprises representatives of the RSWG, the SFR system steering committee and other interested representatives of the GIF SFR community. The work of the task force began in 2011 and aimed at establishing the SDC on Generation IV SFR by the end of 2012. The overall goal of the SDC task force activity is “harmonisation” of enhanced safety features common to all Generation IV SFR systems. The objectives of the SDC task force are to establish the reference criteria of the designs of safety structures, systems and components that are specific to the SFR system, to clarify the criteria systematically and comprehensively when the concept developers apply the GIF safety approach and to use codes and standards with the aim of achieving the safety goals of the Generation IV reactor systems. Once the SDC are defined for the SFR system, the next step would be to define those criteria for other GIF systems, in particular the VHTR. In the wake of the events at Fukushima Daiichi in March 2011, much of the work of the task force aims to take account of relevant lessons learnt from these events. For example, an increased emphasis on the consideration of external events is a likely outcome. Upon the completion of SDC development, efforts will be aimed at developing lower level recommendations and guidance for design (i.e. safety design guideline: SDG) in the next stage, focusing on the important safety attributes of SDC. Those SDGs would provide recommendations on actions, conditions or procedures for meeting the SDC.

Work is currently continuing on the series of safety “white papers” for the six Generation IV systems. These white papers, joint work products of the RSWG and the six SSCs, present high-level information about safety-related design issues and phenomena associated with each of the six Generation IV system concepts, as well as early thinking about safety assessment for these systems. These papers will facilitate the clarification of the safety characteristics and system-specific safety issues, accelerate the usage of the ISAM and enhance the relationship between the RSWG and each system SSC/PMB. These white papers will be maintained and will evolve in parallel with the progress of the six Generation IV design concepts and their associated R&D programmes.

Future efforts

In feedback received from the GIF EG, the RSWG has been asked to work toward the provision of increasingly detailed guidance for application of the ISAM in the development of Generation IV systems. This will form an important focus for the future work of the RSWG. Practical, specific guidance on ISAM application is expected to support the SSCs as they use the ISAM to take into account safety issues while developing their respective systems. Likewise, the RSWG will be looking for opportunities to directly support more systematic and detailed applications of the ISAM by the six SSCs.

A number of detailed analyses and “lessons learnt” investigations will be performed. It is highly likely that the results of some of these analyses will have implications for the development and deployment of Generation IV nuclear systems. Furthermeore, it is likely that some of these analyses will have implications for how to ensure that the scope and depth of Generation IV safety assessments are carried out in a sufficiently robust way as to understand system vulnerabilities under a very broad range of accident conditions, including some that formerly might have been deemed so unlikely as to preclude their consideration. With the benefit of such detailed analyses, safety philosophies and assessment methods, including the ISAM, can then be modified or updated based on lessons that will be learnt from the Fukushima Daiichi experience. In its future programme of work, the RSWG intends to closely monitor events at the Fukushima Daiichi site, as well as lessons learnt from those events, in order to evaluate how those lessons can shape its approach to assessing and ensuring the safety of Generation IV systems, with an emphasis on deterministic severe accident analysis methodologies. These lessons and findings will be used to update the *Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems* and the integrated safety assessment methodology (ISAM).

In addition, the RSWG will maintain its interfaces with the IAEA, INPRO, Multinational Design Evaluation Programme (MDEP) and the PRPPWG, participating in joint meetings or otherwise pursuing mutually beneficial collaborations with each of these entities.

List of abbreviations and acronyms

Generation IV International Forum

CDBOP	Component design and balance of plant (SFR signed project)
CD&S	Conceptual design and safety (GFR signed project)
EG	Experts group
EMWG	Economic Modeling Working Group
FCM	Fuel and core materials (GFR Project)
GACID	Global actinide cycle international demonstration (SFR signed project)
GIF	Generation IV International Forum
GFR	Gas-cooled fast reactor
ISAM	Integrated safety assessment methodology
LFR	Lead-cooled fast reactor
MOU	Memorandum of understanding
MSR	Molten salt reactor
PA	Project arrangement
PG	Policy Group
PMB	Project management board
PR&PP	Proliferation resistance and physical protection
PR	Proliferation resistance
PRPPWG	Proliferation Resistance and Physical Protection Working Group
PSSC	Provisional System Steering Committee
RSWG	Risk and Safety Working Group
SA	System arrangement
SCWR	Supercritical-water-cooled reactor
SDC	Safety design criteria
SFR	Sodium-cooled fast reactor
SIA	System integration and assessment (SFR Project)
SIAP	Senior Industry Advisory Panel
SO	Safety and operation (SFR signed project)
SRP	System research plan
SSC	System steering committee
VHTR	Very-high-temperature reactor

Technical terms

ALFRED	Advanced lead fast reactor European demonstrator
ASTRID	Advanced sodium technological reactor for industrial demonstration
AVR	*Arbeitsgemeinschaft Versuchsreaktor*
CEFR	China experimental fast reactor
DHR	Decay heat removal
ELFR	European lead fast reactor
EVOL	Evaluation and viability of liquid fuel fast reactor system (Euratom FP7 Project)
FHR	Fluoride-salt-cooled high-temperature reactor
GT-MHR	Gas turbine-modular helium reactor

Technical terms *(cont'd)*

HTGR	High-temperature gas-cooled reactor
HTR-PM	High-temperature gas-cooled reactor power generating module
HTR-10	High-temperature gas-cooled test reactor with a 10 MW_{th} capacity
HTSE	High temperature steam electrolysis
HTTR	High temperature test reactor
IHX	Intermediate heat exchanger
LWR	Light water reactor
MA	Minor actinides
MSFR	Molten salt fast reactor
NGNP	New generation nuclear plant
ODS	Oxide dispersion-strengthened
PBMR	Pebble bed modular reactor
PWR	Pressurised water reactor
R&D	Research and development
RPV	Reactor pressure vessel
SCW	Supercritical water
SMR	Small modular reactor
SSTAR	Small, sealed, transportable, autonomous reactor
TRISO	Tristructural isotopic (Nuclear Fuel)
TRU	Transuranic

Organisations

ANS	American Nuclear Society
CNRS	*Centre national de la recherche scientifique* (France)
DOE	Department of Energy (United States)
EU	European Union
FP7	7th Framework Programme
IAEA	International Atomic Energy Agency
ISTC	International Science and Technology Centre (Russia)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre (Euratom)
KIT	Karlsruhe Institute of Technology (Germany)
MDEP	Multinational Design Evaluation Programme
NEA	Nuclear Energy Agency (OECD)
NRC	Nuclear Regulatory Commission (United States)
NRI	Nuclear Research Institute (Czech Republic)
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory (United States)
PSI	Paul Scherrer Institute (Switzerland)
RIAR	Research Institute of Atomic Reactors (Russia)

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This *Technology Roadmap Update* provides an assessment of progress made by the Generation IV International Forum (GIF) in the development of the six systems selected when the original *Technology Roadmap* was published in 2002. More importantly, it provides an overview of the major R&D objectives and milestones for the coming decade, aiming to achieve the Generation IV goals of sustainability, safety and reliability, economic competitiveness, proliferation resistance and physical protection. Lessons learnt from the Fukushima Daiichi nuclear power plant accident are taken into account to ensure that Generation IV systems attain the highest levels of safety, with the development of specific safety design criteria that are applicable across the six systems. Accomplishing the ten-year R&D objectives set out in this new roadmap should allow the more advanced Generation IV systems to move towards the demonstration phase.

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