GEN IV International Forum℠

2008 Annual Report

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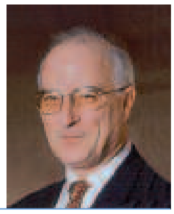

MESSAGE FROM THE CHAIRMAN

The pride and pleasure I take in presenting the GIF 2008 Annual Report result, in the first place, from its technical content, which I feel provides the reader with a vision of the Forum's most recent achievements, obtained despite the numerous technological challenges that pave the way to fourth generation nuclear technologies. However, this feeling of pride and pleasure is greatly fuelled also by the fact that publication of this report appears to be very timely with respect to recent changes in the international context.

Following a rather bleak period, during which nuclear energy was dormant or even in decline in some cases, and with its share in world electricity supply having stagnated at around 16% since the mid 1980s, there recently have been signs of an imminent nuclear renaissance. Rising energy demand resulting from global population growth, increased awareness of the consequences of global warming induced by the use of fossil fuels, skyrocketing prices of such fuels and concerns over security of energy supply have combined to make a stronger case for nuclear power. In many countries, nuclear energy is back on the national policy agendas, and prospects for construction of new nuclear power plants appear bright worldwide. Much has already been said and written about this nuclear revival, and one can only welcome such developments.

At the same time, it is important to note another, more recent trend, in the debates about a possible larger use of nuclear power, namely a growing awareness of the need to develop sustainable nuclear power. It is in regard to meeting this development, in particular, that the GIF can play a major role at the international level, thereby helping to ensure that the challenges raised by a nuclear renaissance can be dealt with satisfactorily. The major nuclear countries worldwide are turning towards two basic types of reactor concepts, both of which are being investigated within the GIF framework:

- fast reactor concepts which, when operated in conjunction with fuel recycling, appear as a very appealing and promising option from the standpoint of uranium resource utilization, waste minimization and management, and proliferation resistance; and/or
- reactor concepts which allow to broaden the fields of nuclear power application.

Research and development programs on fast reactor technology and closing the fuel cycle are being actively pursued in several countries, in some cases being resumed after decades of interruption:

- In China, with the 25 MWe China experimental fast reactor, to be followed by the 600 MWe China prototype fast reactor.
- In France, with the decision to construct a fourth generation reactor prototype to be operated by 2020, which will strengthen the research and development program on fast reactors, following the setback resulting from the SuperPhenix shutdown.
- In Japan, with the Joyo experimental reactor, the 280 MWe Monju prototype commercial fast breeder reactor, and the Japan Standard Fast Reactor concept.
- In Russia, with the BN-600 and BN-800 fast breeder reactors.
- In the United States, with the Global Nuclear Energy Partnership which has fast reactor technology as one leg of its technology development.
- And in India, a non-GIF-member country, with a 500 MWe fast reactor planned to enter operation in 2010 and important works including reprocessing of carbide fuel being carried out.

Some countries are pursuing the fast reactor technology primarily to burn actinides for waste management, while others focus also on breeding fissionable materials. In many cases, the fast reactor demonstration plants are evolving towards commercially viable options.

With regard to concepts aiming towards broadening nuclear power applications, the GIF systems with higher output temperature appear very attractive for process heat applications or for hydrogen production in large quantity and under favourable economic conditions. In this context, let me refer for example to the US NGNP project which is a part of DOE's Generation IV nuclear program, and which focuses on very high-temperature reactor technologies to produce hydrogen and other energy products, in order to ensure the viability of the next-generation of nuclear energy systems. The NGNP is to serve as a prototype for a commercial HTGR.

Against this backdrop of a very favourable international context for the development of fourth generation nuclear systems, major achievements of the Forum, which are described in detail in the body of the present report, are noteworthy. First of all, at the policy level, it is important to mention the signature of the VHTR System Arrangement by the People's Republic of China, only a few months after the completion of its GIF membership procedure. It is also worth mentioning the ratification by the Republic of South Africa of the GIF Framework Agreement, which makes it a full member of the Forum.

In addition to the work performed within already signed Project Arrangements (PAs), the collaborative framework of the GIF was strengthened with the signing of several new PAs, thus giving a boost to the corresponding R&D activities. In particular, two PAs were signed for the VHTR, and progress was made on preparing the signature of additional PAs for the SFR and several PAs for the SCWR and the GFR. For the LFR and the MSR, work has continued on System Research Plans.

Although R&D is being pursued on all six of the concepts selected within the GIF 2002 Technology Roadmap, it appears that progress is not at the same pace for all, with the sodium and gas cooled reactors being clearly ahead of the others. Such a trend can also be noticed within national programs that include plans to build, within the next decade or so, prototypes of these two types of reactors. An initiative was launched in 2008, jointly by the United States, France and Japan, to harmonize their national projects related to sodium-cooled fast reactor prototype construction. Talks have been initiated for the GIF to play an active role in this cooperation.

I would like to mention that the GIF is preparing a Symposium to be held in Paris, France, in September 2009 on the theme "2009 GIF Symposium: 10 years of achievements and the paths forward". This Symposium will include two events: several sessions devoted to the GIF community, aimed at reviewing the progress achieved since the creation of the Forum, and drawing conclusions in terms of the paths forward; and a plenary session of the GLOBAL 2009 Conference open to all interested experts, which will present the accomplishments of the GIF and also allow for a discussion on the issue of industrial implementation of fourth generation nuclear systems through public and private initiatives.

To conclude, I would like to express my optimism regarding the future of GIF. Even though reaping the full benefits of fourth generation nuclear systems will require extensive efforts from the GIF community and will take time, a trend has clearly been set by its members, showing political willingness to support and promote the development of sustainable nuclear energy and enthusiasm from the experts and researchers involved in the Forum's collaborative R&D to work together and overcome the technical challenges that lie ahead of them. I therefore congratulate the GIF community for their endeavour, and thank them for their efforts in helping to achieve the GIF's goals.

Jacques BOUCHARD
GIF Chairman - March 2009

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Chapter 1 Introduction

The importance and benefits of nuclear energy as an option for meeting growing energy needs worldwide gained further recognition in 2008 on the part of policy makers, industry leaders and technical experts. Nuclear energy is increasingly accepted as a proven, large-scale energy supply option with an abundant and secure resource base that can be employed safely, reliably, and with negligible emission of pollutants or greenhouse gases responsible for global climate change. Nuclear energy is currently used to produce 16% of electricity generated worldwide and has great potential to produce significantly more, as well as to supply heat for a variety of applications including production or refinement of transportation fuels and desalination of seawater.

A number of new nuclear plants of evolutionary Generation III or III+ design are currently under construction in several countries, including the People's Republic of China, the Russian Federation, the Republic of Korea, Japan, Finland and France. In other countries planning to add nuclear generation capacity in the future, including a number that make significant use of nuclear energy already, construction or firm commitments to build new plants have not yet materialized for a variety of reasons including the capital intensiveness of nuclear power, regulatory uncertainty, and public acceptance challenges. For countries that are interested in newly adopting nuclear energy, specific plant characteristics meeting their needs, as well as the infrastructure requirements associated with the safe and secure operation of nuclear plants in these countries, have received increased attention.

Because of the rapid growth of energy needs in developing countries and the need to add or replace generation capacity in many developed countries, it is expected that in the coming years a significant number of nuclear plants will be ordered, built and operated. For the most part, these will be state-of-the-art systems of Generation III/III+ design. As nuclear energy generation increases worldwide, the incentive and opportunity will increase to improve the performance of the systems delivering it and to better meet social, environmental and economic requirements of the 21st century. Generation IV nuclear energy systems are under development to meet these future needs. They employ advanced technologies and designs to improve upon the performance of current or evolutionary reactors, particularly through reduced waste generation and improved waste management, improved utilization of fuel resources, enhanced proliferation resistance and physical protection, increased safety and reliability, and improved economics.

The Generation IV International Forum (GIF) is a cooperative international endeavor organized to carry out the research and development (R&D) needed to establish the feasibility and performance capabilities of Generation IV systems. It has selected six systems for further R&D: the Gas Fast Reactor (GFR); the Lead Fast Reactor (LFR); the Molten Salt Reactor (MSR); the Sodium Fast Reactor (SFR); the Super-Critical Water Reactor (SCWR); and the Very High Temperature Reactor (VHTR). In 2008, the GIF cooperative framework was expanded through the accession by the Republic of South Africa to the GIF Framework Agreement (FA), the signature by the People's Republic of China of the System Arrangement (SA) governing R&D cooperation for the VHTR, and the formal establishment of new cooperative research and development Project Arrangements (PA) for the VHTR and the SFR.

VHTR SA signature by China – October 2008

This 2008 Annual Report is the second annual report issued by GIF. It provides an update on the GIF organization, membership, and participation in R&D projects for each Generation IV system. It summarizes the milestones for development of each system and progress of the R&D toward their accomplishment. Finally, it includes a brief description of the cooperation between GIF and other international endeavors for the development of nuclear energy.

Chapter 2 describes the membership and organization of the GIF, the structure of its cooperative research and development arrangements, and the status of Member participation in those arrangements.

Chapter 3 provides a summary of the GIF R&D plans, and its activities and achievements during 2008. It highlights the R&D challenges facing the teams developing Generation IV systems and the major milestones towards the development of these systems. It also describes the progress made regarding the development of methodologies for assessing Generation IV systems with respect to the established goals of GIF.

Chapter 4 reviews other major international collaborative projects in the field of nuclear energy and explains how the GIF interacts and cooperates with them.

Appendix 1 provides an overview on the goals of Generation IV nuclear energy systems and outlines the main characteristics of the six systems selected for joint development by GIF.

The list of abbreviations and acronyms given at the end of the report defines terms used in the various chapters including various nuclear energy systems and international programs referred to in connection with GIF R&D activities.

Some bibliographical references are given in order to facilitate access to public information about R&D progress and achievements on specific technical issues for GIF systems. A public web site (www.gen-4.org) provides a wealth of technical and scientific information on Generation IV systems and methodologies.

2.1 GIF Membership

The Generation IV International Forum has thirteen Members, as shown in Table 2.1, which are signatories of its founding document, the *GIF Charter* (www.gen-4.org/PDFs/GIFcharter.pdf). Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, the United Kingdom and the United States signed the GIF Charter in July 2001. Subsequently, it was signed by Switzerland in 2002, Euratom[1] in 2003, and the People's Republic of China and the Russian Federation, both in 2006. Signing the Charter signifies interest in cooperation on Generation IV systems but does not commit the signatories to take part in the cooperative development of those systems.

Table 2.1: Parties to GIF Framework Agreement and System Arrangements

Member	Implementing Agents	Framework Agreement	System Arrangements			
			GFR	SCWR	SFR	VHTR
Argentina						
Brazil						
Canada	Department of Natural Resources	X		X		X
Euratom	Joint Research Centre (JRC)	X	X	X	X	X
France	Commissariat à l'énergie atomique (CEA)	X	X		X	X
Japan	Agency for Natural Resources and Energy Japan Atomic Energy Agency (JAEA)	X	X	X	X	X
People's Republic of China	China Atomic Energy Authority Ministry of Science and Technology	X				X
Republic of Korea	Ministry of Education, Science and Technology (MEST) Korea Science and Engineering Foundation	X			X	X
Republic of South Africa	Department of Minerals and Energy	X				
Russian Federation						
Switzerland	Paul Scherrer Institute	X	X			X
United Kingdom						
United States	Department of Energy (DOE)	X			X	X

Among the signatories to the Charter, nine Members (Canada, Euratom, France, Japan, the People's Republic of China, the Republic of Korea, the Republic of South Africa, Switzerland and the United States) have signed or acceded to the Framework Agreement (FA) as shown in Table 2.1. Parties to the

[1] The European Atomic Energy Community (Euratom) is the implementing organization for development of nuclear energy within the European Union.

Framework Agreement formally agree to participate in the development of one or more Generation IV systems selected by GIF for further R&D. Each Party to the Framework Agreement designates one or more Implementing Agents (see Table 2.1) to undertake the development of systems and the advancement of their underlying technologies.

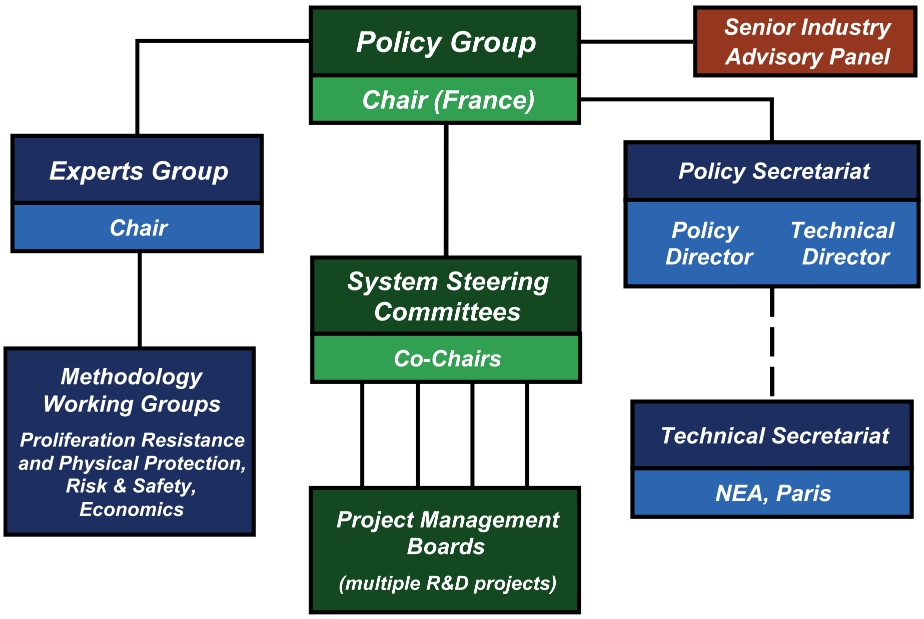
Argentina and Brazil have signed the GIF Charter but not the Framework Agreement, and the United Kingdom withdrew from the FA; accordingly, within the GIF, they are designated as "inactive Members." The Russian Federation is working on the necessary approvals for its accession to the FA.

Members interested in implementing cooperative R&D on one or more of the selected systems have signed corresponding System Arrangements (SA) consistent with the provisions of the FA. The participation of GIF Members in System Arrangements is shown in Table 2.1.

2.2 GIF Organization

The GIF Charter provides a general framework for GIF activities and outlines its organizational structure. Figure 2.1 gives a schematic representation of the GIF governance structure and indicates the relationships among different GIF bodies which are described below.

Figure 2.1: GIF Governance Structure



As detailed in its Charter and subsequent GIF Policy Statements (www.gen-4.org), the GIF is led by the Policy Group (PG) which is responsible for the overall steering of the GIF cooperative efforts, the establishment of policies governing GIF activities, and interactions with third parties. Every GIF Member may nominate up to two representatives in the Policy Group. The PG usually meets three times each year.

Policy Group in Beijing – October 2008

The Experts Group (EG), which reports to the Policy Group, is in charge of reviewing the progress of cooperative projects and of making recommendations to the Policy Group on required actions. It advises the Policy Group on R&D strategy, priorities and methodology and on the assessment of research plans prepared in the framework of System Arrangements. Every GIF Member may appoint up to two representatives in the Experts Group. The EG usually meets twice each year and one of its meetings is adjacent to a PG meeting in order to facilitate exchanges and synergy between the two groups.

Signatories of each SA have formed a System Steering Committee (SSC) in order to plan and oversee the R&D required for the corresponding system. R&D activities for each GIF system are implemented through a set of Project Arrangements (PA) signed by interested bodies. A PA typically addresses the R&D needs of the corresponding system in a broad technical area (e.g., fuel technology, advanced materials and components, energy conversion technology, plant safety). A Project Management Board (PMB) is established by the signatories to each PA in order to plan and oversee the project activities which aim to establish the viability and performance of the relevant Generation IV system in the technical area concerned.

The GIF Charter and Framework Agreement allow for the participation of organizations from public and private sectors of non-GIF Members in PAs and in the associated PMBs, but not in SSCs. Public and private sector organizations, including those from non-GIF Members, may join any PA but, for organizations from non-GIF Members, it requires unanimous approval by the signatories to the PA and to the governing SA. The PG may provide recommendations to the SSC on the participation in GIF R&D Projects by organizations from non-GIF Members.

Three Methodology Working Groups (MWGs) are responsible for developing and implementing methods for the assessment of Generation IV systems against GIF goals in the fields of economics, proliferation resistance and physical protection, and risk and safety. Those Groups – the Economic Modeling Working Group (EMWG), the Proliferation Resistance and Physical Protection Working Group (PRPPWG), and the Risk and Safety Working Group (RSWG) – report to the Experts Group which provides guidance and periodically reviews their work plans and progress. Members of the MWGs may be appointed by the Policy Group representatives of every GIF Member.

A Senior Industry Advisory Panel (SIAP) comprised of executives from the nuclear industries of GIF Members was established in 2003 to advise the Policy Group on long-term strategic issues, including regulatory, commercial or technical aspects. The SIAP contributes to strategic reviews of the GIF R&D activities in order to ensure that technical issues impacting on future commercial introduction of Generation IV systems are taken into account. In 2008, the SIAP focused its contributions and guidance on issues which may be raised eventually by the commercialization and deployment of advanced nuclear systems. In particular, the SIAP provided guidance on taking into account investor-risk reduction and incorporating the associated challenges in system designs at an early stage of its development.

The GIF Secretariat is the day-to-day coordinator of GIF activities and communications. It includes two groups: the Policy Secretariat and the Technical Secretariat. The Policy Secretariat assists the Policy Group and Experts Group in the fulfillment of their responsibilities. Within the Policy Secretariat, the Policy

Director assists with the conduct of the Policy Group whereas the Technical Director serves as Chair of the Experts Group and assists the Policy Group on technical matters. The Technical Secretariat (TS), provided by the Nuclear Energy Agency (NEA) of the Organisation for Economic Cooperation and Development, supports the SSCs, PMBs and MWGs. The NEA is entirely resourced for this purpose through voluntary contributions from GIF Members either financial or in kind (e.g., providing a cost-free expert for supporting TS work.

2.3 Participation in GIF R&D Projects

For each Generation IV system, the relevant SSC creates a System Research Plan (SRP) which is attached to the corresponding System Arrangement. As noted previously, each SA is implemented by means of several Project Arrangements established in order to carry out the required R&D activities in different technical areas as specified in the SRP. Every PA includes a Project Plan consisting of specific tasks to be performed by the signatories.

As of 1 March 2009, System Arrangements had been signed by several Members for four systems (GFR, SCWR, SFR and VHTR). For the LFR and the MSR, collaborative R&D is pursued by interested Members under the auspices of provisional SSCs. Three Project Arrangements had been signed within the SFR system: the Advanced Fuel (AF) PA; the Global Actinide Cycle International Demonstration (GACID) PA; and the Component Design and Balance-Of-Plant (CDBOP) PA. Within the VHTR system, two PAs had been signed: the Fuel and Fuel Cycle (FFC) PA; and the Hydrogen Production (HP) PA. Several other projects are in the process of signature, and others are defined already and their membership agreed upon by interested parties on a provisional basis. Figure 2.2 gives an overview of GIF collaborative R&D structure and Table 2.2 shows the list of signed arrangements and provisional cooperation within GIF as of 1 March 2009.

Figure 2.2: GIF Collaborative R&D Structure

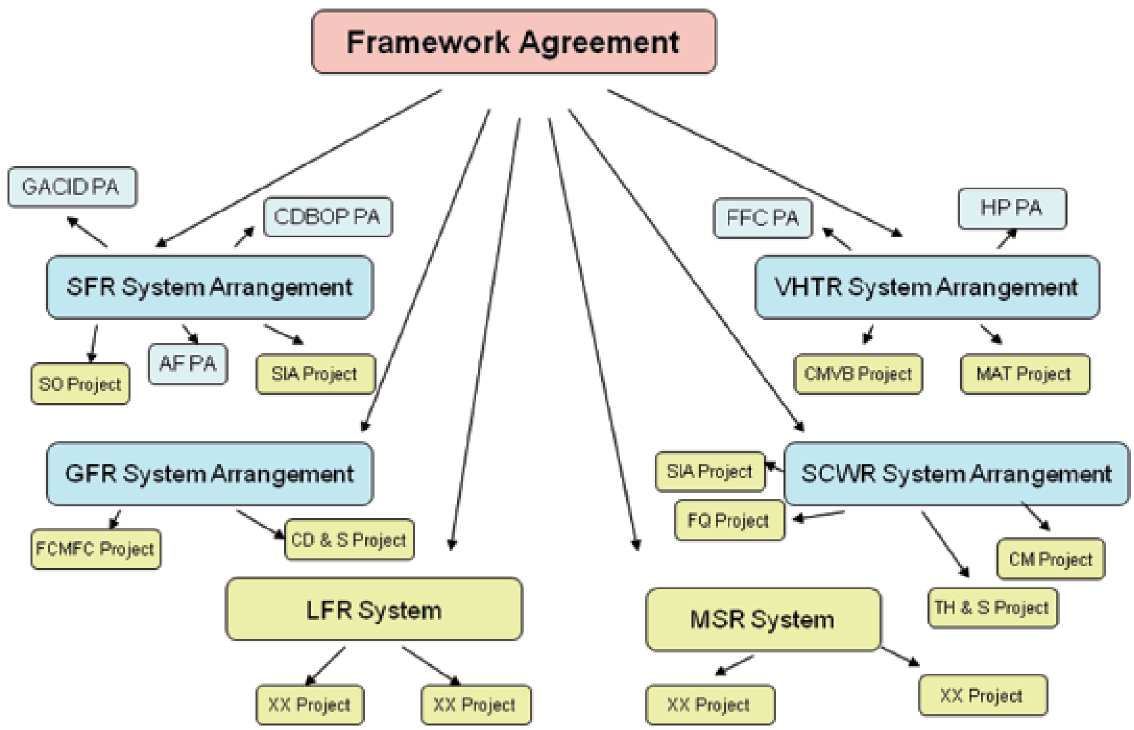


Table 2.2: Signed arrangements and informal cooperation within GIF

	CAN	EUR	FRA	JPN	CHN	KOR	ZAF	RUS	CHE	USA
VHTR SA	X	X	X	X	X	X			X	X
VHTR HP PA	X	X	X	X		X			O	X
VHTR FFC PA	O	X	X	X		X				X
VHTR CMVB Project		P	P	P		P	P			P
VHTR MAT Project	P	P	P	P		P	P		P	P
SFR SA		X	X	X	O	X		O		X
SFR AF PA		X	X	X		X				X
SFR GACID PA			X	X						X
SFR CDBOP PA			X	X		X				X
SFR SO Project			P	P		P				P
SFR SIA Project		P	P	P		P				P
SCWR SA	X	X		X						
SCWR CM Project	P	P	P	P		O				
SCWR TH&S Project	P	P		P		O				
SCWR SIA Project	P	P		P		O				
SCWR FQ Project	P	P		P						
GFR SA		X	X	X					X	
GFR FCMFC Project		P	P	P					O	
GFR CD&S Project		P	P						P	
LFR System		P		P						P
MSR System		P	P							P

X = Signatory P = Provisional participant O = Observer

Acronyms of Projects

HP	Hydrogen Production
FFC	Fuel and Fuel Cycle
CMVB	Computational Methods Validation and Benchmarking
MAT	Materials
AF	Advanced Fuel
GACID	Global Actinide Cycle International Demonstration
CDBOP	Component Design and Balance-Of-Plant
SO	Safety and Operation
SIA	System Integration and Assessment
CM	Materials and Chemistry
TH&S	Thermal-Hydraulics and Safety
FQ	Fuel Qualification
FCMFC	Fuel, Core Materials and Fuel Cycle
CD&S	Component Design and Safety

Beyond the formal and provisional R&D collaborations shown in Table 2.2, many institutes and laboratories cooperate with GIF Projects through exchange of information and results, as indicated in Chapter 3 and in the bibliographical references given at the end of the chapter.

R&D activities within GIF are carried out at the project level and involve all sectors of the research community, including universities, governmental and non-governmental laboratories as well as industry, from interested GIF and non-GIF Members.

3.1 Systems

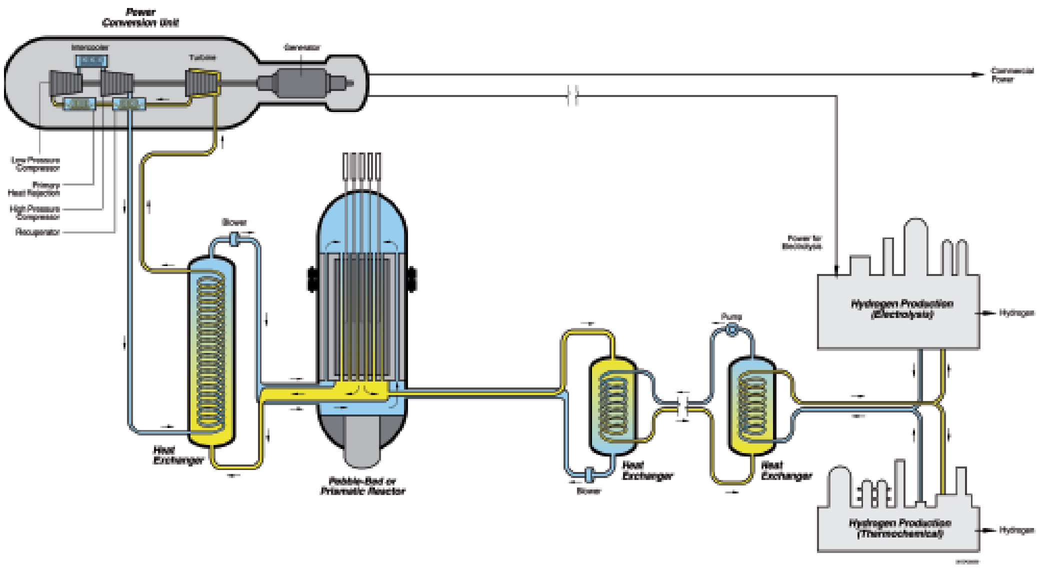
The main results obtained in 2008 for each of the six systems selected by GIF members for further R&D are provided in the following sections. Although the focus is on collaborative work pursued in 2008, a brief overview of the characteristics of each system is given as background for putting the R&D undertaken in perspective. Relevant key outcomes from research programs pursued by GIF Members outside of the GIF collaborative framework are described, especially for systems which had not yet an established/signed System Arrangement in 2008. More detail on scientific and technical aspects of the systems may be found in conference papers and journal articles listed in the bibliography provided at the end of chapter.

3.1.1 Very-High-Temperature Reactor (VHTR)

Main characteristics of the system

The VHTR is the next step in the evolutionary development of high-temperature reactors and is primarily dedicated to the cogeneration of electricity and hydrogen, the latter being extracted from water by using thermo-chemical, electro-chemical or hybrid processes (see Figure 3.1). Its high outlet temperature makes it attractive also for the chemical, oil and iron industries. It is an advanced reactor cooled by helium gas and moderated by graphite with a core outlet temperature greater than 900°C (with an ultimate goal of 1 000°C) to support the efficient production of hydrogen by thermo-chemical processes. The VHTR has potential for high burn-up (150-200 GWd/tHM), completely passive safety, low operation and maintenance costs, and modular construction.

Figure 3.1: Schematic diagram of the VHTR and hydrogen production systems



Two baseline concepts are available for the VHTR core: the pebble bed-type and the prismatic block-type. The fuel cycle will initially be once-through with low-enriched uranium fuel and very high fuel burn-up. Solutions will be developed to manage adequately the back-end of the fuel cycle and the potential for a closed fuel cycle will be assessed. Although various fuel designs are considered within the VHTR systems, all concepts exhibit extensive similarities allowing for a coherent R&D approach.

The electric power conversion unit may operate in either a direct (helium gas turbine) or indirect (gas mixture turbine) Brayton-type cycle. Near-term concepts will be developed using existing materials, whereas more advanced concepts will require the development of new materials.

The basic technology for the VHTR has been established in former high-temperature gas reactors such as the US Peach Bottom and Fort Saint-Vrain plants as well as the German AVR and THTR prototypes. The technology is being advanced through near- and medium-term projects, such as PBMR, HTR-PM, GTHTR300C, ANTARES, NHDD, GT-MHR and NGNP, led by several plant vendors and national laboratories respectively in the Republic of South Africa, the People's Republic of China, Japan, France, the Republic of Korea and the United States. Experimental reactors such as HTTR (Japan, 30 MWth) and HTR-10 (China, 10 MWth) support the advanced reactor concept development, together with research on the cogeneration of electricity and hydrogen, or other nuclear heat applications.

Status of cooperation

The VHTR System Arrangement was signed in December 2006 by Canada, Euratom, France, Japan, the Republic of Korea, Switzerland and the United States. In October 2008, the People's Republic of China formally signed the VHTR SA during the Policy Group meeting held in Beijing. The Republic of South Africa, which has expressed high interest in the VHTR, formally acceded to the GIF Framework Agreement in 2008, and is expected to sign the VHTR SA in 2009.

The Fuel and Fuel Cycle Project Arrangement became effective on January 30, 2008, with Implementing Agents from Euratom, France, Japan, the Republic of Korea and the United States. The Hydrogen Production Project Arrangement became effective on March 19, 2008 with Implementing Agents from Canada, Euratom, France, Japan, the Republic of Korea and the United States.

The Materials Project Arrangement, which addresses graphite, metals, ceramics and composites, has been finalized and is expected to be submitted for signature by interested parties in 2009. It should be noted that the provisional Materials Project Management Board requested that the VHTR System Steering Committee approve the direct participation of PBMR Pty Ltd in the project. The provisions of the GIF Framework Agreement, under Article V, allow a SSC to approve other entities from the public or private sectors to be signatories to a PA subject to the unanimous approval of the SSC. The SSC voted unanimously on October 2, 2008, to approve direct participation of PBMR Pty Ltd in the Materials PA.

The Computational Methods, Validation and Benchmarking Project Arrangement is expected to be finalized and ready for signature in 2009.

Two other projects – on components and high-performance turbo machinery and on design, safety and integration – are being discussed by the VHTR SSC but the associated research plans and Project Arrangements have not been developed yet for those two areas.

R&D Objectives

The VHTR development approach builds on technologies already used for gas reactors that have successfully been constructed and operated, as well as reactors deployed in the United Kingdom using carbon-dioxide gas.

While shorter-term concepts will rely more on existing materials and technology, the long-term VHTR R&D will benefit from work on those short-term concepts through re-establishment of the knowledge base needed for manufacturing and licensing of high-temperature reactors. The VHTR R&D objectives will be addressed eventually within six projects described below, four of which are already ongoing while the work on components and high-performance turbo machinery and on design, safety and integration has not been formally initiated.

Computational methods development and validation in the areas of thermal hydraulics, thermal mechanics, core physics, and chemical transport are major activities for the assessment of the reactor performance, in normal, incidental and accidental conditions. Code validation will be carried out through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported by HTTR and HTR-10 tests or by past technology high-temperature reactor data (e.g. AVR and Fort St Vrain). Normal and abnormal operating analyses will be performed. Improved computational methods will also facilitate the elimination of unnecessary design conservatisms and improve construction cost estimates.

In the area of fuel and fuel cycle, TRISO coated particles, which are the basic fuel concept for the VHTR, need to be qualified for relevant service conditions. R&D will increase the understanding of standard design UO_2 kernel with SiC/PyC coating and examine the use of UCO kernels and ZrC coatings for enhanced burn-up capability, reduced fission product permeation and increased resistance to core heat-up accidents (above 1 600°C). This work will involve fuel characterization, post irradiation examination, safety testing, fission product release evaluation, as well as assessment of chemical and thermo-mechanical materials properties in representative conditions. The information obtained will support a data base to be used for running advanced physical models in order to assess the in-core fuel behavior under normal and off-normal conditions. R&D will examine also spent-fuel treatment and disposal, including used-graphite management, as well as the deep-burn of plutonium and minor actinides (MA) in support of a closed cycle.

R&D on materials is essential for the VHTR system development. For core coolant outlet temperatures around 900°C it is envisioned that existing materials can be used; however, the goal of 1 000°C, including safe operations under off-normal conditions, will require the development and qualification of new materials. Focus areas include: graphite for the reactor core and internals; metallic materials for internals, piping, valves, heat exchanger and gas turbine components; and ceramics and composites for control rod cladding and other internals, as well as for intermediate heat exchangers and gas turbine components. In this field, the aim is to identify and develop materials adequate for very-high-temperature conditions. Characterization tests in relevant service conditions will build a data base on thermo-mechanical properties under irradiation, as well as corrosion resistance. The results will be used to support the development of design codes and standards as well as modeling to predict damage and lifetime assessment.

In conjunction with materials development covered above, design and construction methodologies need to be addressed for key reactor systems and energy conversion components. These components will require advances in modular manufacturing and on-site construction techniques, including new welding and post-weld heat treatment techniques and will need to be tested in dedicated large scale helium test loops, capable of simulating normal and off-normal events.

For hydrogen production, two main processes are considered: the sulfur/iodine thermo-chemical cycle; and the high-temperature electrolysis process. R&D will address feasibility, optimization, efficiency and economics evaluation for small and large scale hydrogen production. Performance and optimization of both processes will be assessed through integrated test loops, from laboratory scale through pilot and demonstration scale, and include component development such as advanced process heat exchangers. Hydrogen process coupling technology with the nuclear reactor will be investigated and design-associated risk analysis will be performed covering potential interactions between nuclear and non-nuclear systems. Thermo-chemical or hybrid cycles will be examined in terms of technical and economic feasibility in dedicated or cogeneration hydrogen production modes, aiming at lowering operating temperature requirements in order to make them compatible with other Generation IV systems.

Integration work will aim at updating the viability and performance assessment of the VHTR baseline concepts against the GIF performance goals and criteria, while integrating the results of technological R&D. It is expected that studies and R&D activities, including code development and mechanical design codification, as well as market analyses, will be undertaken in order to support system integration, safety and economic analyses, and licensing.

Milestones

The major milestones defined in VHTR System Research Plan are:

- Viability stage / preliminary design and safety analysis: 2010
- Performance stage / final design and safety analysis: 2015
- Demonstration stage / construction and preliminary testing: 2020

The schedules of the R&D work to be completed within the VHTR projects for which research plans have been finalized are summarized below.

- Fuel and Fuel Cycle Project
  - ✓ Irradiation and post-irradiation examination
    - 2015 Results from post-irradiation examination
  - ✓ Fuel attributes and material properties
    - 2009 Establishment of fuel material property database
    - 2009 Characterization of fuel attributes and fuel performance modeling
  - ✓ Safety
    - 2012 Pulse irradiation testing, establishment of heating test capability, and source term experiments
    - 2015 Heating test
  - ✓ Enhanced and advanced fuel fabrication (e.g. UCO, ZrC)
    - 2010 Process development
  - ✓ Waste management
    - 2010 Disposal behavior and waste package
  - ✓ Other fuel cycle options
    - 2009 Thorium cycle
    - 2010 Plutonium burning and transmutation

- Materials Project
  - ✓ Graphite
    - 2012 Data, design methodology, and construction
    - 2012 Gen IV database
  - ✓ Metals and design methods
    - 2012 Data generation (mechanical, physical, chemical properties)
    - 2012 Gen IV database
  - ✓ Ceramics and composites
    - 2012 Data generation (mechanical, physical, chemical properties)
    - 2012 Gen IV database

- Hydrogen Production Project
  - ✓ Sulfur/Iodine process
    - 2009 Laboratory-scale test and optimization

- ✓ High temperature electrolysis
  - 2010 Laboratory-scale integrated experiment
  - 2014 Pilot-scale experiment
- ✓ Alternative processes
  - 2009 Evaluation of economics
  - 2010 Screening and technical evaluation
- ✓ Coupling technology
  - 2010 Process evaluation and component technology

Main activities and outcomes

Regarding the FFC Project, the first action plan established for the period 2007-2009 identifies more than one hundred deliverables mainly associated with three work packages on irradiation, post-irradiation evaluation, and fuel attributes and material properties. The efforts devoted by all members of the project to each work package are summarized in Table 3.1.

Table 3.1: VHTR FFC Project – Financial Summary 2008

Work package title	Labor (person x year)	Total estimated cost (10^3 US$)
Irradiation and post irradiation examination	15.8	8 356
Fuel attributes and material properties	13.5	4 467
Safety	9.6	4 198
Enhanced advanced fuel	2	1 950
Waste management	3.6	1 894
Other fuel cycle options	21.25	6 518.5
TOTAL	65.4	27 383.5

Work on irradiation proceeded in the framework of the Euratom PYCASSO-I experiment which started in April 2008 and is expected to be completed in the first quarter of 2009. The PYCASSO-II experiment, expected to start in the first quarter of 2009 for nine cycles of irradiation will have a higher fluence, up to $3x10^{25}$ $n.m^{-2}$. In late 2009, irradiation is scheduled in the United States (Advanced Gas Reactor 2 experiment) with fuel being fabricated for this purpose by the United States, France and the Republic of South Africa. Irradiation of three archive German pebbles and two Chinese pebbles at the Petten high flux reactor will continue in 2009. Finally, work is underway, under Euratom leadership, to establish a database on TRISO fuel materials that will be accessible to all members of the VHTR FFC Project.

The Hydrogen Production Project progressed steadily during 2008 (see financial summary in Table 3.2). Under the Sulfur-Iodine Process work package, experimental results are now used in simulation models to understand better the thermal efficiency of the processes as well as equilibrium constants and parameters. Research activities were conducted on catalysts using platinum/titanium oxide and iron oxide catalysts coated on Hastelloy sheets, as well as copper/iron mixed oxide and barium sulfate coatings to prevent corrosion. In the field of high temperature electrolysis, material balance and operational flow sheets were obtained using a commercial code. Results were obtained from the operation of a 15 kWe high temperature electrolysis experiment. In the area of alternative cycle development, the hydrogen production step of the hybrid copper chloride cycle was demonstrated using a proton-exchange-membrane electrochemical cell with current density of up to 0.5A per cm^2 over a voltage range of 0.56 to 0.9 V. Work continues on the development and improvement of the computational tools for the simulation of the intermediate circuit of a coupling process, and of tritium transport.

Table 3.2: VHTR HP Project – Financial Summary 2008

Work package title	Labor (person x year)	Total estimated cost (10^3 US$)
Sulfur iodine process	46.9	11 310.7
High temperature electrolysis	36.2	13 813
Alternative processes	24.4	6 060
Hydrogen production coupling technology	53.3	14 692
TOTAL	160.8	45 875.7

3.1.2 Sodium-cooled Fast Reactor (SFR)

Main characteristics of the system

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

Three major options are considered: a large size (600 to 1 500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors (Kotake *et al*., 2008); an intermediate-to-large size (300 to 1 500 MWe) pool-type reactor (Mignot *et al*., 2008; Joo *et al*., 2008); and a small size (50 to 150 MWe) modular-type reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor (Chang *et al*., 2005). The outlet temperature is 500-550°C for the three options.

The SFR closed fuel cycle facilitates management of high-level waste and in particular of plutonium and other actinides. Important safety features of the system include a long thermal response time, a large 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 carbon-dioxide are considered as working fluids for the power conversion system to achieve high performance in terms of thermal efficiency, safety and reliability. With innovations to reduce capital cost, the SFR will be economically competitive on electricity markets. In addition, the SFR fast neutron spectrum extends the lifetime of natural resources through using available fissile and fertile materials (including depleted uranium) considerably more efficiently than thermal-spectrum reactors with once-through fuel cycle.

Besides the SFR research and development conducted so far, significant upcoming activities include Phenix end-of-life tests, restart of Monju, and startup of the Chinese Experimental Fast Reactor scheduled in 2009.

Status of cooperation

The System Arrangement for the international research and development of the SFR nuclear energy system was signed on November 2006, and at present the official members of the SA are:

- The Commissariat à l’énergie atomique of France,
- The Department of Energy of the United States,

- The Joint Research Centre of EURATOM,
- The Japan Atomic Energy Agency of Japan, and
- The Ministry of Education, Science & Technology of the Republic of Korea.

Three Project Arrangements were signed in 2007 for Advanced Fuel, Component Design and Balance-Of-Plant and Global Actinide Cycle International Demonstration. The Project Arrangement for Safety and Operation is in the process of signature.

R&D Objectives

The SFR development approach builds on technologies already used for SFRs that have successfully been built and operated in France, Germany, Japan, the Russian Federation, the United Kingdom and the United States. As a benefit from these previous investments in technology, the majority of the R&D needs for the SFR are related to performance rather than viability of the system. Based on international SFR R&D plans, the research activities within GIF have been arranged by the SFR SA signatories into five projects. The scope and objectives of the R&D to be carried out in these five projects are summarized below.

In the field of safety, experiments and analytical model development are planned to address both passive safety and severe accident issues. Options of safety system architectures will be investigated also. The research on operation covers reactor operation and technology testing campaigns in existing SFRs (e.g., Monju and Phenix, including its end-of-life tests).

Fuel related research includes: the development of high-burn-up fuel systems (fuel form and cladding) to complete the SFR fuel database; research on remote fuel fabrication techniques for fuels that contain minor actinides and possibly traces of fission products; and the consideration of alternative fast reactor fuel forms for special applications (e.g., high temperature).

Research on component design and balance-of-plant covers experimental and analytical evaluation of advanced in-service inspection and repair technologies including leak-before-break assessment, and development of alternative energy conversion systems, e.g., using Brayton cycles. Such systems, if shown to be viable, would reduce the cost electricity generation significantly as compared with the Rankine steam cycle owing to its compactness.

The project on global actinide cycle international demonstration aims at demonstrating that the SFR can effectively manage all actinide elements – including uranium, plutonium, and minor actinides (neptunium, americium and curium) – by transmutation. The project includes fabrication and licensing of MA-bearing fuel, pin-scale irradiations, material property data preparation, irradiation behavior modeling and post-irradiation examination, as well as transportation of MA raw materials and MA-bearing fuels. Bundle-scale demonstration will be included. This technical demonstration will be pursued using existing fast reactors in a reasonable time frame.

The main objectives of system integration and assessment are: to maintain and refine system options, reflecting continuous trade-off studies and technology development; to recognize R&D needs and assure that the work scopes of the PMBs are based on these needs; to apply the GIF assessment methodologies to various concepts; and to integrate and assess the R&D results from the other projects.

Milestones

The key milestones of the five SFR system R&D projects are given below.

- AF Project

2006-2007	Preliminary evaluation of advanced fuels
2007-2010	Evaluation of MA-bearing fuels
2011-2015	High-burn-up fuel behavior evaluation
2016-	Demonstration and application of advanced fuel head-end process in the back-end of the fuel cycle

- CDBOP Project

2007-	Viability study of proposed concepts
2007-2010	Performance tests for detail design specification
2011-2015	Demonstration of system performance

- GACID Project

2007-2012	Preparation for the limited MA-bearing fuel irradiation test
2007-2012	Preparation for the licensing of the pin-scale curium-bearing fuel irradiation test
2007-2012	Program planning of the bundle-scale MA-bearing fuel irradiation demonstration

- SIA Project
  - ✓ Definition of SFR system options

2008-	Initial specification of SFR system options

  - ✓ Assessment of SFR system options

2008	Compilation of self-assessment results for SFR system options
2009-11	Assessments of economics, PR & PP and safety using GIF methodologies

  - ✓ Definition of SFR R&D needs

2008	Review and refinement of SFR R&D needs in the SRP
2009	Review of existing Project Plans to identify R&D gaps
2010-	Integration of R&D results to refine the system options and assessment of those results to provide feedback (guidance) to technical Projects.

- SO Project
  - ✓ R&D for Safety

2008-2009	Preliminary assessment of candidate safety provisions and systems
2008-2012	Performance assessment of safety provisions and systems
2011-2015	Qualification of safety provisions and systems

  - ✓ R&D for reactor operation and technology testing

2008-2011	Tasks related to SIA Project . Phenix end-of-life program . Thermal-hydraulics/general system . Feedback from the decommissioning of liquid metal fast reactors
2008-2012	Tasks related to CDBOP Project . Development of in-service inspection techniques for future SFR drawing from existing reactor experience . Sodium chemistry . Sodium technology

Main activities and outcomes

Activities on integration and assessment were conducted through joint meetings of the SFR SSC and the provisional PMB for the SIA project aiming at clarifying the project objectives, identifying integration and assessment work to be performed, and defining the relationship between technical PMBs and concept developers. The integration function of this project will cover a review of the results from technical projects aiming at their integration, regular updating of the system options, and establishment of a comprehensive list of R&D needs. The definition and boundaries of the assessment function have not been finalized yet. In 2008, an economic analysis of the small modular fast reactor concept was performed as a leading case for applying the GIF economic assessment methodology to a GEN-IV system. The Project Plan is expected to be finalized in the coming year in order to complete in 2009 the PA negotiations covering the implementation of the unique aspects of this project.

Two meetings of the provisional Safety and Operation PMB were held in 2008 to prepare and finalize the Project Plan and agree upon a work plan for the first year of collaboration. Collaboration is expected to start early in 2009, after the signature of the PA.

In the field of advanced fuels, the AF Project has been effective since March 2007. The total contributions invested by the AF PA signatories were 16 M US$ in 2007 and 11 M US$ in 2008. Exchange of R&D results started in 2008 on performance evaluations, MA-bearing fuel fabrication technology, and core material for high burn-up fuels. As a result of a technical evaluation, oxide, metal and nitride have been proposed as future fuel options. Carbide fuel R&D was initiated. Performance evaluations were carried out by project members covering: carbide and nitride/carbide fuels, MA-bearing oxide and metal fuel fabrication technologies, advanced fuel fabrication and characterization, irradiation resistance of advanced oxide dispersion-strengthened (ODS) materials, core materials for high burn-up fuels, and candidate cladding for high burn-up fuels.

Regarding component design and balance-of-plant, the CDBOP PA has been effective since October 2007. The total contributions invested by the CDBOP PA signatories were 3M US$ in 2007 and 4 M US$ in 2008. In 2008, work progressed on in-service inspection technologies, repair experience and supercritical-CO_2 (S-CO_2) power cycle turbine system studies. In the study of in-service inspection technologies, various sensors were tested to assess their performance. Review of feedback from the experience acquired in the Phenix lifetime-extension project and Joyo sodium piping replacement provided insights on repair technologies. Good progress was made in the viability demonstration of a very compact and efficient Brayton cycle energy conversion system in which the turbo-machinery uses supercritical CO_2 as the working fluid. A series of single compressor tests has been completed in the United States, that now provide data on most of the technical and hardware issues over the full range of conditions of interest near the CO_2 critical point, including at both liquid-like and vapor-like sides. The results are in excellent agreement with the models, and have confirmed that the system should be very robust and controllable near the critical point. In addition to confirmation of the fundamental issues of compressor fluid performance and system control near the critical point, these recent tests address the essential supporting technologies: measuring bearing loads, sealing leakage rates, and rotor windage losses. A basic S-CO_2 cycle turbine system was designed for KALIMER-600 by the Korean Atomic Energy Research Institute and test were carried out by JAEA on S-CO_2 compressor and material behavior under S-CO_2 flow conditions.

In the field of global actinide cycle, the GACID PA has been effective from September 2007. The total contributions invested by the PA signatories were 3 M US$ in 2007 and 10 M US$ in 2008. During the year 2008, activities performed in common by the members included evaluation of MA-bearing fuel material properties, analysis and evaluation of irradiated fuel data, and preliminary program planning for bundle-scale MA-bearing fuel assembly irradiation demonstration in Monju.

3.1.3 Super-Critical Water Reactor (SCWR)

Main characteristics of the system

The Super-Critical Water Reactor (SCWR) is a high temperature, high pressure water-cooled reactor that operates above the thermodynamic critical point of water (374°C, 22.1 MPa). Two design options – pressure vessel and pressure tube – are considered for the SCWR. The R&D needs to assess technical feasibility (e.g., materials, chemistry, operating conditions) are common to both designs, which provides valuable collaboration opportunities for countries and organizations pursuing either option.

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

Status of cooperation

Four projects have been identified and signatories of the SCWR SA have expressed willingness to contribute to those projects as indicated below:

- System Integration and Assessment (Canada, Euratom, Japan, and Republic of Korea as observer)
- Materials and Chemistry (Canada, Euratom, France, Japan, Republic of Korea)
- Thermal-Hydraulics and Safety (Canada, Euratom, Japan, Republic of Korea)
- Fuel Qualification

In 2008, efforts focused on finalizing the thermal-hydraulics and safety and the materials and chemistry Project Arrangements aiming at signing them at the beginning of 2009. For the system integration and assessment project a provisional PMB was created and worked in 2008 on drafting the technical part of the PA. The project on fuel qualification was recently created with the objective of testing the SCWR fuel in a suitable research reactor under prototypical supercritical conditions. While waiting for the signature of PAs, signatories of the SA are sharing results from R&D through informal exchanges and provisional PMB meetings.

R&D Objectives

Regarding system design, the objective is to pursue pre-conceptual design activities for several concepts in order to investigate their respective potentials.

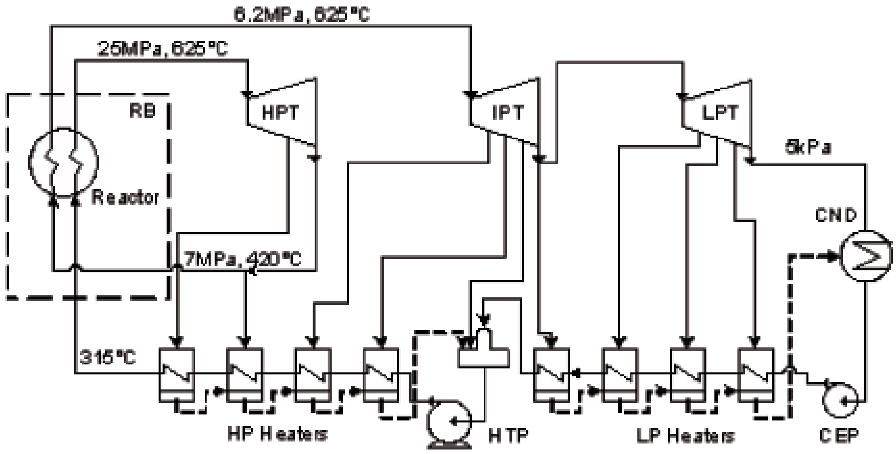
In the field of thermal-hydraulics and safety, significant gaps exist in the heat transfer and safety database for the SCWR. Data needed for thermal-hydraulics and safety analysis at prototypical SCWR conditions will be produced as part of the TH&S project.

In the field of materials and chemistry, the main objective is to select key in-core and out of core materials for the pressure tube and pressure vessel designs. The work includes the definition of a reference water chemistry, based on materials compatibility and radiolysis behavior at supercritical conditions.

Main activities and outcomes in 2008

In the field of system integration and assessment, the main activities were the development of pre-conceptual SCWR designs: CANDU-SCWR in Canada; High Performance Light Water Reactor (HPLWR) pursued by Euratom; thermal and fast spectrum SCWRs in Japan and some conceptual design work in the Republic of Korea.

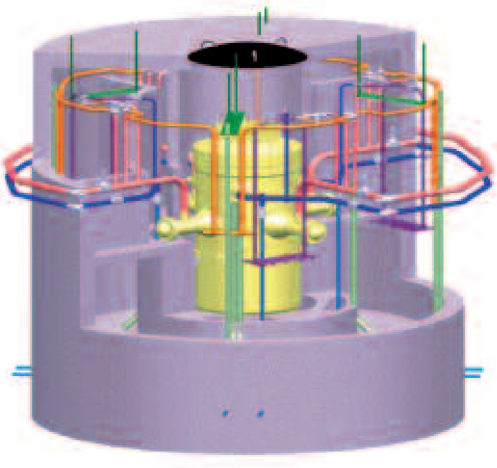
Figure 3.2: CANDU-SCWR reheat channel option



Canada is pursuing a pressure-tube based design with a thermal spectrum that uses conventional UO_2 or thorium fuel (CANDU-SCWR). In 2008, activities in this field focused on optimizing the core parameters for the UO_2 fuel option. Burn-up and coolant void reactivity were computed using lattice pitch, fuel channel thickness, fuel channel insulator thickness, fuel channel insulator porosity, and fuel enrichment as variable parameters. Following success of this method to choose an optimum core and lattice arrangement, optimization was expanded to fuel scoping studies (e.g., use of thorium). In addition, work continued to investigate the use of reheat channels to match the core design to the latest advances in supercritical turbines used in fossil-fuelled power plants. In this concept, some of the channels in the core are used to reheat the low-pressure stream exiting the supercritical turbine (inlet conditions of 25 MPa and 625°C) to produce superheated steam at 625°C or higher but at subcritical pressure (Figure 3.2). Theoretically, efficiencies in excess of 50% are possible with this configuration. This work has resulted in a requirement to conduct new R&D to support the development of the steam-cooled channels.

The HPLWR design pursued by Euratom has a 25 MPa coolant pressure and 500°C or higher coolant temperature. Like in a boiling water reactor, the high-temperature steam is fed directly to the high-pressure turbine, so that a secondary heat-transfer circuit can be eliminated. In 2008, the first coupled neutronic/thermal-hydraulic analyses of the core were completed for full load and steady-state conditions. They showed that the envisaged power profile and coolant density distribution are feasible. Stress and deformation analyses of the reactor pressure vessel, the major reactor internals and the assembly boxes were completed and indicated areas for design optimization that will be performed in the next design iteration. A first design proposal, shown in Figure 3.3, has been worked out for the containment. It includes four redundant, active low-pressure coolant injection systems for residual heat removal, a pressure suppression pool and four flooding pools, an automatic depressurization system and a passive, high-pressure coolant injection system. First transient analyses of design basis accident have started. Stability analyses of coolant flow through the core have been completed showing that inlet orifices can avoid density wave oscillations in the core.

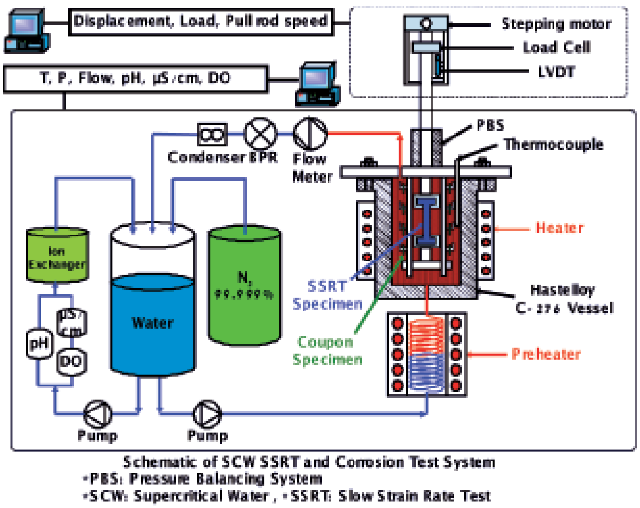
Figure 3.3: Containment and safety system design of the HPLWR (Euratom)



Japan is pursuing two designs (thermal and fast spectrum) for pressure vessel SCWRs. For the thermal-spectrum SCWR, target specifications of the fuel and core have been satisfied by improving both neutronics and thermal-hydraulic designs. Improvements in the fuel/core design have shown the possibility to keep the maximum cladding surface temperature, one of the most important specifications for design and fuel cladding material requirements, below 700°C under normal operating condition. The integrity of fuel cladding has been evaluated using recent material test data and found to be adequate. For the fast-spectrum SCWR, core design has been improved for negative local void reactivity and higher power density. Fuel rod behavior under normal operating condition has been analyzed. Thermal hydraulic behavior in narrow sub-channels has been analyzed. Nuclear transmutation capability of the fast spectrum SCWR and its spent fuel management have been analyzed. Mechanical analyses have been conducted to design and improve in-core structure. Plant control system has been designed and improved. Safety analyses have been conducted for abnormal transients

The Republic of Korea continued further assessment of a conceptual 1 400 MWe SCWR core design which contains 193 fuel assemblies with a typical four-batch fuel-loading pattern. The fuel assembly has a 21 x 21 rod array and is composed of 300 fuel rods, 25 cruciform-type solid moderator pins, and 16 single solid moderator pins. Due to high coolant density variation along the axial direction, an axial zoning of the fuel enrichments is introduced.

Figure 3.4: Corrosion and SCC test loop for the supercritical water environment (Republic of Korea)



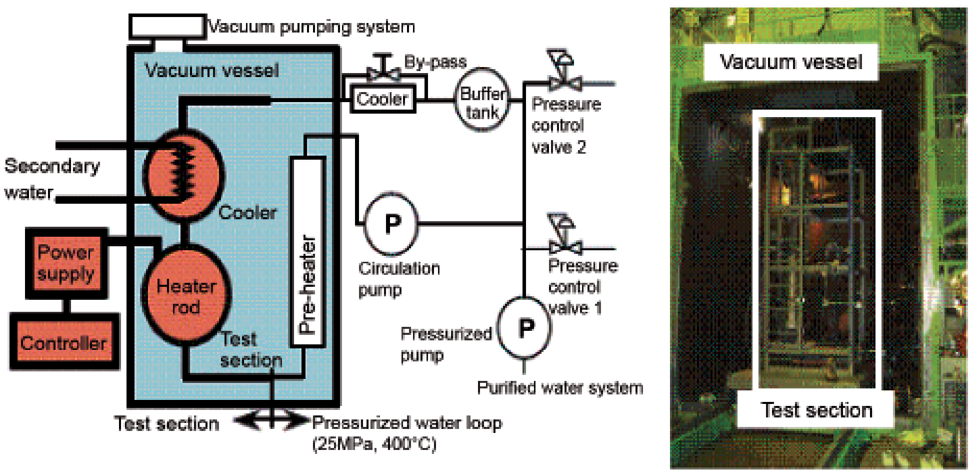
Regarding materials and chemistry, progress was made in 2008 in the areas of corrosion and stress corrosion cracking (SCC) testing, coatings (including material modifications), radiolysis, and modeling. Corrosion and SCC tests are being carried out to evaluate the suitability of existing materials for the SCWR. In 2008, all participants extended test capabilities to higher temperatures (up to 650°C) at supercritical pressures as illustrated in Figure 3.4. In addition, corrosion and SSC tests were conducted by all participants and new data were added to a data base which is being compiled and includes already some 90 alloys. The tests were conducted using pressurized capsules in addition to static autoclaves and flowing loops. General corrosion tests were conducted using many types of materials. Several surface analysis techniques were used to examine these materials following exposure to supercritical water. In addition, SCC tests were performed with three austenitic stainless steels and one ODS steel at 500°C and 25 MPa with 125 ppm dissolved oxygen. Results from these tests were used to update a data base and to plan for future tests. Agreement was reached to conduct more controlled experiments by all participants (round-robin tests).

Work on coatings involves the use of corrosion-resistant coatings on materials which exhibit good mechanical properties but have poor corrosion characteristics, as a back-up option if existing materials are not suitable at supercritical conditions. The preparation of several ceramic samples for preliminary evaluation in a static autoclave was pursued in 2008. In addition, Cr-coated samples, using advanced physical vapor deposition technique, were successfully tested and showed negligible corrosion.

Fundamental work, including experimental test and simulation, continued on the effect of radiation on supercritical water (radiolysis) in a large range of temperature and pressure. Experimental techniques involved the use of a picosecond pulse radiolysis method while molecular dynamics and Monte Carlo simulations were used to study radiolytic reactions. Manufacturing and assembly of the in-pile radiolysis and water chemistry loop at the Rez research center in the Czech Republic has been completed and the loop is ready for out-of-pile commissioning prior to the installation in the research reactor.

Other related activities included the evaluation of mechanical properties of several irradiated materials. High-temperature strength and creep strength, void swelling, helium embrittlement and phase stability have been evaluated by means of pressurized tube tests. The results of these tests have revealed that the creep deformation is dominated by thermal effects rather than irradiation effects at 700°C.

Figure 3.5: Schematic diagram of high temperature pressurized water loop and test section in vacuum vessel (Japan)



Regarding thermal-hydraulics, more tests were conducted at supercritical conditions using water and modeling fluids (Freon and CO_2). In addition, computational fluid dynamics (CFD) simulations were completed and compared to experiments. Tube heat transfer tests using water at supercritical conditions were also completed in a new loop (Figure 3.5), and they will be compared to Freon and CO_2 tests.

The physics of heat transfer deterioration in a supercritical water flow with low mass flux through a tube with high heat flux was studied using CFD. If the boundary layer is well resolved, and if physical properties of supercritical water are included properly in the analysis, the numerical simulation can model the observed phenomena with reasonable accuracy. A numerical study of turbulence enhancement by ribs on the heated wall indicates potential to avoid the deterioration of heat transfer.

Efforts are under way to perform tests in water using annuli and a technique to scan the surface temperature of the test section (rather than using fixed thermocouples at specified locations). If successful, this technique will make it possible to obtain much better coverage in heat transfer tests and will be valuable for investigating the occurrence of deteriorated heat transfer (or the avoidance of deteriorated heat transfer in bundles or on enhanced surfaces). Initial tests resulted in failure of the test section due to improper electrical insulation and overheating of components that were not cooled by design. The test section will be repaired using better insulation materials and testing will resume following the repair.

Other activities carried out in 2008 included participation of some members in a Coordinated Research Program of the International Atomic Energy Agency (IAEA) on heat transfer and safety analysis codes at supercritical conditions.

3.1.4 Gas-cooled Fast Reactor (GFR)

Main characteristics of the system

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 minimization (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, for hydrogen production for example).

The GFR uses the same fuel recycling processes as the SFR and the same reactor technology as the VHTR. Therefore, its development approach is to rely, in so far as feasible, on technologies developed for the VHTR for structures, materials, components and power conversion system. Nevertheless, it calls for specific R&D beyond the current and foreseen work on the VHTR system, mainly on core design and safety approach.

Status of cooperation

Following the signature of the SA at the end of 2006 by Euratom, France, Japan and Switzerland, two projects – on Conceptual Design & Safety and on Fuel, Core Materials and Fuel Cycle – are being considered. Project Arrangements have been prepared in 2008 and their signature by interested members is expected in 2009. The organizations involved in the preparatory activities are the Joint Research Centre of Euratom, the French CEA, JAEA from Japan and the Paul Scherrer Institute in Switzerland. Pending the signature of PAs, participating countries have been exchanging R&D results under the auspices of provisional PMBs.

R&D Objectives

The project on conceptual design and safety aims at studying core design and performance, overall system arrangement, components and materials. It is expected that it will include also investigations on a small experimental GFR that could be built in the coming decades.

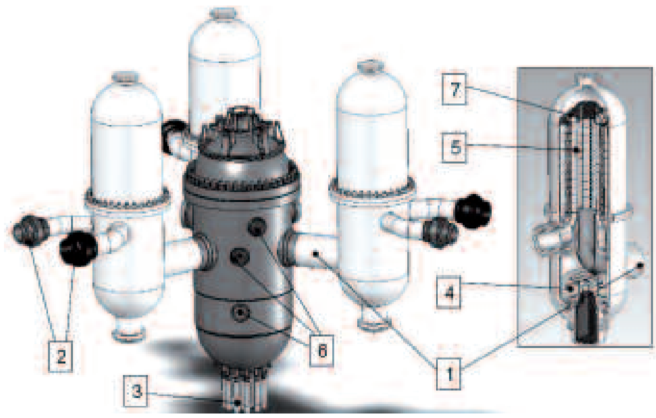
The goals of the fuel, core materials and fuel cycle project are to investigate fuel element design and qualification, ceramic material for cladding, and dense fuel material like carbide or nitride.

Main activities and outcomes

In February 2008, an international technical seminar was held in Paris, France, to present and discuss the main results obtained on the design and performance of the GFR system and its fuel. Eleven countries and international organisations participated in this seminar. It offered opportunities for extensive exchange of information between all interested research institutes and organizations.

Regarding the system design, preliminary studies have focused on the primary circuit and main components, namely the reactor vessel, the heat exchangers between the primary and secondary circuits, and the gas blowers. The reactor vessel considered is a thick metallic structure of large size. A martensitic chromium steel was selected, ensuring negligible creep at operating temperature. The global primary arrangement is based on three main loops of 800 MWth each, fitted with three units of a compact intermediate heat exchanger and a gas blower enclosed in a single vessel (see Figure 3.6).

Figure 3.6: GFR primary-system design

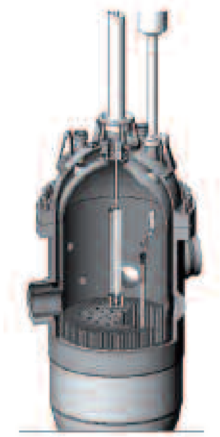


1	Primary cross-duct
2	Secondary pipes with isolating valves
3	Control Rod Drive Mechanisms
4	Primary blower and associated motor
5	Compact Heat Exchanger modules
6	Pipe connections for Decay Heat Removal systems
7	Primary isolation valve

A close containment has been designed to provide and maintain a back-up pressure in case of large gas leak from the primary system. It is a metallic structure, filled with nitrogen slightly over atmospheric pressure to reduce air ingress capabilities. This component limits the consequence of concomitant rupture of the first and second safety barriers (the fuel clad and the primary system). Specific loops for decay heat removal in case of emergency are directly connected to the primary circuit using a cross-duct piping, in extension of the pressure vessel, and are equipped with heat exchangers and forced convection devices. This system arrangement allows the residual power to be extracted in any accidental situations. In addition, thanks to the low pressure drop of the core design, a passive natural gas circulation can be used in several situations.

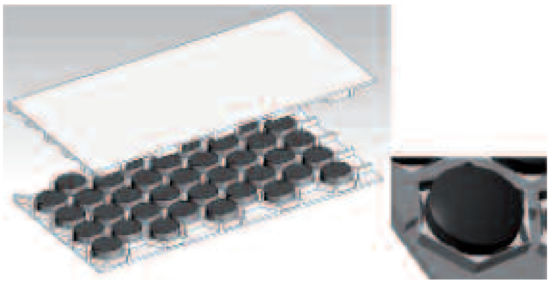
The fuel handling system considered in this concept is based on a jointed-arm system, with fuel element loading and unloading using a fuel storage drum via lock chambers, the vessel being closed (Figure 3.7). A dedicated forced convection device, located outside the reactor vessel, is designed to cool the spent fuel sub-assembly during its handling.

Figure 3.7: GFR fuel-handling system



The choices for the fuel element are designed to ensure an excellent neutron economy for plutonium breeding and minor actinide transmutation in spite of the relatively large volume dedicated to the gas coolant. At least two fuel concepts have the potential to fulfil the requirements: an innovative ceramic plate-type fuel element (see Figure 3.8); or a ceramic pin-type fuel element.

Figure 3.8: GFR plate-type fuel element with carbide fuel pellets

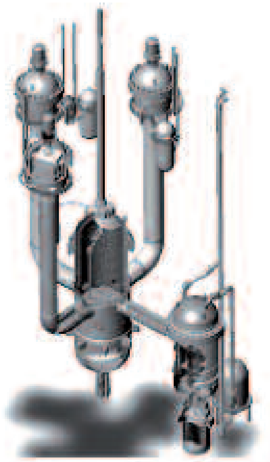


The main advantages of the plate-type fuel is that it optimizes the fuel temperature and burn-up, and that it allows a micro-confinement of the fission products cell-by-cell, expected to reduce radioactive release in case of an accidental scenario. The alternative pin-type fuel is a well-known concept and it is possibly less challenging from a technology standpoint but its design should be optimized in the future to improve performance.

Fuel pellet fabrication has been initiated at the Euratom Joint Research Centre and at CEA, in France. An irradiation test in the High Flux Reactor in Petten, the Netherlands, has been proposed and JAEA is looking for future irradiations tests in JOYO. In France, composite-matrix ceramics for cladding are tested up to high temperature to verify their capability to withstand severe reactor conditions.

Regarding activities pursued by Members of the GFR SA, technical studies of the gas-cooled fast reactor project carried within the 6th Euratom Framework Programme were completed at the end of 2007. The final project deliverables of the project include: a preliminary design and safety report for an experimental demonstration reactor; a forward program on GFR; and a final report on the project. Within the 7th Euratom Framework Programme, the viability phase of the European gas-cooled fast reactor should progress with an experimental demonstration and technology reactor named ALLEGRO (see Figure 3.9).

Figure 3.9: ALLEGRO – Experimental demonstration GFR proposal (Euratom)



With a thermal power around 80 MWth, ALLEGRO will not produce any electricity. It incorporates, at a reduced scale, all the architecture and the main materials and components foreseen for the GFR without the power conversion system. Its safety principles are those proposed for the GFR: core cooling through gas circulation in all situations, with a minimal pressure level in case of a leak being ensured by a specific guard containment surrounding the primary system. It will also mainly contribute to the development and qualification of an innovative refractory fuel element that withstands high temperatures, which is one of the key points to assess for the GFR system viability.

3.1.5 Lead-cooled Fast Reactor (LFR)

Main characteristics of the system

The LFR features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of all actinides from spent fuel and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of a relatively inert coolant. It has the potential to meet the electricity needs of remote sites as well as for large grid-connected power stations.

The designs that are currently proposed as candidates for international cooperation and joint development in the GIF framework are two pool-type reactors:

- the Small Secure Transportable Autonomous Reactor (SSTAR); and
- the European Lead-cooled System (ELSY).

The current reference design for the SSTAR in the United States is a 20 MWe natural circulation reactor concept with a small transportable reactor vessel (Figure 3.10). Specific features of the lead coolant, the nitride fuel containing transuranic elements, the fast spectrum core, and the small size combine to promote a unique approach to achieve proliferation resistance, while also enabling fissile self-sufficiency, autonomous load following, simplicity of operation, reliability, transportability, and a high degree of passive safety. Conversion of the core thermal power into electricity at a high plant efficiency of 44% is accomplished by utilizing a supercritical carbon dioxide Brayton cycle power converter.

Figure 3.10: SSTAR pre-conceptual design and operating parameters

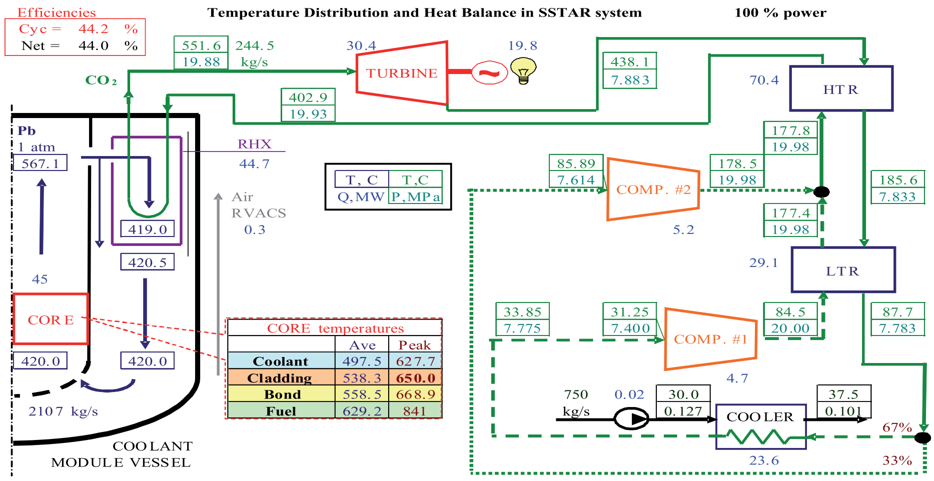
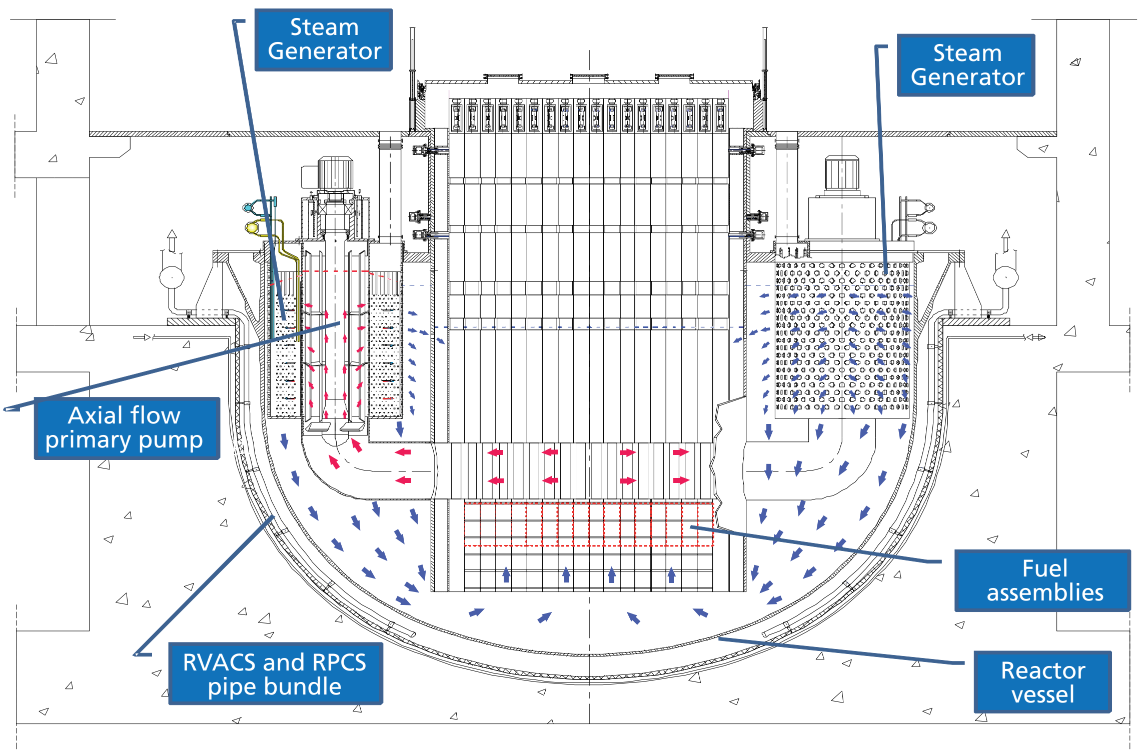


Figure 3.11: ELSY configuration



The ELSY reference design (Figure 3.11) is a 600 MWe reactor cooled by pure lead (Cinotti, *et al.*, 2008). This concept has been under development since September 2006 within the 6th Euratom Framework Programme. The ELSY project is being performed by a consortium consisting of nineteen organizations including seventeen from Europe, and two from the Republic of Korea. ELSY aims to demonstrate the possibility of designing a competitive and safe fast critical reactor using simple engineered technical features while fully complying with the mission identified in the GIF Roadmap of minor actinide burning capability. Typical design parameters of the SSTAR ans ELSY concepts are summarized in Table 3.3.

Table 3.3: Key design parameters of GIF LFR concepts

Parameters	SSTAR	ELSY
Power (MWe)	19.8	600
Convertion Ratio	~1	~1
Thermal efficiency (%)	44	42
Primary coolant	Lead	Lead
Primary coolant circulation (at power)	Natural	Forced
Primary coolant circulation for direct heat removal	Natural	Natural
Core inlet temperature (°C)	420	400
Core outlet temperature (°C)	567	480
Fuel	Nitrides	MOX, (Nitrides)
Fuel cladding material	Si-Enhanced Ferritic/Martensitic Stainless Steel	T91 (aluminized)
Peak cladding temperature (°C)	650	550
Fuel pin diameter (mm)	25	10.5
Active core dimensions Heigh/ equivalent diameter (m)	0.976/1.22	0.9/4.32
Working fluid	Supercritical CO_2 at 20 MPa, 552°C	Water-superheated steam at 18 MPa, 450°C
Primary/secondary heat transfer system	Four Pb-to-CO_2 HXs	Eight Pb-to-H_2O SGs
Primary pumps	-	Eight mechanical pumps integrated in the steam generators
Direct heat removal	Reactor Vessel Air Cooling System + Multiple Direct Reactor Cooling Systems	Reactor Vessel Air Cooling System + Four Direct Reactor Cooling Systems + Four Secondary Loops Cooling Systems

Status of cooperation

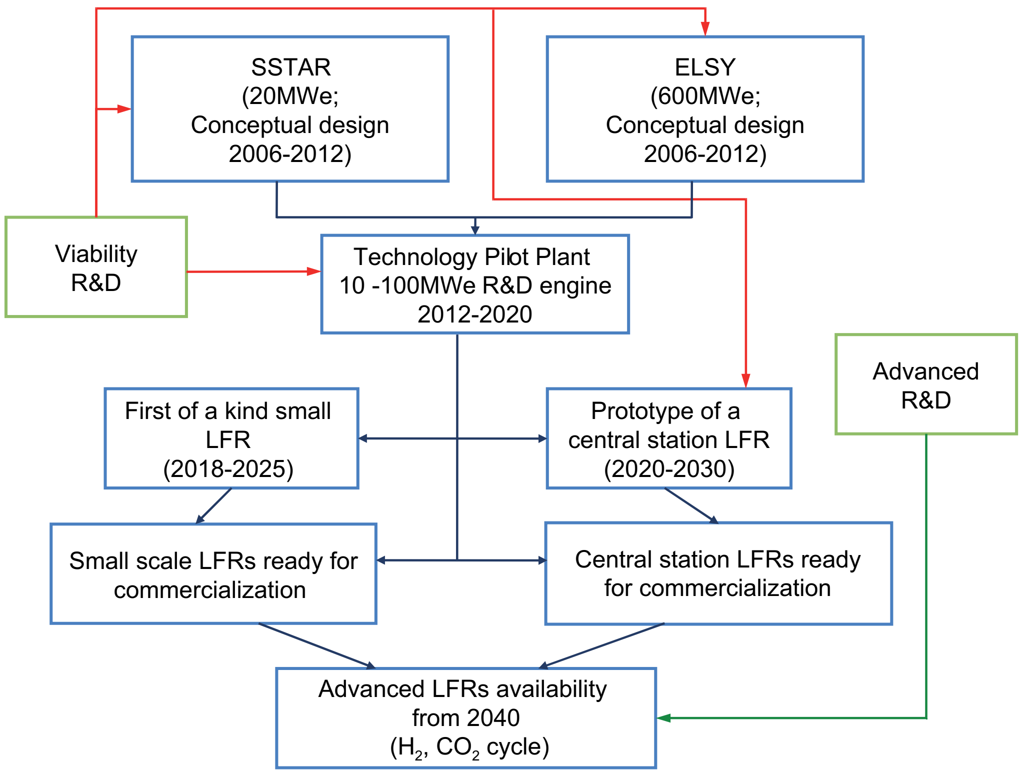
The cooperation on LFR within GIF was initiated in October 2004 and the first formal meeting of the provisional SSC was held in March 2005, with participation of representatives from Euratom, Japan, the Republic of Korea and the United States. The provisional SSC held periodical meetings to prepare a draft SRP which was reviewed by the EG in mid-2007 and mid-2008. In addition, informal meetings were held with representatives of the nuclear industry, research organizations and universities involved in LFR development.

R&D objectives and milestones

The draft SRP for the LFR is based on molten lead as the reference coolant and lead-bismuth as back-up option. The preliminary evaluation of the concepts will cover their performance in the areas of sustainability, economics, safety and reliability and proliferation resistance and physical protection. Given its R&D needs for fuel, materials, and corrosion control, the LFR system is expected to require a two-step industrial deployment: reactors operating with relatively low primary coolant temperature and low power density by 2025; and reactors with more advanced performance by 2035.

Figure 3.12 illustrates the basic approach recommended in the draft SRP. It portrays the dual track viability research program with convergence to a single, combined technology pilot plant leading to the eventual deployment of both types of systems.

Figure 3.12: Conceptual framework for the LFR R&D



The approach adopted aims at addressing the research priorities of each participant party while developing an integrated and coordinated research program to achieve common objectives and avoid duplication of effort. The integrated plan recognizes two principal technology tracks for pursuit of LFR technology:

- a small, transportable system of 10–100 MWe size that features a very long refueling interval; and
- a larger-sized system rated at about 600 MWe, intended for central station power generation and waste transmutation.

Following the successful operation of a demonstration plant around the year 2020, a prototype development effort is expected for the central station LFR leading to a subsequent industrial deployment. In the case of the small transportable (SSTAR) option the development of a first of a kind unit in the period 2018-2025 is foreseen. Because of the small size of the SSTAR it is expected that the main features can be established during the demonstration phase, and that it will be possible to move directly to industrial deployment without going through the prototype phase.

The design of the industrial prototype of the central station LFR and that of the first of a kind SSTAR should be planned in such a way as to start construction as soon as the pilot plant operation at full power has given the main assurances of the viability of this new technology.

In the SRP the viability R&D is organized in three areas: system design and assessment; fuel development; and lead technology and material. General issues of concern for LFR development include corrosion of structural materials, lead technology, in-service inspection, instrumentation, assessment of the steam generator tube rupture (SGTR) accident, fuel development, control rods operating in lead and refueling in lead.

Main activities and outcomes

In 2008, the activities have been largely devoted to large scale LFR development in Europe focusing on the conceptual design of the system reference configuration including the containment system, overall plant layout, core, steam generator units, primary pumps, decay-heat removal systems and refueling system.

The use of a compact and simple primary circuit, with the additional objective that all internal components be removable, are among the reactor features intended to assure competitive electricity generation and long-term investment protection. Simplicity is expected to reduce both the capital cost and the construction time; these are also supported by the compactness of the reactor building (reduced footprint and height). The reduced footprint would be possible due to the elimination of the intermediate cooling system, and the reduced height results from the design approach (reduced-height components).

One of the main objectives of ELSY from the beginning of the activity has been the identification of innovative solutions to reduce the primary system volume and the complexity of the reactor internals. The result is that most of the components are not conventional.

The steam generator, whose volume is about half of a helical-tube steam generator, is characterized by a spiral-wound tube bundle. The inlet and outlet ends of each tube are connected to the feed water header and steam header, respectively, both arranged above the reactor roof. An axial-flow primary pump, located inside the inner shell of the steam generator, provides the pressure required to force the coolant to enter from the bottom of the steam generator and to flow in a radial direction. This scheme is almost equivalent to a pure counter-current scheme, because the water circulates in the tube from the outer spirals towards the inner spiral, while the primary coolant flows in radial direction from the inside to the outside of the steam generator.

The core consists of an array of open fuel assemblies of square pitch surrounded by reflector-assemblies, a configuration that presents reduced risk of coolant flow blockage. An alternative solution with closed hexagonal fuel assemblies is retained as a back-up option. The upper part of the fuel assembly is peculiar to the novel ELSY design, because it extends well above the fixed reactor cover, and the fuel elements, whose weight is supported by lead, are fixed at their upper end in the cold gas space, well above the lead surface. This avoids the classical problem of a core support grid immersed in the coolant, which would complicate in-service inspection owing to the lead environment.

Considering the high temperature and lead environment, any approach that foresees the use of in-vessel refueling equipment would represent a tremendous R&D effort and substantial associated technical risk, especially because of the need to develop reliable bearings operating in lead, an unknown technology at present. For these reasons, the adopted design approach represents a real breakthrough. Installation of steam generators inside the vessel is the real challenge of a LFR design. In operation there is need for a sensitive and reliable leak detection system and a highly reliable depressurization and isolation system. Careful attention has been given also to the issue of mitigating the consequences of the SGTR accident to reduce the risk of pressurization of the primary boundary; to this end, innovative provisions have been conceived which make the primary system more tolerant to SGTR.

In the United States, an initial scoping investigation has been carried out into the viability of a near-term deployable LFR technology pilot plant/demonstration test reactor (demo) operating at low temperatures enabling the use of existing materials such as T91 ferritic/martensitic steel or Type 316 stainless steel shown in numerous worldwide tests conducted during the past decade to have corrosion resistance to lead-bismuth eutectic (LBE) at temperatures up to ~ 550°C with active oxygen control. Neutronic and system thermal-hydraulic analyses indicate that a 100 MWth lead-cooled metallic-fueled demonstration plant with forced flow and a 480°C core outlet temperature, supporting the development of both the ELSY and SSTAR LFRs, may be a viable concept. A supercritical water cycle with reheating using the primary lead coolant as a heat source can provide a power conversion cycle efficiency of 43% versus 41% for a supercritical CO_2 Brayton cycle power converter.

In Japan, research activities related to LFR focus on the properties and use of LBE, corrosion characteristics and corrosion behavior of the reactor coolant, the structural and cladding materials, and polonium behavior in the coolant system. The Tokyo Institute of Technology recently proposed research activities on several LFR systems including the CANDLE reactor that does not require movable reactivity control mechanisms (Sekimoto, 2008).

Two systems are developed in the Republic of Korea, the proliferation-resistant, environment-friendly, accident-tolerant, continual and economical reactor (Hwang, *et al*., 2006) and the BORIS (Kim, *et al*., 2006). In the Russian Federation, two systems are considered: the SVBR-75/100, a LBE-cooled modular fast reactor having a power range of 75 to 100 MWe (Zrodnikov, *et al*., 2006); and the BREST lead-cooled fast reactor concept with its associated fuel cycle (Adamov, *et al*., 2001).

3.1.6 Molten Salt Reactor (MSR)

Main characteristics of the system

In a Molten Salt Reactor, the fuel is dissolved in a fluoride salt coolant. The technology was partly developed in the 1950s and 1960s. Earlier MSRs were mainly considered as thermal-neutron-spectrum concepts. Compared with solid-fuelled reactors, MSR systems have lower fissile inventories, no radiation damage constraint on attainable fuel burn-up, no spent nuclear fuel, no requirement to fabricate and handle solid fuel, and a homogeneous isotopic composition of fuel in the reactor. These and other characteristics may enable MSRs to have unique capabilities and competitive economics for actinide burning and extending fuel resources.

With changing goals for advanced reactors and new technologies, there is currently a renewed interest in MSRs. The new technologies include: Brayton power cycles (rather than steam cycles) that eliminate many of the historical challenges in building MSRs; and the conceptual development of several fast-spectrum MSRs that have large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors.

The development of higher temperature salts as coolants would open new nuclear and non-nuclear applications (Forsberg *et al*., 2007). These salts are being considered for intermediate heat transport loops within all types of high-temperature reactor systems (helium and salt cooled) and for hydrogen production concepts, oil refineries and shale oil processing facilities, amongst other applications. For most of these applications, the heat would have to be transported over hundreds of meters to kilometers.

One of the concepts under consideration, the Advanced High-Temperature Reactor (AHTR) uses liquid salts as a coolant and the same graphite core structures with coated fuel particles as gas-cooled reactors such as the VHTR. The better heat transport characteristics of salts as compared with helium enable power densities 4 to 6 times greater and power levels up to 4 000 MWth with passive safety systems. The fuel cycle characteristics are essentially identical to those of the VHTR, while the power conversion and balance of plant are essentially identical to those of the "reference" MSR.

Status of cooperation

The decision for setting up a provisional System Steering Committee for the MSR was taken by the GIF Policy Group in May 2004. The participating members are Euratom, France and the United States. Other countries have been represented systematically (the Russian Federation) or occasionally (Japan) as observers in the meetings of the provisional SSC. Russia has played an important role in identifying R&D issues based on long-lasting programs initiated in the 1970s.

Beyond the GIF framework, the MSR provisional SSC has significantly contributed to enhance and harmonize international collaboration. A European network on MSR R&D has been active from 2001 until today. The major contribution of Euratom to MSR R&D within GIF has been the ALISIA (Assessment of LIquid Salts for Innovative Applications) project which was part of its 6th Framework Programme. A new MSR proposal has been submitted to the 7th Euratom Framework Programme, but has not been accepted yet.

Partners of the MSR provisional SSC are involved also in the Euratom-funded ISTC-3749 project, to be started in 2009 with official support from France, Germany, the Czech Republic, the United States and the IAEA. This project, subsequent to ISTC-1606, takes advantage of the large expertise and facilities existing in Russia.

R&D objectives

The renewal and diversification of interests in molten salts have led the MSR provisional SSC to shift the R&D orientations and objectives initially promoted in the original Generation IV Roadmap issued in 2002, in order to encompass in a consistent body the different applications envisioned today for fuel and coolant salts.

Two baseline concepts are considered which have large commonalities in basic R&D areas, particularly for liquid salt technology and materials behavior (mechanical integrity, corrosion):

- The Molten Salt Fast-neutron Reactor (MSFR) is a long-term alternative to solid-fuelled fast-neutron reactors offering very negative feedback coefficients and simplified fuel cycle. Its potential has been assessed but specific technological challenges must be addressed and the safety approach has to be established.

- The AHTR is a high temperature reactor with better compactness than the VHTR and passive safety potential for medium to very high unit power (> 2 400 MWth).

In addition, the opportunities offered by liquid salts for intermediate heat transport in other systems (SFR, LFR, VHTR) are investigated. Liquid salts offer two potential advantages: smaller equipment size because of the higher volumetric heat capacity of the salts; and no chemical exothermal reactions between the reactor, intermediate loop, and power cycle coolants.

Liquid-salt chemistry plays a major role in the viability demonstration, with such essential R&D issues as: the physico-chemical behavior of coolant and fuel salts, including fission products and tritium; the compatibility of salts with structural materials for fuel and coolant circuits, as well as fuel-processing material development; the on-site fuel processing; the maintenance, instrumentation and control of liquid-salt chemistry (redox, purification, homogeneity); and safety aspects, including interaction of liquid salts with sodium, water and air.

Those issues have been the basis for defining the following projects:

- Materials (selected as first priority).
- System design, operation and safety.
- Liquid salt chemistry and properties.
- Fuel and fuel cycle.
- System integration and assessment.

The factorization into projects emphasizes cross-cutting R&D areas. A major commonality is the understanding and mastering of fuel and coolant salt technologies, including development of structural materials, fuel and coolant clean-up, measurement of physical properties, chemical and analytical R&D.

Milestones

The MSR SRP describes the R&D program to establish the viability of the Molten Salt Reactor by 2018 and to optimize its design features as well as operating parameters by 2025. As such, it is intended to cover the needs of the viability and performance phases of the development plan described in the Technology Roadmap for the Generation IV Systems. The MSR SRP also accounts for a defined approach to establishing system baseline(s) and accomplishing system integration as needed.

The MSR provisional SSC has re-evaluated the milestones mentioned in the GIF Technology Roadmap owing to the peculiar and more innovative position of MSR among other Generation IV systems. This led to identify a scoping and screening phase (up to 2011), prior to the viability and performance phases, 2012-2017 and 2018-2025 respectively. The main milestones for the demonstration phase (final design, construction and operation of prototypes) have also been discussed, envisioning a MSR prototype after 2035. For the AHTR, the schedule is more compact, with a prototype planned to be in operation by 2031.

Main activities and outcomes

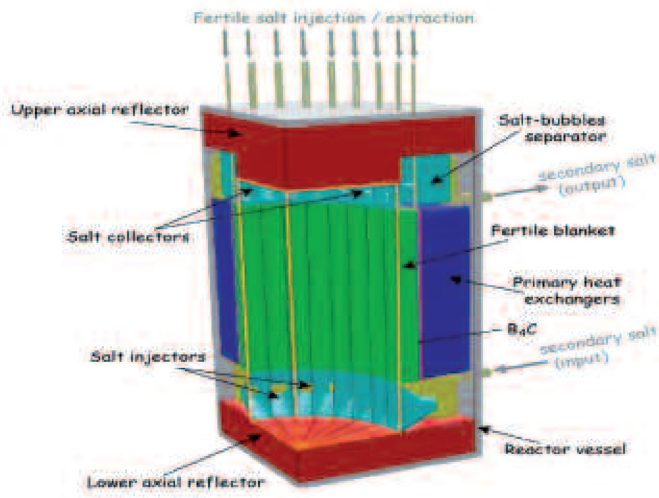
Significant progress was achieved in 2008, including:

- Development of MSFR pre-conceptual designs (France).
- Completion of salt selection for different applications and identification of the needs for complementary data (Euratom 7th Framework Programme, ISTC).

- Significant improvement of fuel salt clean-up scheme (France).
- Identification of candidate materials (Ni-W-Cr alloys) with very high corrosion resistance at temperatures above 750°C (Ignatiev *et al.*, 2008a).
- Demonstration of AHTR performance and safety (United States).
- Criticality tests for the assessment of AHTR fuel and core behavior (United States, Czech Republic).
- Better understanding of the transmutation capabilities, dynamics and safety-related parameters, for fertile and fertile-free fuel concepts (IAEA, Ignatiev *et al.*, 2008b).

The potential of the MSFR has been highlighted (Merle-Lucotte *et al.*, 2008a and 2008b). Realistic drawings showing the main components of the reactor and their arrangement in the vessel have been elaborated (Figure 3.13). In parallel, detailed thermal-hydraulics simulations are performed to evaluate the relevant range for operational parameters (temperature and pressure gradients, velocity distribution).

Figure 3.13: MSFR pre-conceptual design



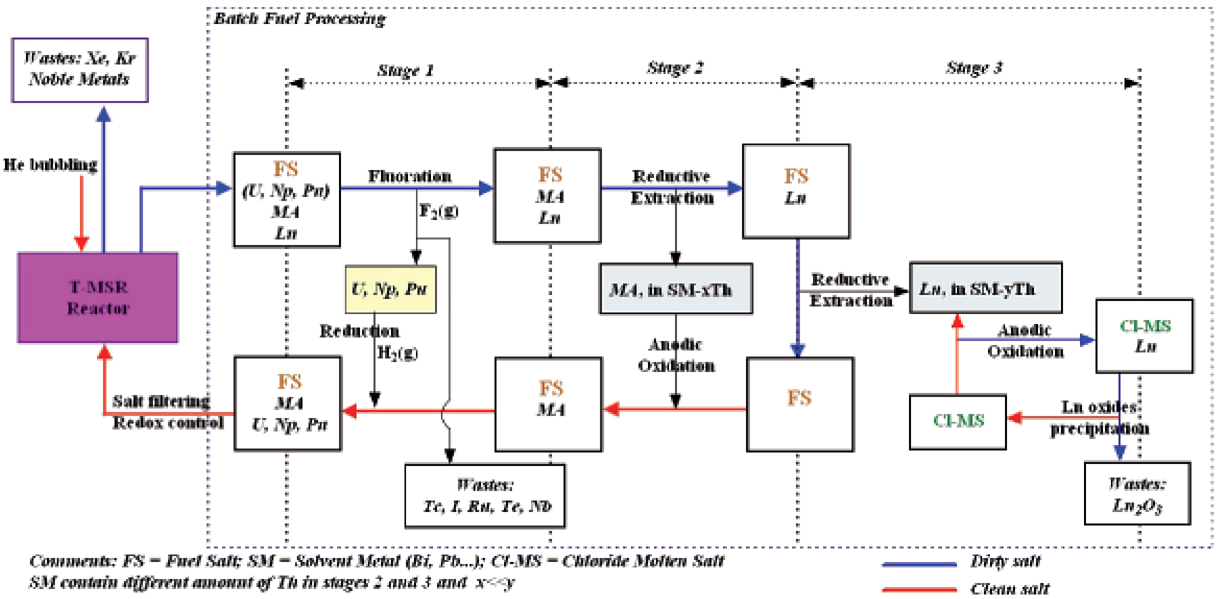
Salt systems have been critically reviewed and reference compositions have been proposed or confirmed, particularly within the ALISIA project (Table 3.4). In addition, alternative candidates can be envisioned and are under study (Benes *et al.*, 2008). Complementary data have been measured in the ISTC-1606 project conducted in Russia (Zherebetsov *et al.*, 2008). Missing or uncertain data for molten salt mixtures containing transuranic (TRU) elements have been identified (melting points, TRU solubility, thermal conductivity, expansivity) and are planned to be acquired within the European and ISTC-3749 projects.

Table 3.4: Main salts assessed in ALISIA project

Reactor type	Neutron spectrum	Application	Carrier salt	Fuel system
MSR-breeder	Thermal	Fuel	$LiF\text{-}BeF_2$	$LiF\text{-}BeF_2\text{-}ThF_4\text{-}UF_4$
	Non-moderated	Fuel	$^7LiF\text{-}ThF_4$	$^7LiF\text{-}ThF_4\text{-}UF_4$
MSR-breeder	Thermal/Non-moderated	Secondary coolant	$NaF\text{-}NaBF_4$	$^7LiF\text{-}ThF_4\text{-}PuF_3$
MSR-burner	Fast	Fuel	LiF-NaF	$LiF\text{-}(NaF)\text{-}AnF_4\text{-}AnF_3$
			$LiF\text{-}(NaF)\text{-}BeF_2$	$LiF\text{-}(NaF)\text{-}BeF_2\text{-}AnF_4\text{-}AnF_3$
			$LiF\text{-}NaF\text{-}ThF_4$	
AHTR	Thermal	Primary coolant	$^7LiF\text{-}BeF_2$	
SFR	Fast	Intermediate coolant	$NaNO_3\text{-}KNO_3\text{-}(NaNO_3)$	

The progress made in core design in the last two years has opened the door for the definition of an improved fuel salt reprocessing scheme (Figure 3.14) with a realistic fuel clean-up rate (40 l/day) and minimized losses to waste (Delpech *et al.*, 2008a and 2008c). The value of the clean-up rate is almost two orders of magnitude less than in the reference scheme.

Figure 3.14: MSFR fuel clean-up scheme



Critical steps of the two main stages (helium bubbling in the primary salt loop, on-site fuel processing) of the new fuel clean-up scheme are addressed and experimentally assessed in new facilities in France. An efficient technique for actinide/lanthanide separation is under qualification (Delpech *et al.*, 2008b).

The Pebble-Bed AHTR performance and safety is being assessed using thermal-hydraulic analyses and experimental facilities (Forsberg *et al.*, 2008b). Results from the pebble recirculation experiment are used to verify pebble injection into the reactor cold leg, lower plenum pebble landing dynamics and pebble defueling from the top of the reactor core. Thermal hydraulic analyses showed that the Pebble-Bed AHTR has very gentle response to loss of forced cooling transient, and can be designed to have acceptable response to anticipated transient without scram. Power levels up to 4 800 MWth can be foreseen.

The EROS project (Hron *et al.*, 2008) has generated the definition of a long-term collaboration program between the United States (University of California Berkeley) and the Czech Republic for the validation of AHTR neutronics models in the LR-0 zero power critical test facility (Figures 3.15 and 3.16). The initial test assembly was loaded in the LR-0 driver core on November 17, 2008. This initial design uses 60% natural LiF and 40% NaF salt. Subsequent experiments will use prototypical salt composition.

Figure 3.15: LR-0 zero power critical test facility

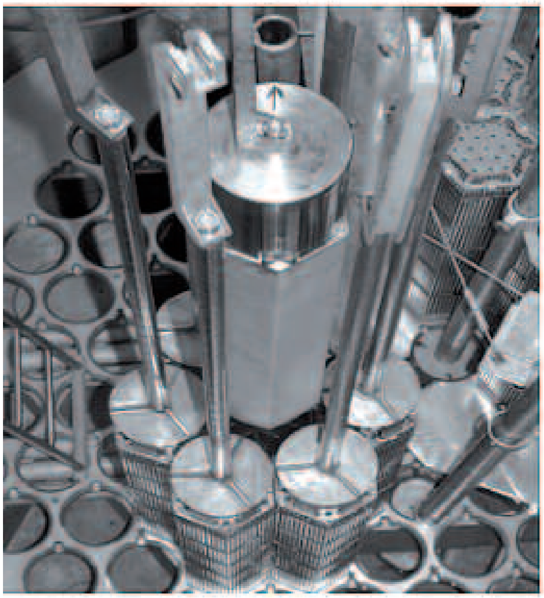
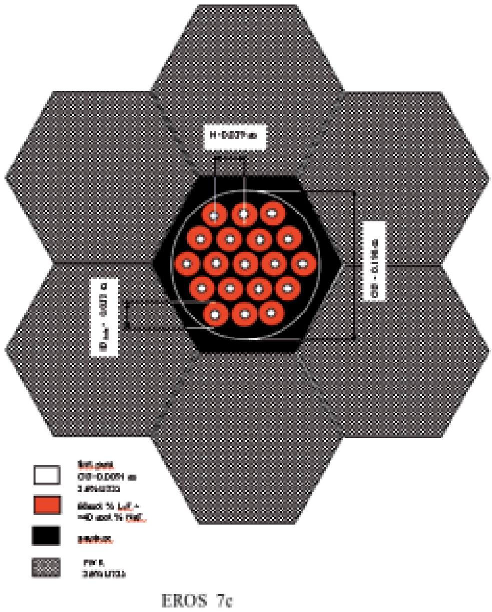


Figure 3.16: EROS test assembly



3.2 Assessment Methodologies

The three Methodology Working Groups (MWGs) of GIF – Economic Modeling (EMWG), Proliferation Resistance and Physical Protection (PRPPWG), and Risk and Safety (RSWG) – were established between late 2002 and early 2005. Their overall objective is to design and implement methodologies for evaluating the GIF systems against the goals defined in the *Technology Roadmap for Generation IV Nuclear Energy Systems* in terms of economics, proliferation resistance and physical protection, and safety.

3.2.1 Economic Assessment Methodology

The EMWG was formed in 2004 for developing a cost estimating methodology to be used for assessing GIF systems against the GIF economic goals. Its creation followed the recommendations from the Economics Crosscut Group of the Generation IV Roadmap Project that a standardized cost estimating protocol be developed to provide decision makers with a credible basis to assess, compare, and eventually select future nuclear energy systems, taking into account a robust evaluation of their economic viability.

The methodology developed by the EMWG is based upon the economic goals of Generation IV nuclear energy systems, as adopted by GIF, to have: a life cycle cost advantage over other energy sources (i.e., to have a lower levelized unit cost of energy on average over their lifetime); and a level of financial risk comparable to other energy projects (i.e., to involve similar total capital investment and capital at risk).

The methodology produced by EMWG consists of:

- A report describing the approach *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems, Rev. 4* (GIF/EMWG/2007/004);
- A computer tool *G4-ECONS Software Package*; and
- A *Users Manual for G4-ECONS Version 2.0* (GIF/EMWG/2007/005).

Sample calculations have been performed using the Cost Estimating Guidelines and the G4-ECONS software for both Generation III and Generation IV systems to demonstrate its validity.

By the very end of 2007, the EMWG, with the agreement of the GIF Experts and Policy Groups, released the methodology for public as well as GIF application. The complete methodology – Rev. 4 of the Guidelines (GIF/EMWG/2007/004), G4-ECONS Version 2.0, and its Users Manual (GIF/EMWG/2007/005) – is contained in a CD-ROM available from the NEA. To date, over 60 copies of the methodology CD have been provided upon request to various experts and organizations. In addition to GIF SSCs, the software has been requested by various IAEA groups and several Universities.

The EMWG monitors the use of the methodology and encourages feedback on its use and possible improvement. The NEA Technical Secretariat maintains a list of recipients of the methodology and keeps track of their comments and suggestions.

In 2008, EMWG members were invited in several meetings, including IAEA technical meetings and the VHTR SSC meeting in October, to present its methodology and provide guidance on economic assessments and the use of the G4-ECONS software. For this purpose, the Group developed a standard training package. The training presentation is modularized so as to be useful for presentations to senior management (executive level) as well as to analysts requiring detailed, user-oriented information.

The detailed user-oriented presentation was tested, in its draft version, at an IAEA meeting. The feedback from participants in the meeting provided guidance for preparing a final version which is available now for general use. Any member of the EMWG can present the training package whenever requested by GIF members or other interested users.

In parallel, the Group pursues testing of the methodology, and in particular of the G4-ECONS software on specific advanced systems in order to validate further the approach and provide future users with illustrative results.

In terms of enhancements of the methodology, the Group is focusing on the development of a more sophisticated approach to fuel cycle cost estimation that would be better adapted to advanced systems operated with closed fuel cycles for managing actinides.

3.2.2 Proliferation Resistance and Physical Protection Assessment Methodology

The role of the PRPPWG is to develop, implement and foster the use of an improved evaluation methodology to assess Generation IV nuclear energy systems with respect to GIF proliferation resistance and physical protection goals (see GIF Annual Report 2007).

In 2008, the PRPPWG focused its activities on:

- testing the methodology described in Revision 5 of the PR&PP Evaluation Methodology report which was released for unrestricted distribution in 2006 (www.gen-4.org/Technology/horizontal/PRPPEM.pdf); and
- interacting with System Steering Committees (SSCs) in order to initiate joint activities eventually leading to preliminary assessments of PR and PP aspects of the six systems under consideration in the framework of GIF.

The PR&PP methodology has found broader application, with its threat/system-response/consequence paradigm being applied in studies performed by Stanford, Princeton and Harvard universities, as well as internal studies performed by the USDOE National Nuclear Security Agency.

Testing of the methodology was pursued in 2008 by the PRPPWG through a case study. The main objective of the case study, initiated in 2007 and carried out through 2008, was to show how the methodology can provide useful feedback to designers at various stages of development of their concept, including at the pre-conceptual design stage. In particular, the case study illustrates the evaluation of the impacts of design variations on PR and PP performance for a given concept.

The system analyzed in the case study is a hypothetical Generation IV sodium fast reactor system – named Example Sodium Fast Reactor (ESFR) – including four medium size units (300 MWe each) with a shared dry fuel storage facility and co-located facilities for fuel reprocessing (pyro-chemical process) and fuel fabrication. The threat strategies considered in the case study include: concealed diversion of material; concealed misuse of the facility; breakout and overt diversion or misuse; and theft of weapon-usable material and sabotage of facility system elements. The status report issued in September 2008 on PR&PP evaluation of the full ESFR system provides preliminary findings and conclusions which will be finalized in the course of 2009.

With regard to the relevance of the PR&PP methodology for application to GIF nuclear energy systems, the case study demonstrated that the methodology can:

- Be applied at qualitative level in a traceable way, leading to accountable and dependable results.
- Analyze proliferation resistance of the system through a qualitative approach, providing useful results to system designers even when detailed design information is largely missing.
- Provide traceability of the analysis outcomes enabling a thorough review of the results and building confidence in their robustness.
- Identify small differences in the rationale and in the measure estimates through qualitative application of the methodology to a diversion or misuse scenario.
- Uncover within the breakout threat strategy a complex and non-intuitive metric for Proliferation Time which depends entirely on the assumed strategy of the proliferant state (which may be changing as political stresses evolve).
- Discern that for theft and sabotage longer response force times have the greatest impact on increasing adversary success when early detection probability in the pathway is low, and that the probability of adversary success decreases rapidly as response time decreases when early detection probability in the pathway is high.

- Determine, considering the proximity of theft and sabotage targets, that the ESFR facility will require a deterrence strategy regardless of the threat since the physical protection system will not be able to detect the adversary intent (theft versus sabotage) early enough. This will require a robust perimeter detection system and a response force deployed throughout the facility near the theft and sabotage target areas.

The status report also identified improvements to the methodology which could be implemented in the coming years and further tested through continuation of the present case study dedicated to the ESFR or through joint studies with SSCs focused on some GIF systems. In particular, the practical use of some proliferation resistance measures needs further investigations and the methodology should be able to capture the global response risk and the impact of foreign policy within its measures.

In accordance with its new Terms of Reference, the PRPPWG has strengthened its relations with Generation IV system designers, in particular with GIF SSCs, in order to promote early consideration of PR and PP issues in the development and design of those systems.

The PRPPWG has a well-established tradition of communication with users, initiated as early as November 2004 with a first Workshop for GIF designers and other stakeholders, held in Washington, DC, United States, and followed by two similar events held in Ispra, Italy, in June 2006 and in Tokyo, Japan, in November 2006. The main recent milestones in the implementation of joint activities with SSCs were: a Special Session for SSC representatives held during the 17th meeting of the PRPPWG in Marcoule, France, at the end of January 2008; a Workshop on Collaboration of the PRPPWG and SSCs held at Brookhaven National Laboratory, NY, United States, in May 2008; and a Seminar on PR&PP methodology held in Seoul, Republic of Korea, in October 2008, in connection with the 18th meeting of the PRPPWG.

The main goals of cooperative activities are: to make the SSCs more familiar with the PR&PP methodology and more sensitive to its usefulness and benefits in design work; to develop a working relationship between experts in the PR&PP methodology and GIF system designers; to develop and establish good PR&PP-related design principles; and to identify techniques and procedures to enhance effective application of the PR&PP methodology by users, including designers and decision makers.

3.2.3 Risk and Safety Assessment Methodology

In accordance with its Terms of Reference, the primary objective of the Risk and Safety Working Group is to promote a harmonized approach on safety, risk and regulatory issues in the development of Generation IV systems.

After its initial meeting in 2005, 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. During 2008, following a period of review and comment by the Experts Group and the Policy Group, the RSWG finalized 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 improvement;
- basic principles for an approach applicable to the design and the assessment of innovative systems including the ways to assess the adequacy of the defense-in-depth principle application and especially to address the treatment of severe plant conditions;
- role of passive features; and
- role of the Probabilistic Safety Assessment (PSA) and other existing analysis approaches, and the need for developing innovative indicators and tools.

Other issues that are still open for resolution include:

- of specific rules for the detailed design and the assessment of the design extension conditions (e.g. the severe plant conditions);
- an agreed way for the integration of physical protection concerns; and
- an agreed approach to address internal and external hazards in a more coherent way.

During 2008, the work of the RSWG turned to the development of a safety assessment methodology that will help guide the development of Generation IV systems. The methodology will be structured to answer the needs for probabilistic safety analysis, and will incorporate and integrate several additional elements. These include the use of the Phenomena Identification and Ranking Table, the Objective Provision Tree, extensive deterministic and phenomenological modeling, and the use of a matrix-based approach to documenting provision with desirable safety attributes. It is intended that certain elements of the integrated assessment methodology are applied, initially, at specific points in the evolution of a particular Generation IV design, and that, in a general sense, the level of detail and sophistication of the safety assessment increases as the design concept matures. Significantly, results obtained from early application of the methodology will help drive the design toward improved levels of safety based on an understanding of design vulnerabilities in the early stages.

As the RSWG works to develop this safety assessment methodology, it will seek to ensure that the following characteristics are embodied in it:

- As appropriate, the methodology must be consistent with the RSWG safety philosophy set out in its report entitled Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems, and with other relevant work previously performed by IAEA, national nuclear regulators, and others.
- To the extent practical, the methodology should consist of, or be based on, existing tools that are widely accepted for their validity.
- The methodology must be transparent, understandable and efficient.
- It must be practical and flexible, allowing for a graded approach to a range of technical issues of varying complexity and importance.
- The methodology must allow for explicit consideration and characterization of various sources and types of uncertainties.

In the future, the work of the RSWG will focus on formulating and documenting the assessment methodology in detail, working through a host of technical issues associated with the methodology, developing and demonstrating sample applications to selected hypothetical and practical problems, and working closely with the SSCs and the SIAP to facilitate successful application of the assessment methodology in the development of the respective Generation IV concepts.

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GIF has established relations with other major international endeavors aiming at the development of advanced nuclear energy systems and, more broadly, at enhancing the contribution of the nuclear option to sustainable energy supply. The increasing interest of policy makers in nuclear energy has triggered many multinational initiatives in the field of its peaceful applications. Exchange of information among those initiatives is a prerequisite to ensure their global effectiveness. GIF has been very attentive since its inception to collaboration with other projects. As GIF activities in the field of R&D on advanced systems are progressing, GIF Members place a high priority on strengthening cooperation with other international projects which have complementary objectives and scopes.

Within most GIF bodies, work programs include specific tasks devoted to cooperation with other projects. Through continued exchange of information and participation on an ad hoc basis in meetings of other projects, GIF ensures coordination whenever appropriate in order to avoid duplication of efforts that would lead, for members contributing to more than one of those endeavors, to wasting time and money and delaying the achievement of major milestones before reaching the goals.

The following sections describe briefly the interactions during 2008 of GIF with the three international projects which are the most relevant for GIF activities at present – the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), the Global Nuclear Energy Partnership (GNEP), and the Multinational Design Evaluation Program (MDEP).

4.1 International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO)

The INPRO (www.iaea.org/INPRO/) initiative started in 2001 under the auspices of the IAEA which ensures its management. Its main objective is to support the safe, sustainable, economic and proliferation-resistant use of nuclear technology to meet the global energy needs of the 21st century. It has 28 members (as of November 2008) participating in its various collaborative projects as well as in joint programs of work in different fields such as methodologies for evaluating innovative nuclear systems and user requirements for those systems.

The missions and activities of INPRO are broader than those of GIF but in many areas the two projects have complementary roles offering potential for creating fruitful synergies. All countries that are members of GIF are also members of INPRO. Therefore, the flow of information between INPRO and GIF is straightforward and its effectiveness relies mainly on representatives of countries participating in both endeavors. In addition, the Technical Secretariats of GIF and INPRO have established mechanisms to facilitate contacts including the organization of joint meetings to exchange information on progress and to help coordinate and cross fertilize the activities of the two projects.

While GIF focuses on R&D and methodologies applicable for system development, INPRO efforts cover infrastructure and institutional aspects, methodologies to assess innovative nuclear systems, and assistance to IAEA Member States for the development and implementation of those systems. One of the goals of INPRO is to offer a forum for sharing viewpoints between holders and users of nuclear technologies, aiming at jointly achieving innovations in nuclear reactors and fuel cycles responding to the requirements of the 21st century.

The areas where exchanges and cooperation between GIF and INPRO are the most relevant are methodologies and user requirements. The comparison and eventual harmonization of methodological approaches adopted have been identified by members of both projects as a key element for cooperation. With regard to user requirements, INPRO can provide GIF technology holders/developers with insights on the needs of future technology users. Collaboration between GIF and INPRO is ongoing also within selected research projects of common interest.

In 2008, the collaboration between GIF and INPRO was pursued through participation of members of the IAEA/INPRO team in the GIF Methodology Working Groups meetings and activities as well as in the GIF Policy and Experts Group meetings. Furthermore, an interface meeting was held in Vienna, Austria, in February to discuss the main outcomes of both initiatives and identify areas for cooperation in the coming years. Methodologies were identified again as the main topic for cooperative activities and, for example, an ongoing effort is devoted to harmonization in the field of proliferation-resistance approaches.

4.2 Global Nuclear Energy Partnership (GNEP)

GNEP is an initiative launched in 2006 by the US Department of Energy that currently brings together nearly fifty participants, including: 25 partners; 3 intergovernmental organizations which are permanent observers (the IAEA, GIF and Euratom); and a large number of observer countries (www.gnep.energy.gov/). The main goal of GNEP is to accelerate development and deployment of nuclear energy worldwide as a means to provide secure and environmentally friendly energy supply, while reducing the risk of proliferation. Cooperation between GNEP, GIF and INPRO is affirmed in the GNEP Statement of Principles.

GNEP focuses on the development of advanced fuel cycle technologies adapted to the management of transuranic elements in order to reduce the long-term burden and stewardship associated with high-level radioactive waste management and disposal. In addition, one of the major goals of GNEP is to ensure security of nuclear fuel supply for all countries, including those that do not wish to build and operate domestic facilities for all the steps of the fuel cycle. It is expected that this objective will be pursued through establishing a framework and infrastructure for international supply of fuel services and developing reactors specifically designed for developing countries.

While GNEP encompasses a broader policy vision than GIF, which focuses on technology progress through collaboration within specific R&D projects, both endeavors have similar goals for future nuclear systems, most notably improvement of waste management and enhancement of proliferation resistance. Objectives of both GNEP and GIF call for the development of advanced technologies, including fast neutron reactors, for the treatment and recycling of spent fuel.

GIF was represented by its Chairman in the meeting of the Executive Committee of GNEP, which is its highest policy body, held in October 2008 in Paris, France. The GNEP Working Groups on Infrastructure Development and Reliable Nuclear Fuel Cycle Services have made significant progress in 2008 and shared the outcomes from their activities with the international community, including GIF. In the field of proliferation resistance, GNEP and GIF have similar goals and the evaluation methodology developed by GIF has proven to be valuable in some of the analyses carried out within GNEP.

4.3 Multinational Design Evaluation Programme (MDEP)

MDEP is a "multinational initiative taken by national safety authorities to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities who will be tasked with the review of new reactor power plant designs" (MDEP Terms of Reference, www.nea.fr/mdep/mdep_ToR.pdf). The main objective of the MDEP effort is to enable increased cooperation and establish reference regulatory practices to enhance the safety of new reactor designs. The activities undertaken within MDEP include the implementation of products to facilitate licensing of new reactors, including those being developed by GIF.

The terms of reference of the MDEP state that its Steering Technical Committee "will interact as needed with GIF and INPRO to ensure effective communication and alignment with activities in similar areas." The NEA, which serves as Technical Secretariat for MDEP as well as for GIF, facilitates exchange of information and realization of synergies between MDEP and GIF, and in particular the interface with the GIF Risk and Safety Working Group.

MDEP is expected ultimately to facilitate the licensing of new reactor designs in different countries through sharing the resources and knowledge of national regulatory authorities assessing new reactor designs, thereby improving the efficiency and effectiveness of the regulatory process. MDEP members are: Canada, Finland, France, Japan, the People's Republic of China, the Republic of Korea, the Republic of South Africa, the Russian Federation, the United Kingdom and the United States. All have signed the GIF Charter except Finland which nevertheless participates in GIF through Euratom. The IAEA, which participates in GIF as an observer, also takes part in the work of MDEP.

The MDEP pilot project report (www.nea.fr/mdep/mdep_pilot_project_report.pdf), issued in May 2008, provides a summary of the findings from the first phase of MDEP activities and an outlook of its future work program. This revised program, which reflects lessons learnt during the pilot project phase, includes two main activities, on design-specific topics and on issue-specific topics, respectively.

In order to achieve its long-term goals, MDEP will focus first on cooperation and convergence of regulatory practices that will eventually develop into convergence of regulatory requirements. Progress towards harmonized regulatory practices and requirements for Generation IV reactor designs will be a natural outcome from the work to be undertaken within MDEP. Obvious synergies exist between GIF activities on risk and safety approach and the MDEP program of work. Therefore, a continued exchange of information will be established between the two projects, each of them benefiting from relevant progress and findings of the other.

A.1.1 Technology Goals of GIF

Eight technology goals have been defined for Generation IV systems in four broad areas: sustainability, economics, safety and reliability, and proliferation resistance and physical protection (see Box A.1, excerpts from www.gen-4.org/PDFs/GenIVRoadmap.pdf). These ambitious goals are shared by a large number of countries as they aim at responding to the economic, environmental and social requirements of the 21st century. They establish a framework and identify concrete targets for focusing GIF R&D efforts.

Box A.1. Goals for Generation IV Nuclear Energy Systems

Sustainability-1	Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objectives and provides long-term availability of systems and effective fuel utilization for worldwide energy production.
Sustainability-2	Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment.
Economics-1	Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
Economics-2	Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
Safety and Reliability-1	Generation IV nuclear energy systems operations will excel in safety and reliability.
Safety and Reliability-2	Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
Safety and Reliability-3	Generation IV nuclear energy systems will eliminate the need for offsite emergency response.
Proliferation Resistance and Physical Protection	Generation IV nuclear energy systems will increase the assurance that they are very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

These goals guide the cooperative R&D efforts undertaken by GIF Members. The challenges raised by GIF goals are intended to stimulate innovative R&D covering all technological aspects related to design and implementation of reactors, energy conversion systems, and fuel cycle facilities.

In light of the ambitious nature of the goals involved, international cooperation is considered essential for a timely progress in the development of Generation IV systems. This cooperation makes it possible to pursue multiple systems and technical options concurrently and to avoid any premature down selection due to the lack of adequate resources at the national level.

A.1.2 GIF Systems

The goals adopted by GIF provided the basis for identifying and selecting six nuclear energy systems for further development. The selected systems rely on a variety of reactor, energy conversion and fuel cycle technologies. Their designs feature thermal and fast neutron spectra, closed and open fuel cycles as well as a wide range of reactor sizes from very small to very large. Depending on their respective degrees of technical maturity, the Generation IV systems are expected to become available for commercial introduction in the period around 2030 or beyond. The path from current nuclear systems to Generation IV systems is described in a 2002 Roadmap Report entitled "A Technology Roadmap for Generation IV Nuclear Energy Systems" (www.gen-4.org/PDFs/GenIVRoadmap.pdf).

All Generation IV systems aim at performance improvement, new applications of nuclear energy, and/or more sustainable approaches to the management of nuclear materials. High-temperature systems offer the possibility of efficient process heat applications and eventually hydrogen production. Enhanced sustainability is achieved primarily through the adoption of a closed fuel cycle including the reprocessing and recycling of plutonium, uranium and minor actinides in fast reactors and also through high thermal efficiency. This approach provides a significant reduction in waste generation and uranium resource requirements. Table A.1.1 summarizes the main characteristics of the six Generation IV systems.

Table A.1.1 : Overview of Generation IV Systems

System	Neutron spectrum	Coolant	Temp. °C	Fuel cycle	Size (MWe)
VHTR (Very-High-Temperature Reactor)	thermal	helium	900-1 000	open	250-300
SFR (Sodium-cooled Fast Reactor)	fast	sodium	550	closed	30-150 300-1 500 1 000-2 000
SCWR (Super-Critical Water-cooled Reactor)	thermal/ fast	water	510-625	Open/ closed	300-700 1 000-1 500
GFR (Gas-cooled Fast Reactor)	fast	helium	850	closed	1 200
LFR (Lead-cooled Fast Reactor)	fast	lead	480-800	closed	20-180 300-1 200 600-1 000
MSR (Molten Salt Reactor)	fast/ thermal	fluoride salts	700-800	closed	1 000

VHTR – The very-high-temperature reactor is a further step in the evolutionary development of high-temperature reactors. The VHTR is a helium-gas-cooled, graphite-moderated, thermal neutron spectrum reactor with a core outlet temperature higher than 900°C, and a goal of 1 000°C, sufficient to support high temperature processes such as production of hydrogen by thermo-chemical processes. The reference thermal power of the reactor is set at a level that allows passive decay heat removal, currently estimated to be about 600 MWth. The VHTR is useful for the cogeneration of electricity and hydrogen, as well as to other process heat applications. It is able to produce hydrogen from water by using thermo-chemical, electro-chemical or hybrid processes with reduced emission of CO_2 gases. At first, a once-through LEU (<20% ^{235}U) fuel cycle will be adopted, but a closed fuel cycle will be assessed, as well as potential symbiotic fuel cycles with other types of reactors (especially light-water reactors) for waste reduction purposes. The system is expected to be available for commercial deployment by 2020.

SFR – The sodium-cooled fast reactor system uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction. It features a closed fuel cycle for fuel breeding and/or actinide management. The reactor may be arranged in a pool layout or a compact loop layout. The reactor-size options which are under consideration range from small (50 to 300 MWe) modular reactors to larger reactors (up to 1 500 MWe). The two primary fuel recycle technology options are advanced aqueous and pyrometallurgical processing. A variety of fuel options are being considered for the SFR, with mixed oxide preferred for advanced aqueous recycle and mixed metal alloy preferred for pyrometallurgical processing. Owing to the significant past experience accumulated with sodium cooled reactors in several countries, the deployment of SFR systems is targeted for 2020.

SCWR – Supercritical water-cooled reactors are a class of high-temperature, high-pressure water-cooled reactors operating with a direct energy conversion cycle and above the thermodynamic critical point of water (374°C, 22.1 MPa). The higher thermodynamic efficiency and plant simplification opportunities afforded by a high-temperature, single-phase coolant translate into improved economics. A wide variety of options are currently considered: both thermal-neutron and fast-neutron spectra are envisaged; and both pressure vessel and pressure tube configurations are considered. The operation of a 30 to 150 MWe technology demonstration reactor is targeted for 2022.

GFR – The gas-cooled fast reactor combines the advantages of a fast neutron core and helium coolant giving possible access to high temperatures. It requires the development of robust refractory fuel elements and appropriate safety architecture. The use of dense fuel such as carbide or nitride provides good performance regarding plutonium breeding and minor actinide burning. A technology demonstration reactor needed for qualifying key technologies could be in operation by 2020.

LFR – The lead-cooled fast reactor system is characterized by a fast-neutron spectrum and a closed fuel cycle with full actinide recycling, possibly in central or regional fuel cycle facilities. The coolant may be either lead (preferred option), or lead/bismuth eutectic. The LFR may be operated as: a breeder; a burner of actinides from spent fuel, using inert matrix fuel; or a burner/breeder using thorium matrices. Two reactor size options are considered: a small 50-150 MWe transportable system with a very long core life; and a medium 300-600 MWe system. In the long term a large system of 1 200 MWe may be envisaged. The LFR system may be deployable by 2025.

MSR – The molten-salt reactor system embodies the very special feature of a liquid fuel. MSR concepts, which may be used as efficient burners of transuranic elements from spent light-water reactor (LWR) fuel, also have a breeding capability in any kind of neutron spectrum ranging from thermal (with a thorium fuel cycle) to fast (with a uranium-plutonium fuel cycle). Whether configured for burning or breeding, MSRs have considerable promise for the minimization of radiotoxic nuclear waste.

List of Abbreviations and Acronyms

AF	Advanced Fuel [SFR signed Project]
AHTR	Advanced High-Temperature Reactor
ALISIA	Assessment of LIquid Salts for Innovative Applications
ANTARES	AREVA New Technology based on Advanced gas-cooled Reactors for Energy Supply
AVR	Arbeitsgemeinschaft Versuchsreaktor
CEA	Commissariat à l'énergie atomique [France]
CDBOP	Component Design and Balance-Of-Plant [SFR signed Project]
CD & S	Component Design and Safety [GFR Project]
CFD	Computational Fluid Dynamics
CM	Materials and Chemistry [SCWR Project]
CMVB	Computational Methods Validation and Benchmarking [VHTR Project]
CNRS	Centre National de la Recherche Scientifique [France]
DOE	Department Of Energy [United States]
EG	Experts Group
ELSY	European Lead-cooled SYstem
EMWG	Economic Modeling Working Group
EROS	Experimental zeRO power Salt reactor [Czech Republic Project]
ESFR	Example Sodium Fast Reactor
FA	Framework Agreement
FCMFC	Fuel, Core Materials and Fuel Cycle
FFC	Fuel and Fuel Cycle [VHTR signed Project]
FQ	Fuel Qualification
FZK	ForschungsZentrum Karlsruhe [Germany]
GACID	Global Actinide Cycle International Demonstration [SFR signed Project]
GIF	Generation IV International Forum
GFR	Gas Fast Reactor
GTHTR300C	Gas Turbine High Temperature Reactor 300 for Cogeneration [Japan]
GT-MHR	Gas Turbine-Modular Helium Reactor [United States]

HP	Hydrogen Production [VHTR signed Project]
HPLWR	High Performance Light Water Reactor
HTR-PM	High temperature gas-cooled reactor power generating module [China]
HTR-10	High temperature gas-cooled test reactor with a 10 MWth capacity [China]
HTTR	High Temperature Test Reactor [Japan]
IAEA	International Atomic Energy Agency
ISTC	International Science & Technology Center
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Center [Euratom]
LFR	Lead Fast Reactor
MA	Minor Actinides
MAT	Materials [VHTR Project]
MSFR	Molten Salt Fast Reactor
MSR	Molten Salt Reactor
MWG	Methodology Working Group
NEA	Nuclear Energy Agency [OECD]
NGNP	New Generation Nuclear Plant [United States]
NHDD	Nuclear Hydrogen Development and Demonstration [Republic of Korea]
NRI	Nuclear Research Institute [Rez, Czech Republic]
ODS	Oxide Dispersion-Strengthened
ORNL	Oak Ridge National Laboratory [United States]
PA	Project Arrangement
PBMR	Pebble Bed Modular Reactor
PG	Policy Group
PMB	Project Management Board
PP	Physical Protection
PR	Proliferation Resistance
PRPPWG	Proliferation Resistance and Physical Protection Working Group
PYCASSO	PYrocarbon irradiation for Creep And Shrinkage/Swelling on Objects [Euratom]
R&D	Research and Development
RSWG	Risk and Safety Working Group
SA	System Arrangement
SCC	Stress Corrosion Cracking
SSC	System Steering Committee
SCWR	Super-Critical Water Reactor
SGTR	Steam Generator Tube Rupture
SFR	Sodium Fast Reactor
SIA	System Integration and Assessment [Project]
SIAP	Senior Industry Advisory Panel
SO	Safety and Operation [SFR Project]
SRP	System Research Plan
SSTAR	Small Secure Transportable Autonomous Reactor

TH & S	Thermal-Hydraulics and Safety
THTR	Thorium High Temperature Reactor
TRISO	Tristructural isotopic [nuclear fuel]
TS	Technical Secretariat
TRU	Transuranic
VHTR	Very-High-Temperature Reactor

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This second edition of the GIF Annual Report highlights the main achievements of the Forum in 2008. It gives insights on the research plans being implemented by institutes and laboratories contributing to this international endeavor and illustrates the challenges that they have to face for the development and implementation of nuclear energy systems of the fourth generation. Its publication occurs in a period of renewed interest for the nuclear option worldwide. Nuclear energy is back on the agendas of policy makers and increasingly recognized as a means to address global climate change and security of supply issues. In this context, GIF can play a major role for the design of nuclear energy systems responding better to social requirements of the 21^{st} century owing to enhanced performance in the fields of safety and reliability, proliferation resistance and physical protection, and economics.

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