GEN IV International Forum℠

2007 Annual Report

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MESSAGE FROM THE CHAIRMAN

It is indeed a great privilege as well as a great honour for me to present this 2007 Annual Report:

• A great privilege because it is the first such report issued by the GIF, which offers me an excellent opportunity to assess how far the GIF has gone with respect to its initial goals six years after its launch, with a special focus on the Forum's accomplishments over the year 2007.

• A great honour because, in history, apart from one or two examples, mainly in the aerospace field, there exists no cooperation program of a scale as wide as the one induced by the GIF. This is a clear sign that the goal pursued by this initiative relates to a concern shared worldwide: meeting the growing energy demand while addressing the issue of climate change. The concept of sustainable development has become more and more prominent in policy making over the years, and addressing energy issues is essential to ensure the viability of such a concept. As Chairman of the GIF for more than a year, I realize even more the responsibilities and duties which lie on our shoulders, as well as the hopes our achievements will help fulfil.

The GIF brings the considerable resources of its member nations, and of most of the nuclear system designers and suppliers, to bear on the need to expand nuclear energy worldwide as a sustainable and reliable energy supply.

The Technology Roadmap provides the Forum with the main technical guidelines on the R&D necessary to advance the six most promising Generation IV systems to technical maturity.

I first of all would like to recall, among the major achievements of the Forum in the recent past, the signing in November 2006 of the VHTR, GFR and SCWR System Arrangements, which occurred a few months after that of the SFR System Arrangement. This clearly confirms the Forum's determination to assess the feasibility of all concepts and therefore, the Forum's aim to investigate all options.

However, it is clear that, whatever the concept, fruitful international cooperation can only be achieved provided the interest of each partner is well preserved. In relation with such requirements, I would like to stress another major accomplishment by the GIF, which took about three years, and which consisted in fixing the legal rules and bases that ensure that the R&D performed through international cooperation fully recognizes each participant's past and future contributions. Even though technical exchanges had already begun between GIF partners, especially on a bilateral level, multilateral cooperation was given a clear boost after negotiations led in 2007 to conclusions accepted by all, which duly recognize background property information, and deal satisfactorily with all property rights (intellectual, industrial, etc.). The GIF thus appears as the only existing structure enabling multinational cooperation within a sound legal basis that ensures that its R&D activities are carried out in an equitable manner between partners. This major step was followed by the signing of a series of Project Arrangements, starting from those on Advanced Fuel, Component Design and Balance of Plant and Global Actinide Cycle International Demonstration for the SFR, as well as that on Fuel and Fuel Cycle for VHTR. Others, such as those on Hydrogen Production and Materials for the VHTR and Thermal-Hydraulics for the SCWR, are about to be completed and signed.

As for the two other GIF Systems for which System Arrangements are yet to be signed, namely LFR and MSR, work is actively pursued, with interesting and promising findings, such as, in the case of the LFR, the need to exploit R&D synergies with other Gen IV systems, particularly the SFR in areas such as high burnup advanced fuel, design rules and codes, equipment, instrumentation, components and system. As for MSR, it appears that interfaces exist with other Gen IV systems on liquid salts as options for primary or intermediate coolant.

An issue common to all Gen IV Systems, which is actively being tackled within the framework of a specific Project Arrangement (the System, Integration and Assessment Project Arrangement), is related to the need for carrying out for each system conceptual design development, integration, and assessment functions. This Project Arrangement will provide or refine R&D requirements, integrate results from the R&D projects through their incorporation in conceptual system designs, and periodically assess the system for conformance to Generation IV Technology Goals and other requirements. Owing to the advanced stage reached by certain systems, such as the SFR, the need for such a Project Arrangement has become urgent.

All these technical achievements are a clear proof that, after a first phase mainly devoted to building the legal framework for fair and equitable cooperation, and even though the way to go is still long before achieving the final goal, the GIF members have started reaping the benefits of their joined efforts. Two other factors will contribute to speeding up this trend in the future. The first factor is the recent membership within the GIF of two prominent nuclear energy countries, China and Russia. Their extensive experience as well as their ambitious national programs will most certainly play an important role in shaping the GIF programs. The second factor is the increasing role of the industrial sector in the Forum's activities. Through recommendations from the GIF Senior Industry Advisory Panel as well as through the implications, in Project Arrangements, of industrial partners along with R&D centers and universities, the Gen IV system designers are constantly kept aware of investors' perspectives on issues such as licensing, economic and political risks.

Finally, and as a sign that the GIF's outcomes also benefit an audience far greater than its members, I would like to underline the interactions between the GIF and other international initiatives. Synergies exist and are presently being examined between the GIF and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), mainly in the fields of safety and non proliferation. In these fields, GIF and INPRO can be mutually complementary, the former being a group of providers, while the latter acts more as a group of users. As for the Global Nuclear Energy Partnership, the GIF will contribute to helping reach its goals by providing it with R&D in areas such as fuel services and infrastructure development.

I am therefore glad to see the way R&D is being performed within the GIF and the pace at which results are being achieved. The international community's favourable outlook on the achievements of GIF is a clear sign of its recognition of the Forum's value and input in the field of nuclear energy. I therefore seize this opportunity to thank the Forum's teams involved in the daily R&D tasks, and I congratulate them not only for their expertise, but also for their determination to pursue their efforts to ensure the sustainable development of nuclear energy.

Jacques BOUCHARD

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Chapter 1

The vast potential of nuclear energy as an attractive option from the viewpoints of security of supply and global climate change is increasingly recognized by energy policy makers, industry leaders and technical experts. Accordingly, nuclear energy is expected to play an essential role in meeting growing energy needs worldwide in a safe, environmentally clean and affordable manner. A resurgence of interest in nuclear energy for electricity generation and non-electrical applications is apparent as new nuclear plants are built or planned in a number of countries, and as an increasing number of countries, including developing and emerging countries, are evaluating its possible use in the future.

Nuclear power technology has evolved through roughly three generations of system designs: a first generation of prototypes and first-of-a-kind units implemented during the period 1950 to 1970; a second generation of industrial power plants built from 1970 to the turn of the century, most of which are still in operation today; and a third generation of evolutionary advanced reactors built by the turn of the 20th century, usually called Generation III/III+, which incorporate technical progress based on lessons learnt through more 12 000 reactor-years of operation.

The Generation IV International Forum (GIF) is a cooperative international endeavor to develop advanced nuclear energy systems responding better to the social, environmental and economic requirements of the 21st century. Generation IV systems under development by GIF promise to enhance the future contribution and benefits of nuclear energy. These systems employ advanced technologies and designs to improve the performance of reactors and fuel cycles as compared with current nuclear systems. Additionally, Generation IV systems target new applications of nuclear energy such as process heat supply, water desalination and hydrogen production.

This annual report is the first to be issued by GIF. It summarizes the GIF goals and accomplishments throughout 2007, describes its membership and organization, and provides an overview of its cooperation with other international endeavors for the development of nuclear energy. Future editions will focus on technical progress.

Chapter 2 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.

Chapter 3 describes the membership and organization of the GIF, the structure of its cooperative research and development (R&D) arrangements, and the status of Member participation in these arrangements.

Chapter 4 summarizes the R&D plans and achievements of the Forum until now. 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.

Chapter 5 reviews other major international collaborative projects in the field of nuclear energy and explains how the GIF interacts and cooperates with them.

Bibliographical references are provided in each chapter in order to facilitate access to public information about the GIF objectives, goals and outcomes. A public web site (http://www.gen-4.org/) provides a wealth of technical and scientific information on Generation IV systems and methodologies.

2.1 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 1, excepts from http://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 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.

2.2 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" (http://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 2.1 summarizes the main characteristics of the six Generation IV systems.

Table 2.1 - 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

These systems are described in more detail in Chapter 4; a brief summary of each system follows.

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 closed fuel cycle for fuel breeding and/or actinide management. The reactor may be arranged in a pool layout or a compact loop layout. Reactor size options 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 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.

3.1 GIF Membership

The founding document of Generation IV International Forum, the GIF Charter (http://www.gen-4.org/PDFs/GIFcharter.pdf), was signed in July 2001 by Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, the United Kingdom and the United States. Subsequently, Switzerland in 2002, Euratom[1] in 2003, and most recently the People's Republic of China and the Russian Federation, both in November 2006, signed the Charter, as shown in Table 3.1. Signing the Charter signifies interest in cooperation on Generation IV systems but does not commit the twelve signatories to take part in the cooperative development of these systems.

Among the signatories to the Charter, eight Members (Canada, Euratom, France, Japan, the People's Republic of China, the Republic of Korea, Switzerland and the United States) have signed or acceded to a Framework Agreement (see Table 3.1). Parties to the Framework Agreement (FA) formally agree to participate in the development of one or more Generation IV systems. Each Party to the Framework Agreement designates one or more Implementing Agents (see Table 3.1) to undertake the development of systems and the advancement of their underlying technologies.

Table 3.1 - Parties to GIF Framework Agreement and System Arrangements (SA)

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
P.R. of China	China Atomic Energy Authority Ministry of Science and Technology	X				
Rep. of Korea	Ministry of Science and Technology Korea Science and Engineering Foundation	X			X	X
Republic of South Africa						
Russian Federation						
Switzerland	Paul Scherrer Institute	X	X			X
United Kingdom						
United States	Department of Energy	X			X	X

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

It should be noted that Argentina, Brazil and the Republic of South Africa have signed the GIF Charter but not the Framework Agreement, and that the United Kingdom withdrew from the FA; within the GIF they are designated accordingly as "inactive Members." The Russian Federation is currently working on the necessary approvals for its accession to the FA. The Republic of South Africa, while currently an inactive Member, has indicated that it is working on the necessary Parliamentary approvals for its accession to the FA.

System Arrangements consistent with the provisions of the FA have been signed by Members interested in implementing cooperative R&D on the different selected systems. Signatories of each SA have formed a System Steering Committee (SSC) in order to plan and oversee the R&D required for each system. The participation of GIF Members in System Arrangements is shown in Table 3.1.

R&D for each GIF system is implemented through a set of Project Arrangements (PA). A PA typically addresses the R&D needs of the corresponding system in a broad technical area (e.g., fuels technology, advanced materials and components, energy conversion technology, plant safety, etc.). Participation in Project Arrangements is not restricted to the Parties to the FA. Public and private sector organizations, including those from non-GIF Members, may join any PA. It should be noted, however, that participation of organizations from non-GIF Members in a PA requires unanimous approval by the signatories to the PA and of the governing SA. The policy group may provide recommendations to the SSC on the participation in GIF R&D Projects by organizations from non-GIF Members.

3.2 GIF Organization

The GIF Charter provides a general framework for GIF activities and outlines its organizational structure. Figure 3.1 gives a schematic representation of the different GIF bodies and indicates the relationships among them.

As detailed in its Charter and subsequent GIF Policy Statements, the GIF is led by the Policy Group 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.

Every GIF Member may appoint up to two representatives in the Experts Group which reports to the Policy Group. The Experts 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 by each SSC.

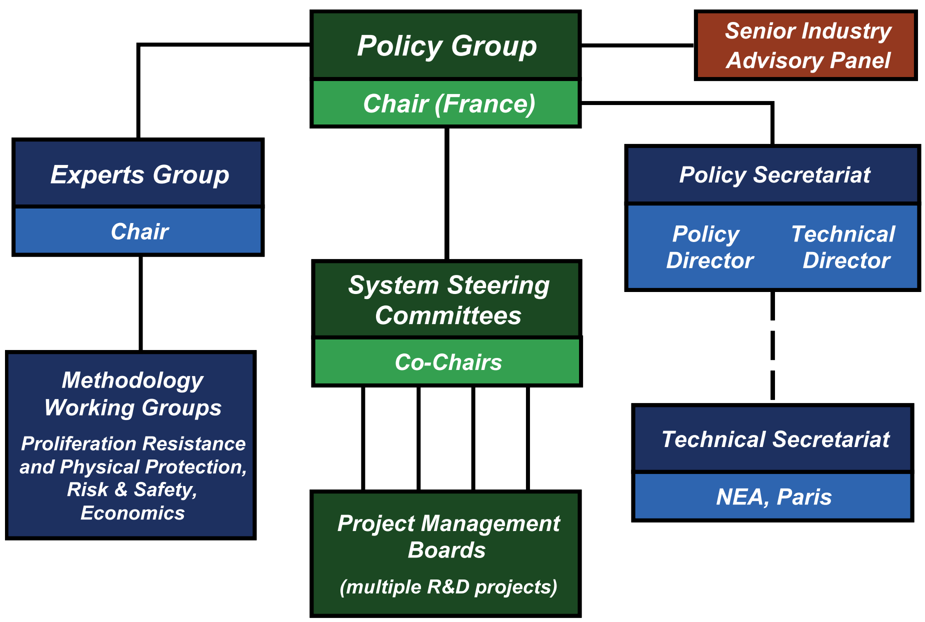
Left to right:
Ed Mc Ginnis, Massimo Salvatores, Jacques Bouchard, Sylvana Guindon, Jean-Louis Carbonnier, Yutaka Sagayama, Jean-Marc Cavedon, Young Sik Kim, Wei Huang, Pierre Frigola, Kazuaki Matsui, Ralph Bennett, Moon-Hee Chang, Simon Webster.

SSCs are established by the signatories to each SA to plan and oversee R&D activities for the corresponding Generation IV system. Within each system, Project Management Boards (PMBs) are established by the signatories to each PA in order to plan and oversee the corresponding project activities aiming at establishing the viability and performance of the corresponding Generation IV system. 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.

Three Methodology Working Groups (MWGs) are responsible for developing and implementing methods for the assessment of Generation IV systems in the fields of economics, proliferation resistance and physical protection, and risk and safety. 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 issues. The SIAP will contribute 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.

Figure 3.1 - GIF Governance Structure

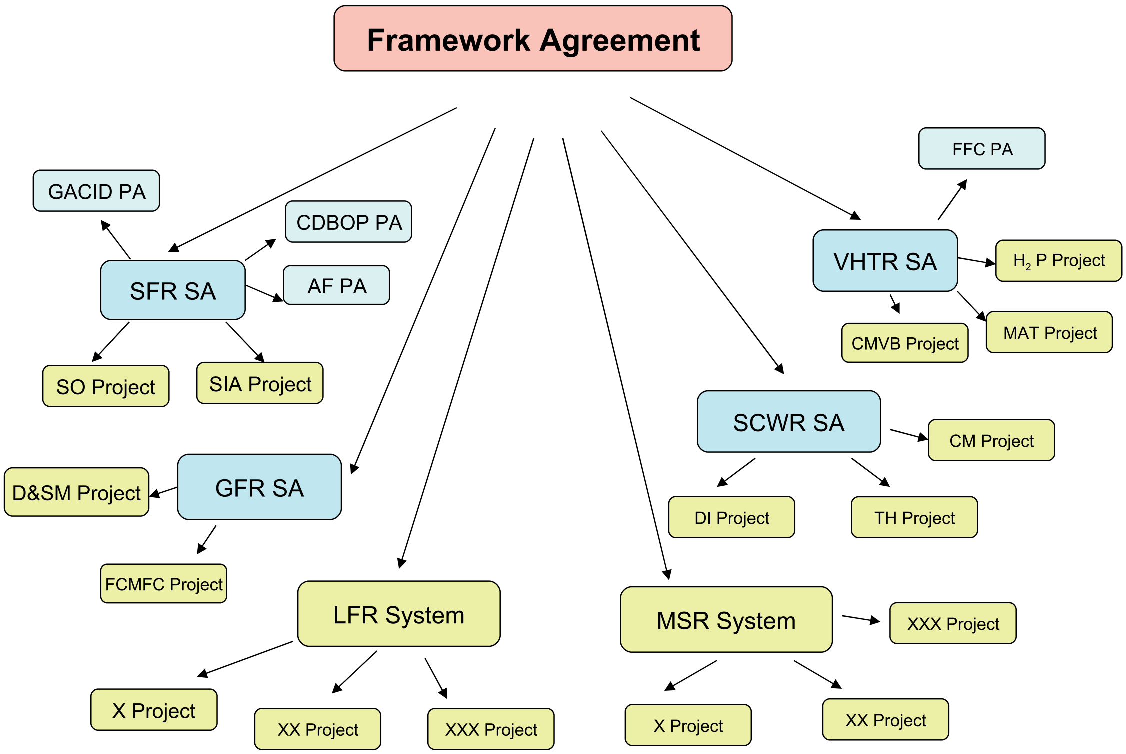


The GIF Secretariat is the day-to-day coordinator of GIF activities and communications. It includes two groups: the Policy Secretariat and the Technical Secretariat. The Policy Secretariat assists the Policy Group and Experts Group in the fulfillment of their responsibilities. Within the Policy Secretariat 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, provided by the OECD Nuclear Energy Agency (NEA), supports the SSCs, PMBs and MWGs. The NEA is entirely resourced for this purpose through voluntary financial or in kind (effort) contributions from GIF Members.

3.3 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. Figure 3.2 illustrates the structure of R&D projects within GIF.

Figure 3.2 - GIF Collaborative R&D Structure



While four arrangements have been signed by several Members for four systems (GFR, SCWR, SFR and VHTR), collaborative R&D is pursued on a provisional basis by interested Members for the other two systems (LFR and MSR).

As of mid-March 2008, 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, one PA had been signed: the Fuel and Fuel Cycle (FCF) PA. Many other projects are defined already and their membership agreed upon by interested parties on a provisional basis, although the corresponding project arrangements have not been signed yet. Table 3.2 gives an overview of signed arrangements and provisional R&D cooperation among GIF Members.

Table 3.2 Status of signed arrangements and informal cooperation within GIF

	CAN	EUR	FRA	JPN	PRC	ROK	RSA	RUF	CHE	USA
VHTR SA	X	X	X	X	O	X			X	X
VHTR HP Project	P	P	P	P		P	O			P
VHTR FFC Project	O	X	X	X		X				X
VHTR CMVB Project		P	P	P		P	O			P
VHTR MAT Project	P	P	P	P		P	O		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	O	P		P				
SCWR TH Project	P	P		P		P				
SCWR DI Project	P	P		P		O				
GFR SA		X	X	X					X	
GFR FCMFC Project		P	P	P					O	
GFR D&SM Project		P	P						P	
LFR System		P		P		O				P
MSR System		P	P							P

X = Signatory P = Provisional participant O = Observer

Acronyms of Projects

HP	Hydrogen Production
CMVB	Computational Methods Validation and Benchmarking
MAT	Materials
SO	Safety and Operation
SIA	System Integration and Assessment
CM	Materials and chemistry
TH	Thermal-hydraulics and safety
DI	Design and Integration
FCMFC	Fuel, Core Materials and Fuel Cycle
D&SM	Design and Safety Management

Beyond the formal and provisional R&D collaborations shown in Table 3.2, many institutes and laboratories cooperate with GIF Projects through exchange of information and results as indicated in Chapter 4 and in 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 Members. Other research bodies from non-GIF Members may participate in any project provided their participation is approved unanimously by both the corresponding PMB and the SSC concerned.

4.1 Systems

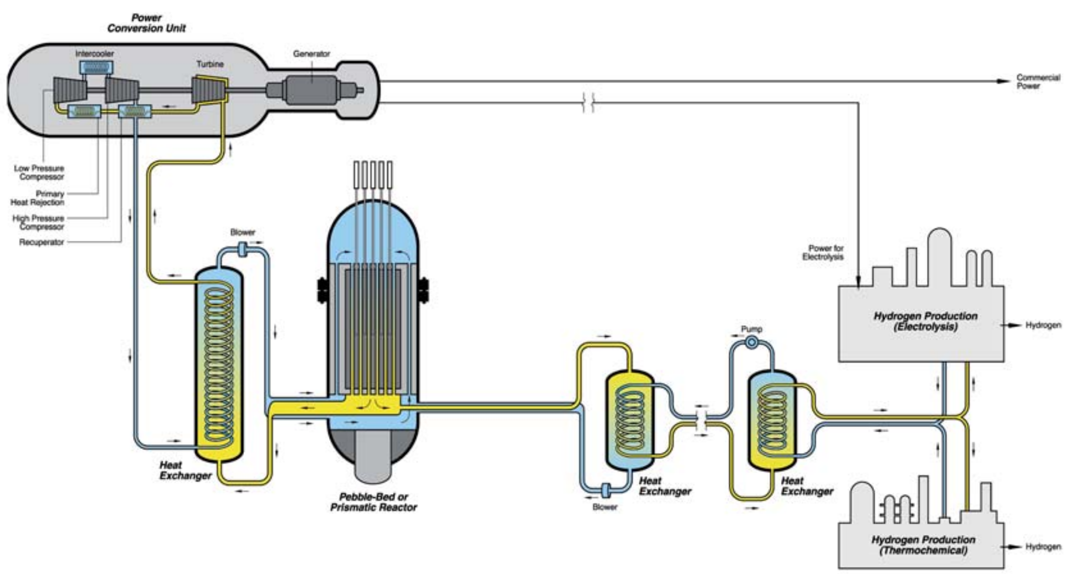
The following sections summarize very briefly, for each GIF system, the key characteristics of the systems under consideration, the R&D objectives and the status of collaborative research undertaken in the framework of GIF. The main results obtained in 2007 are provided as well as key relevant outcomes from research programs pursued by GIF Members outside of the GIF collaborative framework. More details 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.

4.1.1 Very-High-Temperature Reactor (VHTR)

4.1.1.1 Main characteristics of the system

The Very-High-Temperature Reactor (VHTR) is the next step in the evolution of high-temperature reactors (HTRs) and is primarily dedicated to the cogeneration of electricity and hydrogen from only heat and water by using thermo-chemical, electro-chemical or hybrid processes (see Figure 4.1). Its high outlet temperature makes it also attractive for chemical, oil and iron industry. It is a helium-cooled and graphite-moderated reactor with a core outlet temperature above 900°C (with an ultimate goal of 1 000°C) in order to support the efficient production of hydrogen by thermo-chemical processes. The VHTR has a potential for high fuel burn-up (150-200 GWd/tHM), complete passive safety, low operation and maintenance cost and modular construction.

Figure 4.1 - Schematic configuration of the VHTR and hydrogen-production systems



Two baseline concepts are being investigated for the VHTR core: the pebble-bed and the prismatic-block concepts. Initially, the fuel cycle will be a once-through very-high burn-up cycle using low enriched uranium but solutions will be developed to manage adequately the back end of the fuel cycle; potential operation with a closed fuel cycle will be assessed. The electric power conversion unit may operate in either a direct (helium gas turbine) or indirect (gas-mixture turbine) Brayton-type cycle. The standard UO_2 fuel TRISO coated particles concept (UO_2 kernel, SiC/PyC coating) may be enhanced by a UCO fuel kernel or by a ZrC coating.

The basic VHTR technology has been established in former high-temperature gas reactors such as the US Peach Bottom and Fort St-Vrain plants as well as the German AVR and THTR prototypes. The technology is being advanced through short- and medium-term projects led by several plant vendors and national laboratories, such as PBMR, GT-HTR300C, ANTARES, NHDD, GT-MHR and NGNP, respectively in the Republic of South Africa, Japan, France, the Republic of Korea, Russia and the United States. Experimental reactors such as HTTR (Japan, 30 MWth) and HTR-10 (China, 10 MWth) support the advanced concept development, and the cogeneration of electricity and hydrogen, and other nuclear heat applications.

4.1.1.2 Status of cooperation

The VHTR System Arrangement was signed in November 2006 by Canada, Euratom, France, Japan, the Republic of Korea, Switzerland and the United States. The Republic of South Africa is expected to join in 2008. The Fuel and Fuel Cycle Project Arrangement was signed early in 2008 by Euratom, France, Japan, the Republic of Korea and the United States. The Hydrogen Production Project Arrangement is expected to be signed later in 2008 by the same Members. The Materials (MAT) Project Arrangement, which addresses graphite, metals, ceramics and composites, has been finalized and is expected to be signed early in 2008. The Computational Methods Validation and Benchmarking (CMVB) Project Arrangement will be finalized and ready for signature by the end of 2008. Two other projects on components and high-performance turbo-machinery, and on design, safety and integration, are being discussed by the VHTR Steering Committee.

4.1.1.3 R&D objectives

The VHTR development approach builds on technologies already used for gas reactors that have successfully been built 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 plans will also support these short-term concepts through the re-establishment of knowledge base needed for manufacturing and regulatory licensing. VHTR R&D objectives have been addressed within six projects as noted above.

Design, safety and system integration – Integration work aims at updating the viability and performance assessment of the VHTR baseline concepts against the GIF performance goals and criteria, while integrating the results of the 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 systems integration, safety and economic analyses, and licensing.

Materials – 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; high temperature metallic materials for internals, piping, valves, high temperature heat exchangers, gas turbine sub components; ceramics and composites for control rod cladding and other specific reactor internals as well as for intermediate heat exchangers; and gas turbine components 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 development of design codes and standards as well as modeling to predict damage and lifetime assessment.

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

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 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 chemical and thermo-mechanical materials properties in representative conditions which will support a data base which will all be factored into advanced physical models to further allow the assessment of the fuel in-core behavior under normal and off-normal conditions. R&D will also examine spent fuel treatment and disposal, including used-graphite management, as well as the deep-burn of plutonium and minor actinides in support of a closed fuel cycle.

Hydrogen Production – The two main hydrogen production processes are the sulfur/iodine (S/I) thermo-chemical cycle and the high-temperature electrolysis (HTE) 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 examine mass and thermal balance and perform design-associated risk analysis for limiting, to the extent possible, the interactions between nuclear and non-nuclear systems, notably hydrogen explosion, tritium permeation and thermal disturbance caused by the hydrogen production system. Other thermo-chemical or hybrid cycles will be examined in terms of technical and economic feasibility in dedicated or cogeneration hydrogen production modes, with an eye to lowering operating temperature requirements in order to make them compatible with other Generation IV systems.

Computational Methods Validation and Benchmarking – 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 assessed through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported by HTTR tests or by past technology high-temperature reactor data (e.g. AVR, Fort St. Vrain, etc.). Normal and abnormal operating analyses will be performed, including criticality safety, coupled neutronics and thermal hydraulics, flow mixing in the hot plenum, air ingress, fission product transport and plate-out, pressurized or depressurized loss of coolant accident, and seismic behavior. Improved computational methods will also facilitate the elimination of unnecessary design conservatisms and improve construction cost estimates.

4.1.1.4 Milestones

The major milestones defined in the GIF technology roadmap and the VHTR System Research Plan are: end of viability stage (preliminary design and safety analysis) by 2010, end of performance stage (final design and safety analysis) by 2015 and end of demonstration phase (construction and preliminary testing) by 2020. Detailed milestones will be reexamined in 2008 in order to coordinate them with various national programs and near term projects.

4.1.1.5 Main activities and outcomes

In the Materials Project the work package on graphite has made significant progress in three tasks: graphite irradiation effects, graphite irradiation induced creep and graphite codes & standards development. The work package on metals has made progress through the development of codes and standards in cooperation with ASME.

Within the Hydrogen Project a number of technical reports will be completed in 2008 on static benchmark analysis and materials testing for the S/I process and system analysis of HTE. The initial assessments of alternative thermo-chemical processes are in progress and the initial studies of coupling the HP process to the reactor are underway.

The FFC PMB has organized its first common activity devoted to the experimental irradiation programme PYCASSO in the EU/JRC High-Flux Reactor (HFR) in Petten. This in-pile irradiation is aimed at gathering basic thermo-mechanical and thermo-physical properties of materials making up the fuel particles. Irradiations are scheduled to start in early 2008. Samples from Euratom (within the frame of the RAPHAEL Project of the 6th Euratom Framework Programme), CEA, JAEA, and KAERI have been fabricated and sent to the HFR. In the same field (materials properties), reports will be issued on specific phenomena (SiC under irradiation, buffer thermal diffusivity measurement, etc.). Fuel samples have been manufactured in conjunction with the U.S. Next Generation Nuclear Plant fuel program that began irradiation at the Idaho National Laboratory Advanced Test Reactor in December 2006. Finally, work is underway to establish a TRISO fuel materials database.

The VHTR system is benefiting from R&D conducted in a wide range of national programs and near term projects by commercial entities. While these near term projects may not fully accomplish all Generation IV goals, they will significantly increase the breadth of knowledge and provide important design and operational performance data which can be used to further guide future research and development activities needed to achieve Generation IV goals. The main features and status of these programs and projects are provided in the bibliography.

4.1.2 Sodium-cooled Fast Reactor (SFR)

4.1.2.1 Main characteristics of the system

The Sodium-cooled Fast Rreactor (SFR) 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. The reactor unit can be arranged in a pool layout or a compact loop layout.

The SFR system already benefits from considerable technological experience acquired in several GIF countries including China, France, Japan, Russia, the United kingdom Kingdom and the United States. Also, it offers the potential to operate with a high conversion fast spectrum core, with the resulting benefit of increasing the utilization of fuel resources. The envisaged SFR capability to efficiently and nearly completely consume trans-uranium as fuel would reduce the actinide loadings in the high-level radioactive waste it produces. Such reductions would bring benefits in the radioactive waste disposal requirements associated with the system and enhance its non-proliferation attributes. Reducing the capital cost and improving safety are the major challenges for the SFR system.

The reactor unit can be arranged in a pool layout or a compact loop layout. Typical design parameters of the SFR concept being developed in the framework of GIF are summarized below:

Outlet Temperature	500-550°C
Pressure	~1 Atmosphere
Power Rating	50-2000 MWe
Fuel	Oxide, metal alloy, others
Cladding	Ferritic-Martensitic, ODS, others
Average Burn-up	150 GWd/tHM
Breeding Ratio	0.5-1.3

Plant sizes ranging from small modular systems to large monolithic reactors are considered. Figures 4.2, 4.3 and 4.4 illustrate respectively loop-type, pool-type and modular-type systems.

- A medium or large size (600 to 1 500 MWe) loop-type sodium-cooled 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.*, 2005; Mizuno *et al.*, 2005).
- A medium or large size (600 to 1 500 MWe) pool-type sodium-cooled reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuele, supported by a fuel cycle based on pyrometallurgical processing facilities (Hahn, *et al.*, 2005).
- A small size (50 to 150 MWe) modular-type sodium-cooled 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).

Figure 4.2 Loop-configuration SFR

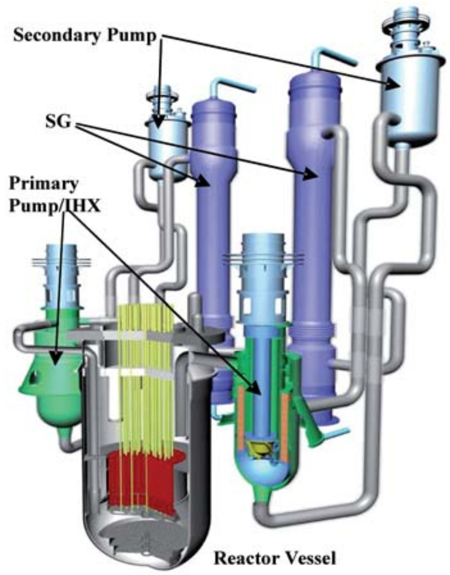


Figure 4.3 Pool-configuration SFR

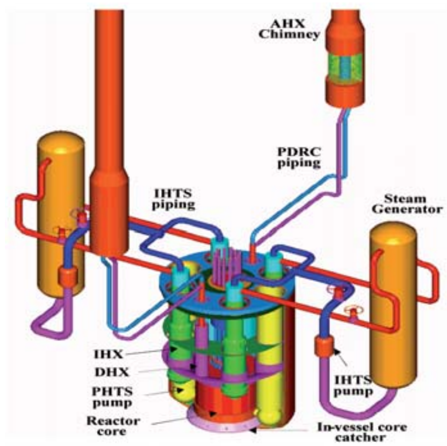
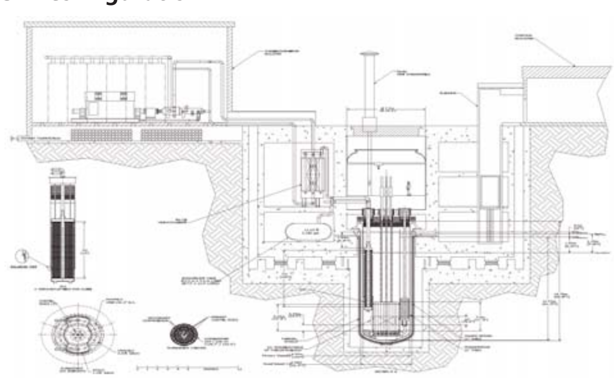


Figure 4.4 Small modular SFR configuration



The design and performance parameters of the three options are shown in Table 4.1.

Table 4.1 - Key Design Parameters of Generation IV SFR Concepts

SFR Design Parameters	Loop Configuration	Pool Configuration	Small Modular
Power Rating, MWe	1500	600	50
Thermal Power, MWth	3570	1525	125
Plant Efficiency, %	42	42	~38
Core outlet coolant temperature, °C	550	545	~510
Core inlet coolant temperature, °C	395	370	~355
Main steam temperature, °C	503	495	480
Main steam Pressure, MPa	16.7	16.5	20
Cycle length, years	1.5–2.2	1.5	30
Fuel reload batch, batches	4	4	1
Core Diameter, m	5.1	3.5	1.75
Core Height, m	1.0	0.8	1.0
Fuel Type	MOX (TRU bearing)	Metal (U-TRU-10%Zr Alloy)	Metal (U-TRU-10%Zr Alloy)
Cladding Material	ODS	HT9M	HT9
Pu enrichment (Pu/HM), %	13.8	24.9	15.0
Burn-up, GWd/t	150	79	~87
Breeding ratio	1.0–1.2	1.0	1.0

The SFR is well suited for the management of high-level waste types, in particular 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 level performance on thermal efficiency, safety and reliability. With innovations to reduce capital cost, the SFR can be competitive on electricity markets. The SFR fast spectrum also makes it possible to use available fissile and fertile materials (including depleted uranium) considerably more efficiently than in thermal spectrum reactors with once-through fuel cycles.

4.1.2.2 Status of cooperation

The System Arrangement for the international research and development of the Sodium-cooled Fast Reactor nuclear energy system was signed in February 2006 by Euratom, France, Japan, and the United States and acceded to later on by the Republic of Korea. Three Project Arrangements on Advanced Fuels, Global Actinide Cycle International Demonstration, and Component Design and Balance Of Plant have been signed in 2007. Two other Project Arrangements on Safety and Operation, and System Integration and Assessment are expected to be signed in 2008.

4.1.2.3 R&D objectives

The SFR development approach builds on technologies already used in several countries. As a benefit of 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, these research activities have been arranged by the SFR Signatories into five “Projects” to organize the joint GIF research activities:

- System Integration and Assessment (SIA) – The overall objective of this project is to review and integrate the outcomes of the other projects, evaluate their results and assess compliance of the designs under development with GIF goals.
- Safety and Operation (SO) – In order to contribute to the safety assessment of (preliminary) conceptual designs, this project consists of two projects, namely safety and operation. Experiments and analytical model development are planned in the safety project covering both passive safety and severe accident issues. Options of safety system architectures will also be investigated. The operation project aims at operation and technology testing campaigns in existing reactors, (e.g., Monju and Phenix) including the end-of-life test in Phenix.
- Advanced Fuels (AF) – This project 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 recycle fuels that contain minor actinides and possibly trace fission products; and the consideration of alternative fast reactor fuels .
- Component Design and Balance-Of-Plant (CDBOP) – This project covers the development of the balance of plant for the SFR system. It aims at meeting the GIF criteria in the field of safety, economy, sustainability, and proliferation resistance and physical protection. Experimental and analytical evaluation of advanced in-service inspection and repair technologies including leak-before-break assessment are being carried out. The project includes the development of alternative energy conversion systems with Brayton cycle.
- Global Actinide Cycle International Demonstration (GACID) – This project aims at conducting R&D with a view to demonstrate on a significant scale that the SFR can manage effectively all actinide elements in the fuel cycle, including uranium, plutonium, and minor actinides (neptunium, americium and curium). In that frame, the GACID project consists of minor actinide bearing fuel fabrication, licensing and pin-scale irradiations in the Monju reactor.

4.1.2.4 Milestones

The key dates defined in the five R&D projects of the SFR system are summarized below.

- SIA Project
  - 2006-2007 Design evaluation and systems integration
  - 2007-2010 Conceptual design
  - 2011-2015 Design optimization

- SO Project

R&D for Safety
  - 2006-2007 Safety review of design options
  - 2007-2010 Preliminary safety assessment of design options
  - 2011-2015 Safety assessment

R&D for Reactor Operation and Technology Testing

- ○ 2006-2010: Tasks related to design and safety
- ○ 2006-2010: Tasks related to component design and BOP

- AF Project
  - ○ 2006-2007: Preliminary evaluation of advanced fuels
  - ○ 2007-2010: Evaluation of minor-actinide-bearing fuels
  - ○ 2011-2015: High-burn-up fuel behavior evaluation
  - ○ 2016- Demonstration and application of advanced fuel head-end process in the fuel cycle backend

- CDBOP Project
  - ○ 2007 Viability study of proposal concepts
  - ○ 2007-2010 Performance tests for detail design specification
  - ○ 2011-2015 Demonstration of system performance

- GACID Project
  - ○ 2007-2012 Preparation for the limited minor-actinide-bearing fuel preparatory irradiation test
  - ○ 2007-2012 Preparation for the licensing of the pin-scale curium-bearing fuel irradiation test
  - ○ 2012- Program planning of the bundle-scale minor-actinide-bearing fuel irradiation demonstration

4.1.2.5 Main activities and outcomes

In 2007, the SSC completed a comprehensive System Research Plan (SRP) including development targets and design requirements, and R&D projects. In addition, several project plans were developed. The objectives of the SIA project were defined, according to the guidance of the Experts and Policy Groups, and its future activities are being defined and formulated within a project plan taking into account the specific characteristics of the SFR system development. After the restructuring of the design and safety project a revised project plan is being prepared for the SO Project.

Four options are being considered within the AF project for the SFR fuel: oxide, metal, nitride and carbide. Various fuel irradiation tests were ongoing in 2007 aiming at selecting advanced fuel options. Reactors available for those irradiation tests include Phenix in France, ATR in the United States and Joyo in Japan. Fuel evaluation studies and analytical work using fuel performance codes are in progress based on available information from previous tests including fuel property measurements and irradiation tests. The fuel evaluation covers minor-actinide-bearing fuel performance, minor-actinide-bearing fuel fabrication and high burn-up capability. The results will support the selection of advanced fuel options.
Within the CDBOP project, a program of sodium tests with external ultrasonic sensors is being defined in France for the study of *in situ* inspection and repair technologies. Results of a feasibility study for under-sodium visualization technologies will be reported by the Republic of Korea. A feasibility study of alternative energy conversion system concepts, thermodynamic cycle evaluation coupled with a SFR is being implemented in France. The United States are contributing results of compact heat exchanger test for super-critical CO_2 Brayton cycle, closed Brayton loop test and analysis. Japan is providing results of preliminary design study of plant system adopting supercritical CO_2 turbine system, thermal-hydraulic test, liquid sodium/CO_2 reaction test, and material corrosion test under supercritical CO_2 flow.

In 2007, the joint activities within the GACID project focused on the evaluation of minor-actinide-bearing fuel material property, and analysis and evaluation of irradiated-fuel data. In addition, preparation for minor-actinide-bearing fuel material property measurement (high Am content fuel and Cm-bearing fuel) was performed in France. Also, raw material preparation for material property measurement, and preparation for MA-bearing fuel material property measurement (supplemental data) was carried out in the United States. Finally, Japan contributed results from previous irradiation tests in Joyo (e.g. Am-1 test) and carried out preparation for minor-actinide-bearing fuel material property measurement (low Am content fuel).

4.1.3 Super-Critical Water Reactor (SCWR)

4.1.3.1 Main characteristics of the system

SCWRs are a class of high temperature, high pressure water-cooled reactors that operate above the critical pressure of water (374°C, 22.1 MPa). The GIF Technology Roadmap has identified several key technical advantages of the SCWR compared to conventional water technologies. These advantages translate into improved economics because of the higher thermodynamic efficiency and plant simplification opportunities afforded by a high-temperature, single-phase coolant.

Two design options will be considered for the SCWR: a) pressure vessel,[2] and b) pressure tube[3] designs. The R&D needs to assess technical feasibility (e.g., thermo-hydraulics, materials, chemistry, operating conditions) are common to both designs, which provides valuable collaboration opportunities for countries and organizations pursuing either option.

4.1.3.2 Status of cooperation

The SCWR System Arrangement was signed in November 2006 by Canada and Euratom and acceded to later by Japan. Much effort was spent in 2007 to finalize Project Arrangements (PAs) aiming at signing at least two of them in 2008. The countries interested in these projects are:

- Design and Integration (DI): Canada, Euratom, Japan
- Thermal-hydraulics and Safety: Canada, Euratom, Japan, Republic of Korea
- Materials and Chemistry: Canada, Euratom, France, Japan, Republic of Korea

Informal collaboration has been ongoing for a number of years between those countries. Summary results were shared between the participants during informal information exchange meetings.

4.1.3.3 R&D objectives and milestones

The main objective of the SCWR R&D is to establish the viability of the baseline SCWR while meeting future capital and market cost targets. The R&D plan is intended to address feasibility issues, such that, at the end of the plan, an SCWR technology demonstration could be constructed in the early 2020s.
The following critical-path R&D projects were identified to achieve the above objective:

- Design and integration: the main objective here is to define a reference design(s) that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance. An important collaborative component of this project is to design and construct an in-reactor fuel test loop to qualify the reference fuel design.
- 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 this project.

[2] The fuel and other components are contained in one large pressure vessel.

[3] The fuel and other components are contained in a large number of “small pressure vessels” called pressure tubes.

- Materials and chemistry: the main objective is to select key materials for use both in-core and out-core, for both the pressure tube and pressure vessel designs. Part of the work will require the definition of a reference water chemistry, based on materials compatibility and radiolysis behavior at supercritical conditions.

Other R&D areas, which would improve SCWR performance, but are not on the critical path to the development of the SCWR concept, include: hydrogen production, advanced fuel and fuel cycles, and the development of a fast-spectrum SCWR design. Work on these activities is currently at a moderate level but is expected to increase at a later stage when information has been obtained from the critical-path projects.

4.1.3.4 Main activities and outcomes

Design and Integration

Concepts for SCWR are being developed based on supercritical water fossil fired plant (SCW-FFP) experience, which have successfully been operated for more than 30 years; and reliable nuclear technologies based on BWR, PWR, and PHWR designs. Therefore, the development of SCWR systems is based on maximum utilization/adoption of such technologies with minor modifications.

Prior to the signature of the project arrangement an exploratory phase started with the conceptual design of different SCWR types based on the expertise existing within the member countries. During this phase Canada, Euratom, Japan and the Republic of Korea shared the work by developing the different SCWR options as described in Table 4.2.

Table 4.2 - Work sharing for the design exploratory phase

	Thermal-spectrum	Fast-spectrum
Pressure tube concept	Canada (CANDU-SCWR)	
Pressure vessel concept	Euratom (HPLWR*), Japan, Rep. of Korea	Japan

* HPLWR: High Performance Light Water Reactor

Canada developed an advanced pressure tube SCWR concept (CANDU-SCWR) operating at supercritical pressure. The economic evaluation of various design options led to an estimate capital cost reduction of ~40-50%. Canada considered also advanced fuel cycles with a preliminary evaluation of the use of thorium in a more sustainable fuel cycle. A parallel activity on hydrogen production focused on the evaluation and improvement of the Cu-Cl process which is the most suitable for the SCWR envisaged reactor outlet temperature (up to 650°C).

In September 2006, the High Performance Light Water Reactor (HPLWR)-Phase 2 project was launched under the 6th Euratom Framework Programme, bringing together ten partners from eight member States. The plant characteristics of the High Performance Light Water Reactor (HPLWR) include a supercritical coolant pressure of around 25 MPa and a coolant temperature up from 280°C to more than 500°C. The high steam enthalpy increases the power density of the steam cycle by more than 40%. The envisaged net efficiency shall reach 44%. An innovative concept for a thermal core, in which the coolant is heated up in 3 steps with intermediate mixing, was worked out in 2006. It minimizes hot spot temperatures of fuel claddings in the core to less than 630°C. In addition, preliminary studies were performed on a fast neutron spectrum core.

Japanese design activities in the past years had a wide scope. Japan developed fuel and core concepts for both thermal-spectrum and fast-spectrum SCWR to clarify the characteristics and assess the needs for technology development. Plant concept development (including control system and startup procedures) and turbine system preliminary design (estimating thermal efficiency and turbine building size) were also undertaken.

The Republic of Korea has worked on the conceptual design of a 1 400 MWe SCWR concept.

Overall, the main achievements in 2007 include:

- Evaluation of improved thermodynamic cycles adapted to the pressure tube concept using re-heat channels to further improve thermal efficiency (~50%).
- Improvements to a test facility used to evaluate the CANDU-SCWR fuel channel design.
- Experimental studies of direct hydrogen production from the Cu-Cl cycle.
- Studies related to the HPLWR mechanical design of fuel assemblies, of the reactor pressure vessel and of its internals (Figure 4.5 shows the latest design status).
- First step in validation of codes needed for the future detailed analysis of such a core design.
- Continuation of the work on the Japanese thermal-spectrum concept with improvement of the fuel concept to reduce its maximum cladding surface temperature, evaluation of the thermo-mechanical behavior of the fuel cladding and advancement of the safety system design and analysis.
- Further assessment of the SCWR-R.
- Fast-spectrum SCWR development including conceptual design of fuel and core, plant control system design and dynamics analysis (Figure 4.6 shows the cross sectional view of fast spectrum SCWR core and fuels).

Figure 4.5 - Design concept of the HPLWR

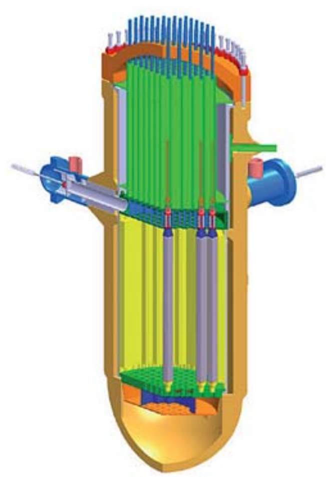
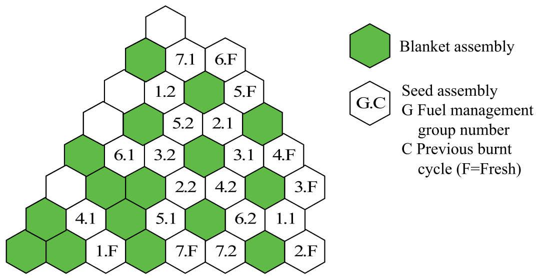
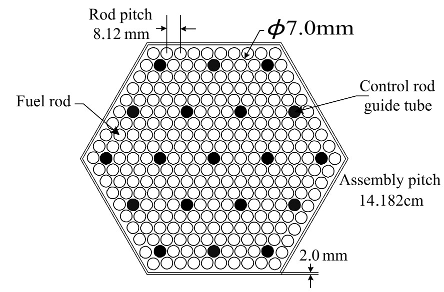


Figure 4.6 - Cross sectional view of fast-spectrum SCWR core and fuels

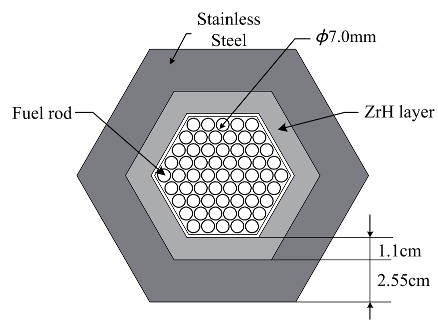


(a) Cross sectional view of core and loading pattern of fuels



(b) Seed assembly

(c) Blanket assembly



Thermal-hydraulics and Safety

The design criteria for SCWR are based on the cladding temperature limit for normal operation and trip analyses. Experimental data on heat transfer and pressure drop are crucial in establishing this limit accurately. The SCWR may be susceptible to dynamic instability due to the sharp variation in fluid properties (such as density) at the vicinity of the critical point. This instability may lead to high cladding temperature in the fuel prematurely impacting on the operating and safety margins. In support of the design and operation of the reactor safety (or relief) valve and the automatic depressurization system, the critical (or choked) flow characteristic must be established at supercritical conditions since current information has been obtained at sub-critical conditions. This established characteristic is also required in the analysis of a postulated large-break loss-of-coolant accident event. Research needs in thermal-hydraulic and safety includes thus heat transfer, hydraulic characteristics, critical flow, stability, simulation of system performance and behavior during transient and accident and therefore identification of safety requirements and evaluation, development of system code and relevant methodologies and sub-channel analysis.

Both analytical and experimental activities are performed to fulfill these needs. Ongoing activities include:

- Thermal-hydraulic experiments at supercritical conditions with water and surrogate fluids such as Freon and CO_2 to obtain heat transfer and pressure drop data. Figure 4.7 shows the Korean SPHINX test loop which uses CO_2 as surrogate fluid for water.
- Computational fluid dynamics to study the thermal-hydraulics behavior of different types of fuel channels.
- Modification and validation of existing analysis tools for the application to SCWR conditions.
- Development of passive safety systems.

- Investigation of safety criteria and safety systems and estimation of safety margin.
- Stability analysis for both thermal-hydraulic stability and coupled neutronic thermal-hydraulic stability.

Figure 4.7 - SPHINX Test Loop

Material and chemistry

Due to the higher reactor outlet temperature, new alloys are necessary for the fuel cladding or in-core components of SCWR. The R&D plan for the SCWR cladding and structural candidate materials focuses on acquiring data and a mechanistic understanding relating to the following key properties of candidate materials: corrosion and stress corrosion cracking (SCC); strength, embrittlement and creep (thermal and irradiation-induced) resistance; and dimensional and micro-structural stability. The standard approach to performing the R&D in each of the key areas is to perform scoping tests out of pile on un-irradiated alloys, companion tests out-of-pile on irradiated alloys, and in-pile tests in an SCW loop (or loops) to be constructed. Past and present activities are mainly centered on the first stage of the experimental process which consists in gathering existing information on un-irradiated material and in performing additional tests (either in autoclaves or in SCW tests loop built or under construction by most of the participants to the project). Figure 4.8 shows a Corrosion and SCC Test Loop built in the Republic of Korea. Canada additionally works on the development of corrosion resistant coatings using a low pressure/temperature plasma jet.

Figure 4.8 - Corrosion and SCC Test Loop

The long-term viability of a SCWR will also depend on the ability of designers to predict and control water chemistry to minimize corrosion rates and SCC, as well as to minimize deposition on in-core (e.g., fuel cladding) and out-core (piping, turbine blades) surfaces. Meeting this goal requires an enhanced understanding of the chemistry of supercritical water. The marked change in the density of SCW through the critical point is accompanied by dramatic changes in chemical properties. These complications are further exacerbated by in-core radiolysis, which preliminary studies suggest is markedly different than what would have been predicted from simplistic extrapolations of the behavior encountered in conventional water-cooled reactors. Both experimental and analytical work are currently performed to better understand radiolysis at supercritical condition and to measure the impact of the water chemistry on the corrosion process.

4.1.4 Gas-cooled Fast Reactor (GFR)

4.1.4.1 Main characteristics of the system

The Gas-cooled Fast Reactor (GFR) system features a high temperature helium cooled fast spectrum reactor with a closed fuel cycle. It associates the advantages of fast spectrum systems (long term resources sustainability, in terms of use of uranium and waste minimization, through fuel multiple reprocessing, grouped recycling and transmutation of long-lived actinides) with those of the high temperature (high thermal cycle efficiency and the possibility of hydrogen production). Its high-temperature gas technology is to rely, as much as possible, on technologies already used for the High-Temperature Reactor (HTR). For its fuel and safety features, the GFR uses significant innovations, if not breakthroughs, for example on advanced materials or helium cooling systems.

4.1.4.2 Status of cooperation

The System Arrangement for the international research and development of the Gas-cooled Fast Reactor nuclear energy system was signed in November 2006 by Euratom, France, Japan and Switzerland. Two projects have been identified (see below) and project plans, covering technical content and participants' technical contributions for each project, are under preparation. The GFR System Steering Committee also started the preparation of an international GFR Technical Seminar to be held in 2008 with the objective to give an overview of the work performed within GIF to potential additional contributors.

4.1.4.3 R&D objectives

The unique combination of the GFR characteristics and design objectives leads to the need of resolving two major challenges: the development of an innovative fuel and its associated fuel cycle technologies and the design and safety analysis of the GFR system. These major goals are translated in the definition of two projects: the fast neutron Fuel, other Core Materials, and specific Fuel Cycle process (FCMFC) project; and the Design and Safety Management (D&SM) project.

The main objectives of the FCMFC project are:

- The development of an innovative fuel with a high fissile-atom density, able to sustain high levels of operating temperatures, fast flux, high burn-up, and compatible with a close confinement strategy for fission products.
- The development of the associated fuel cycle technologies with the aim to afford the integral recycling of actinides.

The D&SM project aims at developing a coherent system – fuel, reactor, cycle options – with a self generating core, a robust safety approach and an attractive level of power density. Neither experimental reactors nor prototypes of the GFR system have been built so far and, therefore, the construction and operation of a first experimental reactor – the Experimental Technology Demonstration Reactor (ETDR) – is needed in the performance phase to qualify key technologies (Mitchell, C., et al.et al., 2006). In this context, the main goals of the D&SM Project are:

- The definition of a GFR reference conceptual design and operating parameters meeting the following requirements:
  ◦ self-breeding cores practically with optional need for fertile blankets;
  ◦ capability for global multi-recycling of plutonium and minor actinides;
  ◦ selection of an adequate level for power density to meet requirements in terms of economics, ease of deployment of the reactor fleet, and management of safety issues; and
  ◦ coupling approach between the reactor and process heat applications.
- The identification and assessment of alternative design features regarding the GIF goals and criteria (lower temperatures, indirect cycle).
- The safety analysis for the reference GFR system and its alternatives.
- The ETDR conceptual design and safety analysis.
- The assessment of economic performance.
- The development and validation of computational tools needed for the design and the analysis of operating transients (design basis accidents and beyond).

4.1.4.4 Milestones

The key milestones identified in the GFR System Research Plan (SRP) are:

◦ 2006-2007 GFR pre-conceptual design on the above selected options. At the end of this phase, release of the GFR preliminary viability report.

◦ 2008 Establishment of the ETDR as a separate R&D project.

◦ 2008-2012 GFR conceptual design phase ending with the prototype system and fuel options selection (one or two open options may still require some further R&D).

- ◦ 2012 ETDR decision of construction.
- ◦ 2019 End of the GFR preliminary design studies. Final GFR options selection.
- ◦ 2020 Start of ETDR operation with GFR experimental fuel assemblies.

4.1.4.5 Main activities and outcomes

Prior to the signature of a the GFR System Arrangement, Euratom, France, Japan, Switzerland, the United Kingdom and the United States jointly explored the GFR option in the framework of GIF activities. They formulated an International Collaboration Plan (ICP) comprised of bilateral work scopes. A GIF Gen IV GFR compendium entitled "End-of-Exploratory Phase Design and Safety Studies" was prepared which summarized the design and safety analyses work on the GFR performed over the years 2004 to 2006 by these international partners under this ICP. In accordance with the draft GFR SRP, this work covered the exploratory phase where seven design options and concepts were investigated to provide the basis for down selection. The conclusions from this work are summarized below.

GFR options selection

- A high GFR unit power looks preferable as the decreased neutron leakage makes it easier to design self-sustainable cores with less challenging fuels. A 2 400 MWth unit power is chosen for the pre-conceptual design phase.
- An attractive 100 MW/m^3 power density is selected in order to achieve a reasonable in-core plutonium inventory and also as an acceptable compromise between economics and safety considerations.
- Cores should be designed to achieve a low pressure drop value in order to facilitate gas circulation in any situation.
- A medium containment pressure safety strategy is chosen for the decay-heat removal; this means that the design has to include a guard vessel capable of maintaining a pressure of 0.6 to 1.0 MPa in case of primary circuit failure. The rationale for this is to establish compatibility with low pumping power for the emergency gas circulation and to offer the possibility to rely on natural circulation only after several hours after shutdown.

Fuel selection

Table 4.3 gives the results of four of the seven studied combinations of options. All the options considered need the development of adequate SiC-based structural and matrix materials.

There are two fuel strategies, both based on refractory materials:

- Fission product confinement close to the place of birth. Table 4.3 shows the advantages of the plate core over the particle one, indicating that in this fuel family the plate concept has to be ranked first.
- Classical strategy of pellet fuel with fission product gases retained in a plenum. Two pin concepts have been characterized, one with carbide fuel and the other one with oxide fuel. The carbide core was recommended for further studies.

Table 4.3 - Exploratory phase design results for the four combinations of options used for fuel selection

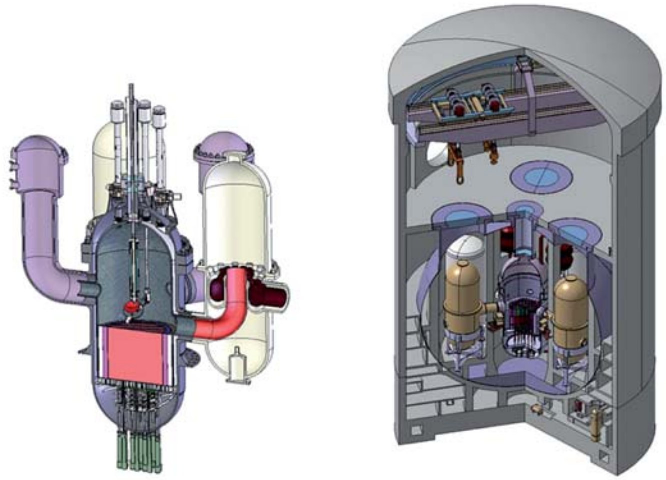
Option	Case 3 Carbide plate, Direct (indirect) cycle	Case 4 Carbide pin, Direct cycle	Case 5 Particle fuel, Vertical flow A, Direct cycle	Case 6 Oxide pellets in ceramic pins, Direct cycle
Unit power (MWth/MWe)	2400/1157/(1087)	2400/1128	2400/1124	2400/?
Power density (MW/m³)	100	100	90	76.5
Specific power (W/gHM)		42	36	29
Core outlet temperature (°C)	850	850	850	850
Core inlet temperature (°C)	480 (400)	480	460	480
Mass flow rate (kg/s)/ He speed (m/s)	~1300/61	1249/58	1184/100	~1250/46 (mean)
Core volume (m³)	24	24	25	31.4 m³ (fuel SAs)
Core height/diameter (m)	1.55/4.44	1.34/4.77	0.9/5.9	2/4.47
Subassembly height (m)	~5	5.43	~3.6	6.7
Core pressure drop (bar)	0.66 (0.44)	0.54	2	0.9 (uncertainty>30%)
Fuel	carbide in SiC plates	carbide in SiC pins	(U, TRU) N enriched in ^{15}N	oxide in SiC pins
Structures	SiC	SiC	TiN, SiC	SiC
Core structures/coolant/fuel volumes (%)	17.6+20/40/22.4	23/55/22	55/25/20	23.9/47.9/28.2
Fuel/Clad Max. temperatures (°C)	1250/950	1200/985	1100 (cooling tubes)	2000 (tentative and pessimistic)/1100
Reflector material	Zr_3Si_2	Zr_3Si_2	$SiC+B_4C$	Zr_3Si_2
In core Pu inventory (t/GWe)	7.7 (10.1, incl. 0.6 MAs)	8.68	7 Pu fissile (12 Pu total)	11
Mean fuel Pu content (%)	15.2 (18.5 with MAs)	17.3 with MAs	23 (total HM)	15.5 PuO_2 volume BOL
Breeding gain (BOL/EOL)	-5% without MA recycling	0	1.03 (without RB), 1.11 (with RB)	-4.8% without MA recycling
Core management (X x EFPD)	3x831	3x786	6x1550	6x512.6
Burn-up (% FIMA)	10.1(mean)/14.7(max)	9.9	13 (core average)	9.5 average discharge
ß BOL/EOL (pcm)	388/344	346/346	296	396.3/397 BOC1/EOC1
Doppler BOL/EOL	1872/1175 (pcm)	0.3/0.28 (¢/°K)	-0.011 (T dk/dT')	1620/1549 BOC1/EOC1 (pcm)
He void BOL/EOL (pcm)	0.7 $	1.1 $	0.3 $	0.9 $

Abbreviations and acronyms

HM	Heavy Metal
SA	(fuel) Sub Assembly
BOL	Beginning Of Life
EOL	End Of Life
RB	Radial Blanket
EFPD	Equivalent Full Power Day
FIMA	Number of fissions that have occurred per 100 heavy metal atoms
BOC1	Beginning Of Cycle 1
EOC1	End Of Cycle 1

Energy Conversion System

Figure 4.9 - Preliminary schematic diagram of the GFR concept.



Very advanced options such as the direct He-Brayton cycle and the super critical CO_2 cycle, both associated with capabilities of high efficiency and compactness, are considered together with more proven technologies such as the indirect combined cycle using a He/N_2 mixture for the secondary circuit and a Rankine one for the tertiary. This later proven technology is compatible with high temperature and super critical water energy conversion that offers a good efficiency prospect and needs to develop a high temperature heat exchanger which is studied in the framework of the VHTR system. Figure 4.9 illustrates a preliminary diagram of the GFR concept.

Proposed strategy for the pre-conceptual design phase

Figure 4.10 - GFR design options for the pre-conceptual design phase

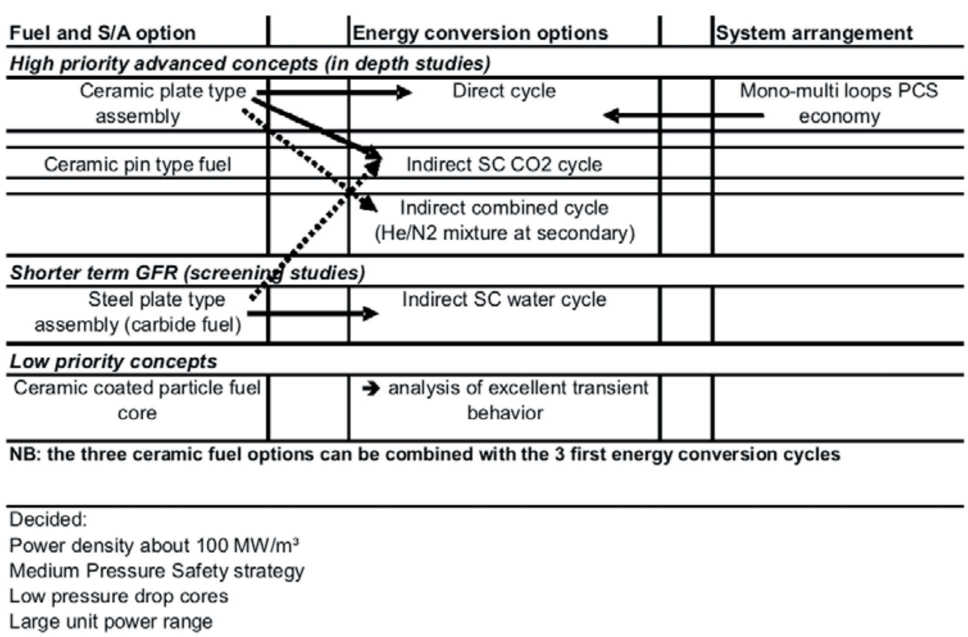


Figure 4.10 gives the proposed options studied in the pre-conceptual GFR design phase. The overall logic was to go ahead with in depth studies on a 2 400 MWth concept aiming at full GFR goals, including high efficiency, self-sustainable core, and design compatible with long term decay-heat removal by natural circulation, with reference options (ceramic plate type fuel, direct cycle) and back-up options (ceramic pin type fuel, indirect SC CO_2 and He/N_2 combined cycle).

As a potential step to the GFR, it is proposed to evaluate, by screening studies, the feasibility and performance of a shorter-term GFR compatible with available structural materials (metallic) and energy conversion technology.

A preliminary viability study completed at the end of 2007, compiles the findings from the phase of informal collaboration pursued until the signature of the System Arrangement. The study shows that research studies conducted on the Gas-cooled Fast Reactor since 2001 has produced a consistent design of the reactor and its fuel and describes all the components of the system.

This consistent design reaches the following initial set of performance:

- self-generation of plutonium in the core to ensure uranium resource saving;
- optional fertile blankets to reduce the proliferation risk;
- limited mass of plutonium in the core to facilitate the deployment of a fleet;
- ability to transmute long-lived nuclear waste resulting from the recycling of spent fuel, without lowering the overall performance of the system; and
- favorable economics owing to a high power conversion ratio.

In parallel, the safety architecture that is used correctly covers the potential defects fitted to this system, thanks to the following elements:

- a fuel element that uses refractory materials and withstands very high temperatures;
- a gas voiding reactivity effect naturally not significant in the core; and
- a capacity of the reactor to be cooled down even in case of a large leak, thanks to different systems of moderate power supply and to a closed containment.

Contrary to the designs studied in the past, the approach based on these elements enables the decay heat to be removed when pressure is lost. Nevertheless, additional research is needed on design, fabrication and behavior under irradiation of the refractory fuel element, improved technology and architecture of large components, management of anticipated severe accidents that affect the core, instrumentation and control. Finally, some basic options need to be reconsidered taking advantage of newly developed calculation tools, the consistent design and the historical knowledge. This will lead to an updated concept with improved performance by 2012, marking the end of the feasibility phase.

4.1.5 Lead-cooled Fast Reactor (LFR)

4.1.5.1 Main characteristics of the system

The *GIF Technology Roadmap* identified the LFR as a technology with potential to meet the electricity needs of remote sites as well as for large grid-connected power stations. It features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also, like other fast spectrum reactors, 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 relative inert coolant.

The LFR was primarily envisioned for missions in electricity and hydrogen production, and actinide management. Given its R&D needs for fuel, materials, and corrosion control, the LFR system was estimated to afford a technology demonstration by 2025. The preliminary evaluation of the LFR concepts considered by the LFR Provisional System Steering Committee (PSSC) covers their performance in the areas of sustainability, economics, safety and reliability and proliferation resistance and physical protection.

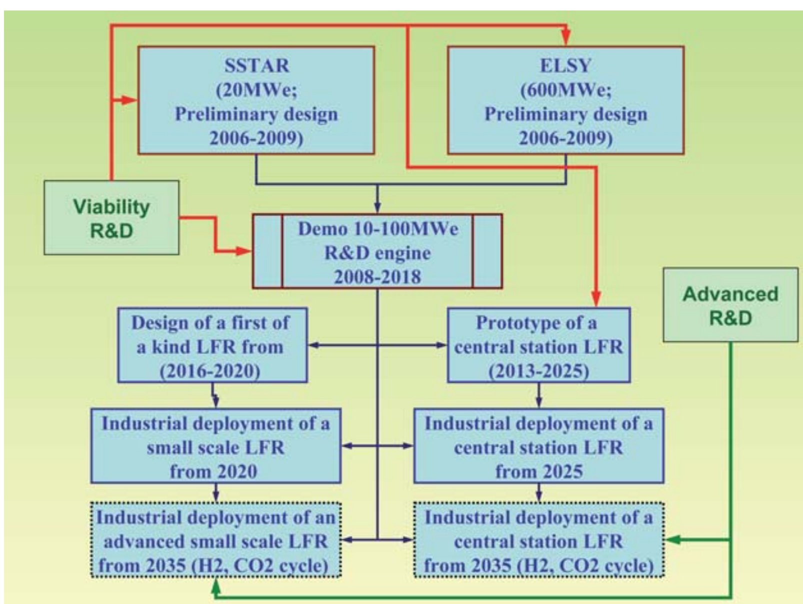
4.1.5.2 Status of cooperation

The cooperation on LFR within GIF was initiated in October 2004 and the first formal meeting of the PSSC was held in March 2005 with participation of representatives from Euratom, Japan, the Republic of Korea and the United States. The PSSC held periodical meetings, roughly twice a year, since its creation to prepare a draft System Research Plan (SRP) which was submitted to the EG for review by mid-2007. PSSC meetings were augmented by informal meetings with representatives of the nuclear industry, research organizations and universities involved in LFR development.

4.1.5.3 R&D objectives and milestones

The draft SRP for the Lead-Cooled Fast Reactor is based on molten lead as the reference coolant and lead-bismuth as backup option. Figure 4.11 illustrates the basic approach recommended in the draft SRP. It portrays the dual track viability research program with convergence to a single, combined demonstration facility (Demo) leading to eventual deployment of both types of systems.

Figure 4.11 - LFR SRP conceptual framework



The approach adopted aims at addressing to some extent the research priorities of each participant 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.

Provided the successful operation of the demo around the year 2018, a prototype development effort is expected for the central station LFR leading to industrial deployment at the horizon of 2025. In the case of the Small Secure Transportable Autonomous Reactor (SSTAR) option the development of a first of a kind unit in the period 2016-2020 is foreseen. Because of the small size of the SSTAR it is expected that the main features can be established during the demo phase, and that it will be possible to move directly to industrial deployment without going to 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 carried out in parallel to the construction of the demo and planned in such a way as to start construction as soon as beginning of the demo operation at full power has given the main assurances of the viability of this new technology.

4.1.5.4 Main activities and outcomes

The main ongoing activities in GIF members are:

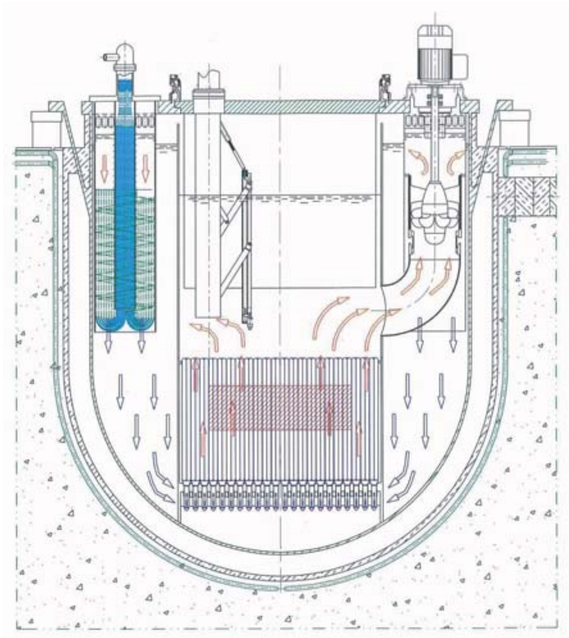
- the development of ELSY (European Lead-cooled System) in Europe;
- the development of the SSTAR in the United States; and
- the development of advanced materials for Lead Bismuth Eutectic (LBE) applications and thermo-hydraulic studies on LBE systems in Japan.

The two first ongoing research initiatives are currently introduced as potential for international cooperation and joint development in the GIF framework.

The ELSY reference design is a 600 MWe pool-type reactor cooled by pure lead. This concept has been under development since September 2006, and is sponsored by the Sixth Framework Programme of Euratom (Figure 4.12). 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 Generation IV goal of minor actinide burning capability.

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 electric energy 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, the reduced elevation the result of the design approach of reduced-height components.

Figure 4.12 - Preliminary scheme of the ELSY reactor



In 2007, activities concentrating on large-scale LFR development have focused 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 coolers, and refueling system.

The reactor assembly configuration has started with main focus on the interface of the reactor vessel, the anchored safety vessel and the reactor building with the reactor vessel air cooling system, the reactor concrete cooling system and the reactor pit cooling system. The reactor vessel air cooling system mainly consists of an innovative decay heat removal system with a number of air U-tubes always in operation to dissipate to external air the heat radiating from the outer wall of the reactor vessel. The reactor concrete cooling system consists of U tubes grouped in two loops normally operating in water forced circulation to cool the reactor pit concrete during all operating conditions. The reactor pit cooling system consists of open ended tubes located in the reactor pit exchanging hot air from the bottom with an equivalent inflow of cold air from the reactor building for protection of the operators and of the equipment during *in situ* inspection activities.

A convenient solution has been identified to combine the three cooling systems and to provide a large access to the reactor pit for *in situ* inspection needs. To improve the mechanics of the reactor vessel support, a solution has been identified which reduces the radial distance between the reactor vessel and the reactor pit. This result is possible thanks to a safety vessel anchored to the reactor pit as for the European Fast Reactor and to the innovative compact reactor vessel air cooling system configuration which allows limiting the distance between reactor vessel and anchored safety vessel with the additional advantage of reducing the losses of lead inside the reactor pit in case of reactor vessel leakage.

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. A new proposed steam generator is characterized by a spiral-wound tube bundle arranged in the bottom-closed, annular space formed by a vertical outer and an inner shell. 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, provides the head required to force the coolant to flow through the perforated inner shell in a radial direction and, past the tube spirals, through the outer shell. 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. New solutions are also under evaluation for the fuel element support and for the refueling system.

Two different core types are being preliminarily considered: the core made up of hexagonal wrapped assemblies as used in sodium-cooled reactors; and wrapperless square assemblies, as commonly used in PWRs. ELSY features a conversion factor of one and burns its minor actinides. The transmutation of a larger amount of minor actinides is being considered to address the issue of minor actinide legacy. A key-point of the core design approach is the small temperature difference between the coolant mean outlet temperature (480°C) and the limiting cladding temperature that requires a rather flat radial power distribution. This implies a core sparsely fitted with control rods, but offering the required worth.

Research activities in Japan concentrated on:

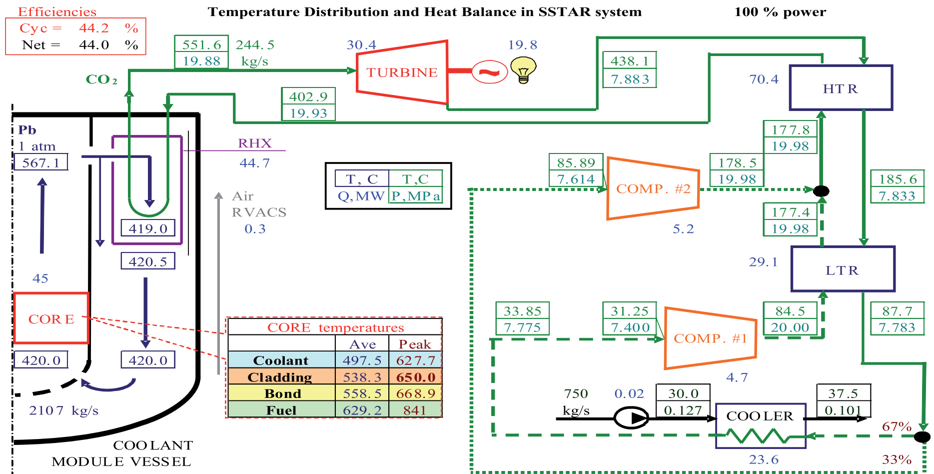
- the heat transfer performance of LBE in the intermediate loop;
- two phase flow characteristics of LBE in water and steam;
- the gas and steam lift performance of LBE;
- LBE material compatibility;
- LBE-water direct contact boiling mechanism;
- the corrosion characteristics and corrosion behavior of the reactor coolant, the structural and cladding materials;
- oxygen control with steam injection into LBE;
- Polonium behavior in the coolant system.

The CRIEPI Static Corrosion Test Facility investigates the corrosion behavior of stagnant LBE at 650oC on high chromium martensitic stainless steel, a promising candidate of structural material for LFRs. A series of corrosion tests were performed jointly by CRIEPI and JAEA.

The Tokyo Institute of Technology proposed research activities on the development of Steam Lift Pump Type LBE Cooled Reactors (SLPLFR), the Pb-Bi Cooled Direct Contact Boiling Water Fast Reactor (PBWFR) with electric power of 150 MW and on the CANDLE reactor that does not require movable reactivity control mechanisms. The LFR program is supported by the Center of Excellence – Innovative Nuclear Energy Systems (COE-INES).

The current reference design for the SSTAR in the United States is a 20 MWe natural circulation pool-type reactor concept with a small shippable reactor vessel (Figure 4.13). 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, as well as a high degree of passive safety. Conversion of the core thermal power into electricity at a high plant efficiency of 44% is accomplished utilizing a supercritical CO_2 Brayton cycle power converter.

Figure 4.13 - SSTAR preliminary design concept and operating parameters with S-CO2 Brayton-cycle energy converter



Two systems are developed in the Republic of Korea, the proliferation-resistant, environment -friendly, accident-tolerant, continual and economical reactor (PEACER, Hwang, I.S., *et al.*, 2006) and the BORIS (Kim, W.J., *et al.*, 2006). In the Russian Federation, two systems are considered the SVBR-75/100, a lead-bismuth eutectic-cooled modular fast reactor having a power range of 75 to 100 MWe (Zrodnikov, A.V., *et al.*, 2006) and BREST lead-cooled fast reactor concept and the associated fuel cycle (Adamov, E.O., *et al.*, 2001).

4.1.6 Molten Salt Reactor (MSR)

4.1.6.1 Main characteristics of the system

In a Molten Salt Reactor (MSR), the fuel is dissolved in a fluoride salt coolant. The technology was partly developed in the 1950s and 1960s. With changing goals for advanced reactors and new technologies, there is currently a renewed interest in MSRs. The new technologies include (1) Brayton power cycles (rather than steam cycles) that eliminate many of the historical challenges in building MSRs and (2) 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.

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 single isotopic composition of fuel in the reactor. These and other characteristics may enable MSRs to have potentially unique capabilities and competitive economics for actinide burning and extending fuel resources.

Table 4.4 summarizes some essential characteristics and performance of different MSR concepts (Forsberg, 2007; Mathieu, 2006; Merle-Lucotte, *et al.*, 2007; Hron, *et al.*, 2007; Ignatiev, *et al.*, 2007a; Ignatiev, *et al.*, 2007b; Ignatiev, *et al.*, 2006).

Table 4.4 - Characteristics and performance of different MSR concepts

Family	Concepts	F/T	Fuel cycle	Thermal Power (MW)	Comments
MSR-Breeder	MSBR	T	^{233}U/Th	2 250	Reference breeder concept BR > 1.05 Feedback reactivity coefficient > 0 (slightly)
	AMSTER-B	T	^{233}U/Th	2 250	BR > 0.95
	REBUS	F	U/Pu	3 700	
	FUJI	T	^{233}U/Th	150-200 electrical	
	TMSR	T or F	^{233}U/Th	2 500	BR > 1 Feedback reactivity coefficient < 0 (T and F)
MSR-Burner	AMSTER-I	T	U-Pu-MA	2 250	
	SPHINX	F	Pu-MA	1 208	
	MOSART	F	Pu-MA	2 400	Reactivity coefficient < 0

F = fast; T = thermal

Apart from MSR systems, other advanced concepts are being studied, which use the liquid salt technology.

The Advanced High-Temperature Reactor (AHTR) uses clean liquid salts as a coolant and the same graphite core structures with coated fuel particles as gas-cooled reactors such as the VHTR (Forsberg, 2007a). The better heat transport characteristics of salts compared to helium enable power levels up to 4 000 MWth with passive safety systems. The fuel cycle characteristics are essentially identical to those of the VHTR. The AHTR is a longer term high-temperature reactor option with potentially superior economics because of the coolant salt properties. It can be built in larger sizes, it operates at lower pressure and the equipment is smaller because of the superior heat transfer capabilities of liquid salt coolants compared to helium.

For Sodium Fast Reactors (SFR), one option that is being examined is a salt-cooled intermediate loop between the sodium-cooled primary system and a steam-water or a supercritical carbon dioxide power cycle – an advanced power cycle with potentially higher efficiency and lower capital costs than the traditional steam cycles. Liquid salts offer two potential advantages: (1) smaller equipment size because of the higher volumetric heat capacity of the salts and (2) no chemical exothermal reactions between the reactor, intermediate loop, and power cycle coolants. There is experience with this type of system because the Aircraft Reactor Experiment (the first MSR) used a sodium-cooled intermediate loop.

In the last five years, there has been a rapid growth in interest in the use of high-temperature (700 to 1 000°C) fluoride salts as coolants and for other functions in nuclear power systems (Forsberg, 2007b). This interest is a consequence of new applications for high-temperature heat and the development of new reactor concepts. The salt coolants have melting points between 350 and 500ºC and are, therefore, of use only in high-temperature systems. Nitrate salts with a peak operating temperature of around 600°C are the highest temperature commercial liquid coolant available today. The development of higher temperature salts as coolants would open new nuclear and non-nuclear applications. 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. Table 4.5 shows candidate salts for the different applications.

Table 4.5 - Candidate salts for different applications

<table>
<tr><th>Family</th><th>F/T</th><th>Fuel</th><th>Fuel cycle</th><th>Concepts</th><th>Salt type</th><th>Selection criteria</th></tr>
<tr><td rowspan="2">MSR-Breeder</td><td rowspan="2">T (epithermal)</td><td rowspan="2">liquid fuel</td><td rowspan="2">$^{233}U/Th$</td><td>primary coolant</td><td>$LiF-BeF_2-(HN)F_4$</td><td>melting temperature, low neutron absorption</td></tr>
<tr><td>secondary coolant</td><td>$NaBF_4-NaF$ (MSBR)</td><td>chemical compatibility with water, low cost</td></tr>
<tr><td rowspan="2">MSR-Breeder</td><td rowspan="2">F (non-moderated)</td><td rowspan="2">liquid fuel</td><td rowspan="2">$^{233}U/Th$</td><td>primary coolant</td><td>$LiF-(HN)F_4$</td><td>low melting temperature, low neutron absorption and moderation</td></tr>
<tr><td>secondary coolant</td><td></td><td></td></tr>
<tr><td>MSR-Breeder</td><td>F</td><td>liquid fuel</td><td>U-Pu-MA</td><td>primary coolant</td><td>$NaF-LiF-BeF_2-(HN)F_3$</td><td>actinide (Pu, MA) solubility</td></tr>
<tr><td>AHTR</td><td>T</td><td>solid fuel</td><td>U/Pu</td><td>primary coolant</td><td>$-BeF_2$ salts, -NaF salts</td><td>melting temperature, low cost</td></tr>
<tr><td>VHTR</td><td>T</td><td>solid fuel</td><td></td><td>intermediate coolant (H2)</td><td>$LiCl-KCl-MgCl_2$, $KF-KBF_4$ FLiNaK</td><td>heat transfert efficiency, low cost</td></tr>
<tr><td rowspan="2">SFR</td><td rowspan="2">F</td><td rowspan="2">solid fuel</td><td rowspan="2">U/Pu</td><td>primary coolant</td><td>$NaF-KF-ZrF_4$</td><td>low melting temperature, low neutron absorption and moderation</td></tr>
<tr><td>secondary coolant</td><td>nitrates, chlorides, hydrooxides</td><td>low melting temperature, chemical compatibility with water</td></tr>
</table>

F = fast; T = thermal

4.1.6.2 Status of cooperation

The decision for setting up a Provisional System Steering Committee (PSSC) for the MSR was taken by the GIF Policy Group in May 2004. The participating members are France, the United States and Euratom. Other countries (Japan and Russia) have been represented occasionally as observers in the meetings of the PSSC.

4.1.6.3 R&D objectives

Recent studies in Europe including national programs and the 5th and 6th European Commission Framework Programmes have confirmed the potential of MSR for breeding and waste minimization on both thermal (T) and fast (F) neutron spectrum (Renault, *et al.*, 2005). In parallel, viability analyses performed mainly in the USA have highlighted the assets of liquid salts as a coolant for heat transport at high temperature for various applications.

This renewal and diversification of interests have led the MSR PSSC to shift the R&D orientations and objectives initially promoted in the original Generation IV Technology Roadmap so as to encompass in a consistent body the different applications envisioned today for fuel and coolant salts.

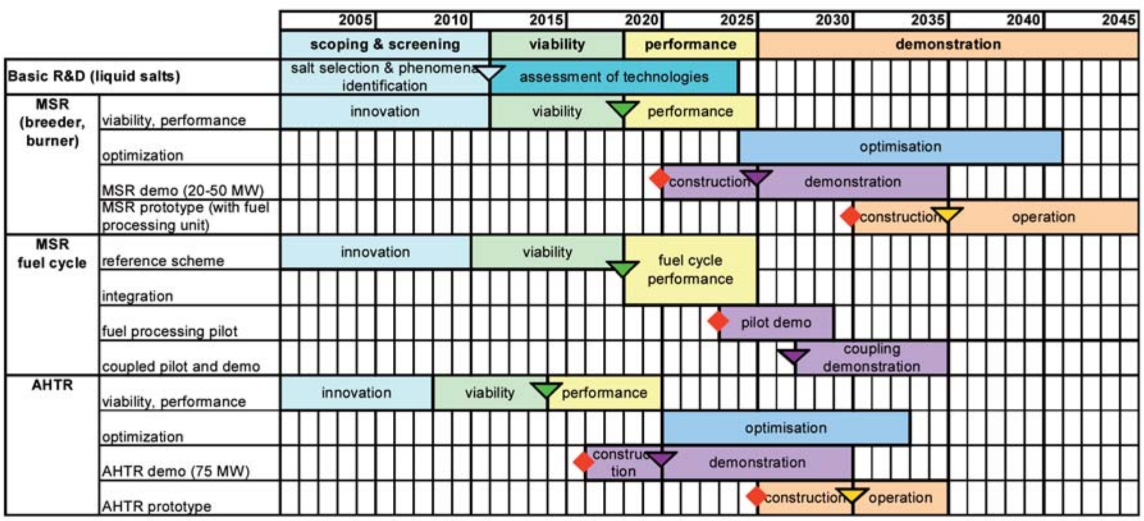
Cross-cutting R&D areas have been identified. A major commonality is the understanding and mastering of fuel and coolant salt technologies, including development of structural materials, fuel and coolant clean up, measurement of physical properties, chemical and analytical R&D. The updated MSR roadmap aims at optimizing the R&D effort to be conducted timely for the different promising applications of molten and liquid salts.

4.1.6.4 Milestones

The initial period (March 2005 – December 2005) was devoted to the definition of the general scope and preliminary identification of generic and specific fields of interest (R&D commonalities). In the second period (2006-2007), the MSR System Research Plan (SRP) will be submitted to the EG by mid-2008. Since mid-2007, the activity is focusing on the R&D Master Plan. The elaboration of Project Plans has been initiated October 2007.

According to the GIF Technology Roadmap, the viability phase would end in 2015 and the performance phase in 2020. However, the MSR PSSC has re-evaluated those milestones owing to the peculiar and more innovative position of MSR among other Generation IV systems. This led to identify a scoping and screening phase, prior to the viability and performance phases. The main milestones for the demonstration phase (final design, construction and operation of prototypes) have also been discussed. The MSR master plan is shown on Figure 4.14.

Figure 4.14 - MSR Master Plan



4.1.6.5 Main activities and outcomes

The MSR PSSC has significantly contributed to enhance and harmonize international collaboration within and beyond the context of the partners directly involved in the GIF activities related to MSR (Euratom, France and the United States).

The ALISIA project, sponsored by Euratom in the 6th FWP and started February 2007 (kick-off meeting), represents the European effort to develop liquid salt technologies for a variety of innovative nuclear technology applications, including the MSR. ALISIA is coordinated by the Chair of the MSR PSSC. The main objective of ALISIA is to strengthen the existing European network of expertise in this area, enabling the coordination of actions and sharing of results from national programs on MSR and other liquid salt applications. The ALISIA consortium involves 15 partners in 9 countries, including Russia (RRC-KI as a full partner), and gathers all major institutions involved in liquid salts R&D in Europe. In the short term, ALISIA is the major part of the Euratom contribution to GIF activities on the MSR and liquid salt applications.

Since the end of 2006, the Project Scientific Leader of International Science & Technology Centre (ISTC)-1606 has systematically attended as an observer the MSR PSSC meetings and has reported on R&D programs on liquid salts in Russia, therefore keeping a tight link with the complementary ISTC projects. The MSR PSSC has provided a forum for the definition of the new ISTC-3749 project, to be started in 2008 with official support by the Czech Republic, Euratom, France, Germany, the United States and the IAEA.

The MSR PSSC has fostered the exchange of students between the partners involved as members or observers. A student from Berkeley University (USA) is presently in France (Cadarache). Discussions are underway with Japan (Hokkaido University).

The MSR SRP to be released by mid-2008 underlines the major role of liquid salt chemistry in the viability demonstration, with such essential R&D issues as:

- fuel salt, coolant, fission product and tritium behavior;
- compatibility with structural materials for fuel and coolant circuits, as well as fuel processing materials development;
- on-line fuel processing;
- maintenance, instrumentation and controls development;
- safety issues, including interaction of liquid salts with sodium, water, air.

In addition, a rethinking of the safety approach is needed (fuel in liquid form and chemistry-controlled phenomena).

The development of adequate simulation tools coupling neutronics, thermal-hydraulics and chemistry (design and safety) together with basic models (physico-chemistry) is a high priority task. Experimental (analytical and integral) infrastructures are needed at mid term (liquid salt loops). The availability of experimental R&D means is rather limited in Europe. There is strong potential with existing facilities in Russia.

4.2 Methodology Working Groups

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 (GIF, 2002) in terms of economics, proliferation resistance and physical protection, and safety.

4.2.1 Economic Modeling Working Group

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:

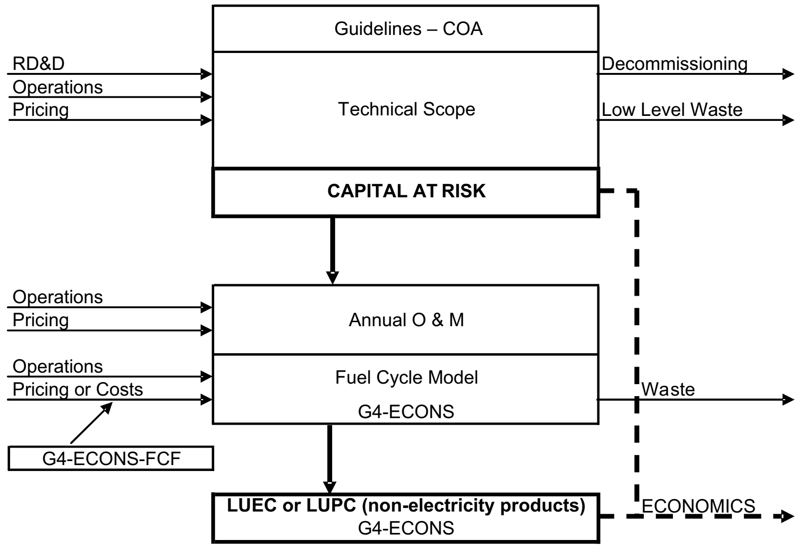
- to have a life cycle cost advantage over other energy sources (i.e., to have a lower levelized unit cost of energy on average over their lifetime); and
- to have a level of financial risk comparable to other energy projects (i.e., to involve similar total capital investment and capital at risk).

The main outcomes from the EMWG up to the end of 2007 include a document entitled *"Cost Estimating Guidelines for Generation IV Nuclear Energy Systems"* providing guidance on cost estimation approach, and a software package G4-ECONS facilitating the implementation of the GIF cost estimation methodology using a PC-based computer tool. The software package includes a detailed User's Manual and a simplified User's Guide.

The *"Cost Estimating Guidelines for Generation IV Nuclear Energy Systems"* (in short Guidelines, http://www.gen-4.org/Technology/horizontal/EMWG_Guidelines.pdf) is a comprehensive document setting forth the guidance and ground-rules for GIF system cost evaluations. Between 2004 and 2006, three revisions of the Guidelines were issued for GIF Members review and trial application. The Group carried out sample cost evaluations of Generation III nuclear systems to test the robustness of the methodology and validate its results. In 2007, the EMWG published Revision 4 of the Guidelines. This document provides a uniform set of assumptions, a comprehensive Code of Accounts (COA), and cost-estimating guidelines to be used in developing cost estimates for advanced nuclear energy systems. It discusses the development of all relevant life cycle costs for Generation IV systems, including the planning, research, development, demonstration (including prototype), deployment, and commercial stages.

The software package, *G4-ECONS*, was developed by the EMWG to facilitate implementation of the Guidelines, especially with respect to assessing whether the GIF economic goals are met by the nuclear systems being evaluated. Figure 4.15 provides a schematic representation of the integrated model developed by the EMWG and implemented in the software package *G4-ECONS*.

Figure 4.15 - Structure of the GIF integrated model for cost estimation



Version 1 of the software – *G4-ECONS 1.0* – was released for use by GIF Members in 2006. It included capabilities to estimate electricity generation costs for Generation IV systems. The Group carried out numerous sample calculations to test the software on known systems.

In particular, cost estimations were carried out with the software using input data available for a Generation III nuclear power plant (System 80+) and for a Generation IV system (the Japanese Sodium Fast Reactor). Those calculations were performed by the EMWG for benchmarking and validation purposes and do not intend to represent cost estimates for the systems considered. Their results have been compared with those of other, more detailed, analyses conducted by different experts and the comparison shows good agreement between the results obtained using the EMWG methodology and *G4-ECONS* and other results. The test applications demonstrated that the software produces reasonable results for Generation IV systems and that it could also be used for earlier reactor designs.

In 2007, Version 2 of the software – *G4-ECONS 2.0* – was completed and issued to GIF. This version provides the ability to analyze applications other than electricity production, such as hydrogen production or desalination, or a combined production of electrical and non-electrical energy production. An additional module was developed and implemented to calculate the cost of fuel cycle services not available on the market. This software – *G4ECONS-FCF* – allows estimating the cost of a service, e.g., reprocessing, starting from the fuel cycle facility investment and running costs. Sample calculations were performed as the software was completed to validate the approach adopted.

In June of 2007, the EMWG presented its methodological approach and key results from its sample calculations at the annual meeting of the American Nuclear Society in Boston, MA, United States. A total of seven papers were presented explaining the content of the Guidelines and the use of *G4-ECONS*. The sessions were well attended and the papers well received. Also, in 2006 and 2007, several members of the EMWG presented papers in various international meetings and published articles in scientific journals based on the outcomes from the work carried out within the Group.

The PG has authorized public release of the Guidelines and the software package in order to foster a wide use of the cost estimation approach and tools developed by GIF. Further improvement of the Guidelines and software is planned, based upon feedback from GIF and other users of the methodology, including integration of *G4-ECONS 2.0* with *G4-ECONS-FCF*.

4.2.2 Proliferation Resistance and Physical Protection Working Group

The PRPPWG was created in December 2002 to develop and implement an improved evaluation methodology to assess Generation IV systems with respect to GIF proliferation resistance and physical protection goals, i.e., generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism. The methodology is intended for performing PR and PP evaluation at the earliest phase of system design, when flow diagrams and physical arrangement drawings are first being developed and initial target identification can be performed.

The PRPPWG has been meeting about three times a year since the end of 2002 to fulfill its original mission of developing:

- concepts and vocabulary to discuss diversion of materials and misuse of technologies by host States (proliferation resistance) and theft and sabotage by non-host-State actors including terrorists (physical protection robustness); and
- a systematic methodology for evaluating comparative PR and PP, to support designers and policy makers in comparing options.

An illustrative example case was formulated (the Example Sodium Fast Reactor, or ESFR, loosely based on the US Integral Fast Reactor concept), and a portion of the system (a part of the pyroprocessing facility) was evaluated using three different evaluation approaches.

During the course of this work, a fairly broad international consensus was gained on concepts and terms and the evaluation approach. The methodology and sample evaluations have been documented in a report with appendices and a preliminary user's guide.

Revision 5 of the PR and PP Evaluation Methodology report was released for unrestricted distribution in 2006 (http://www.gen-4.org/Technology/horizontal/PRPPEM.pdf). Three major workshops were conducted by the PRPPWG with representatives of GIF SSCs, policy makers, and other stakeholders. They were conducted in North America, Europe, and Asia to allow the relevant local interested parties to attend these events.

The methodology documented in Revision 5 covers PR and PP of Generation IV systems in a comprehensive manner. Clear definitions of PR and PP are important to set the scope of the evaluation methodology. The definition of PR adopted by the Group agrees with the definition established at the international workshop sponsored by the IAEA in Como, Italy, in 2002. The adopted definitions are:

- ***Proliferation resistance*** is that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material and the misuse of technology by the host State seeking to acquire nuclear weapons or other nuclear explosive devices.
- ***Physical protection (robustness)*** is that characteristic of a nuclear energy system that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices and the sabotage of facilities and transportation by sub-national entities or other non-host State adversaries.

Figure 4.16 illustrates the methodological approach at its most basic. For a given system, analysts define a set of ***challenges***, analyze ***system response*** to these challenges, and assess ***outcomes***. The challenges to the nuclear energy system are the threats posed by potential "proliferant" States and by sub-national adversaries. The technical and institutional characteristics of the Generation IV systems are used to evaluate the response of the system and determine its ***resistance*** to proliferation threats and ***robustness*** against sabotage and terrorism threats. The outcomes of the system response are expressed in terms of PR and PP ***measures*** and assessed.

Figure 4.16 - Basic framework for the PR&PP evaluation methodology

CHALLENGES	→	**SYSTEM RESPONSE**	→	**OUTCOMES**
Threats		***PR & PP***		***Assessment***

The structure for the PR and PP evaluation can be applied to the entire fuel cycle or to portions of a nuclear energy system. The methodology is organized as a progressive approach to allow evaluations to become more detailed and more representative as system design progresses. PR and PP evaluations may be performed at the earliest stages of design when flow diagrams are first developed in order to systematically integrate proliferation resistance and physical protection robustness into the designs of Generation IV systems along with the other high-level technology goals of sustainability, safety and reliability, and economics. This approach provides early, useful feedback to designers, program policy makers, and external stakeholders from basic process selection (e.g., recycling process and type of fuel), to detailed layout of equipment and structures, and to facility demonstration testing.

To facilitate the comparison of different evaluations, a standard Reference Threat Set (RTS) can be defined, covering the anticipated range of actors, capabilities, and strategies for the time period being considered. Reference Threat Sets should evolve through the design and development process of nuclear fuel cycle facilities, ultimately becoming Design Basis Threats (DBTs) upon which regulatory action is based.

For PR, the threats include:

- Concealed diversion of declared materials.
- Concealed misuse of declared facilities.
- Overt misuse of facilities or diversion of declared materials.
- Clandestine dedicated facilities.

For PP, the threats include:

- Radiological sabotage.
- Material theft.
- Information theft.

The PR and PP methodology does not determine the probability that a given threat might or might not occur. Therefore, the selection of what potential threats to include is performed at the beginning of a PR and PP evaluation, preferably with input from a peer review group organized in coordination with the evaluation sponsors. The uncertainty in the system response to a given threat is then evaluated independently of the probability that the system would ever actually be challenged by the threat. In other words, PR and PP evaluations are contingent on the challenge occurring.

When threats have been sufficiently detailed for the particular evaluation, analysts assess system response, which has four components:

1. **System Element Identification.** The nuclear energy system is decomposed into smaller elements or subsystems at a level amenable to further analysis. The elements can comprise a facility (in the systems engineering sense), part of a facility, a collection of facilities, or a transportation system within the identified nuclear energy system where acquisition (diversion) or processing (PR) or theft/sabotage (PP) could take place.

2. **Target Identification and Categorization.** Target identification is conducted by systematically examining the nuclear energy system for the role that materials, equipment, and processes in each element could play in each of the strategies identified in the threat definition. Initial target identification can occur when the first flow diagrams are developed. PR targets are nuclear material, equipment, and processes to be protected from threats of diversion and misuse. PP targets are nuclear material, equipment, or information to be protected from threats of theft and sabotage. Targets are categorized to create representative or bounding sets for further analysis.

3. **Pathway Identification and Refinement.** Pathways are potential sequences of events and actions followed by the actor to achieve objectives. For each target, individual pathways are divided into segments through a systematic process, and analyzed at a high level. Segments are then connected into full pathways and analyzed in detail. Selection of appropriate pathways will depend on the scenarios themselves, the state of design information, the quality and applicability of available information, and the analyst's preferences. Initial pathway identification can occur when the first physical arrangement drawings are developed.

4. **Estimation of Measures.** The results of the system response are expressed in terms of PR and PP measures. Measures are the high-level characteristics of a pathway that affect the likely decisions and actions of an actor and therefore are used to evaluate the actor's likely behavior and the outcomes. For each measure, the results for each pathway segment are aggregated as appropriate to compare pathways and assess the system so that significant pathways can be identified and highlighted for further assessment and decision making.

For PR, the measures are:

- Proliferation Technical Difficulty – The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the multiple barriers to proliferation.
- Proliferation Cost – The economic and staffing investment required to overcome the multiple technical barriers to proliferation including the use of existing or new facilities.
- Proliferation Time – The minimum time required to overcome the multiple barriers to proliferation (i.e., the total time planned by the host State for the project).
- Fissile Material Type – A categorization of material based on the degree to which its characteristics affect its utility for use in nuclear explosives.
- Detection Probability – The cumulative probability of detecting a proliferation segment or pathway.
- Detection Resource Efficiency – The efficiency in the use of staffing, equipment, and funding to apply international safeguards to the nuclear energy system.

For PP, the measures are:

- Probability of Adversary Success – The probability that an adversary will successfully complete the actions described by a pathway and generate a consequence.
- Consequences – The effects resulting from the successful completion of the adversary's action described by a pathway.
- Physical Protection Resources – the staffing, capabilities, and costs required to provide PP, such as background screening, detection, interruption, and neutralization, and the sensitivity of these resources to changes in the threat sophistication and capability.

By considering these measures, system designers can identify design options that will improve system PR and PP performance. For example, designers can reduce or eliminate active safety equipment that requires frequent operator intervention, making the safety equipment more inaccessible for PP sabotage pathways.

The final steps in PR and PP evaluations are to integrate the findings of the analysis and to interpret the results. This commonly involves comparisons of pathways, using the PR and PP measures, to identify potential weaknesses and vulnerabilities. Evaluation results should include best estimates for numerical and linguistic descriptors that characterize the results, distributions reflecting the uncertainty associated with those estimates, and appropriate displays to communicate uncertainties.

The information is intended for three types of users: system designers, program policy makers, and external stakeholders. Thus, the analysis of the system response must furnish results easily displayed with different levels of detail. Program policy makers and external stakeholders are more likely to be interested in the high-level measures for dominant pathways, while system designers will be interested in measures and metrics that more directly relate to the optimization of the system design and selection between design options.

Based on the work done over the past several years, it is evident that further progress on the PR and PP methodology requires a more comprehensive evaluation of a complete reactor/fuel cycle system to gain practical experience, to discern the needs for further methodology development and presentation of results, and to confirm the usefulness and usability of the evaluation methodology. An action aimed at reaching this goal has been launched, which will call for substantial involvement from the different GIF members and a strong interaction between the PRPPWG and the SSCs.

4.2.3 Risk and Safety Working Group

The RSWG was created early in 2005. In addition to members nominated by GIF countries and Euratom, each SSC and the PRPPWG were invited to send one representative in the RSWG.. The RSWG may involve other experts from external organizations as resources for advice and resolution of specific tasks. Through its co-chairs the RSWG maintains an active interface with the IAEA, which participates as an observer in the RSWG, and other international organizations that address safety and regulatory issues.

The primary objective of the RSWG is to promote a harmonized approach on safety, risk and regulatory issues in development of Generation IV systems. The RSWG will advise and assist the Experts Group and the Policy Group particularly on:

- Gen IV safety goals and evaluation methodologies to be considered in the design and thus to guide R&D plans
- interactions with the nuclear safety regulatory community, the IAEA and other relevant stakeholders

The RSWG achieved 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:

- the content of a non-prescriptive cohesive safety philosophy applicable to all the Generation IV systems;
- high-level safety objectives for Generation IV systems, and potential attributes which may be useful in meeting those objectives;
- basic principles for an approach to the design and the assessment of innovative systems including assessment of the defense-in-depth principle application and the treatment of severe plant conditions;
- the role of passive features; and
- the possible role of available assessment tools, e.g., Probabilistic Safety Assessment (PSA), and the need for developing additional indicators and tools.

The first major work product from the Group has been the *Report on the Safety of Generation IV Nuclear Systems* which presents the findings and recommendations based on work completed by mid 2007. The primary objective of this report is to discuss GIF safety goals, safety principles and evaluation methodology of next generation nuclear systems. As a complementary objective the document aims at helping identify necessary safety-related crosscutting R&D. The identification of system-specific and dedicated R&D efforts also may be facilitated by the findings and recommendation from the report which provides in addition insights to assist the EG and the PG in the definition of adequate safety and related R&D. Major findings and recommendations presented in the report include the following:

- Opportunities exist to further improve on nuclear power's already excellent safety record in most countries.
- The quantitative safety objectives applicable to the reactors of the third generation are ambitious and offer an improved level of safety compared with Generation II systems,. These objectives can be used as a reference point for Generation IV systems. Meanwhile, the RSWG believes that, although not required, further increases in the level of safety are possible through progress in knowledge and technologies and the early application of a cohesive safety philosophy.
- Potential safety improvements should simultaneously be based on several elements, for which more stringent technical requirements, compared to those applicable to the third generation of nuclear systems, should be considered only if they can bring a real and demonstrable benefit.
- The diversity of the Generation IV systems and the need for a balanced strategy for design and assessment of these systems justify re-examination of the safety approach.
- The principle of defense-in-depth has served the nuclear power industry well, and must be preserved in the design of Generation IV systems.
- The Generation IV design process should be driven by a risk-informed approach (i.e. considering both deterministic and probabilistic methods).
- For Generation IV systems, in addition to prototyping and demonstration, modeling and simulation should play a large role in the design and the assessment.
- The Design Basis for Generation IV systems should cover the full range of safety significant conditions.
- Objectives and practices for the design improvement are identified within the report.
- The recommendation for the improvement of safety demonstration's robustness rests on the capacity of the designer and the developer to demonstrate and to guarantee exhaustiveness in the recognition of risks stemming from phenomena considered for the design.
- Practical instruments are suggested to be used by the designers to support the design activity as well as the assessment activities.

The report also presents a pilot application of the Objective Provision Tree (OPT) methodology for assessment of defense-in-depth implementation in the design and operation of a nuclear power plant along with a simplified visualization of the OPT methodology. Insights from the pilot application of this methodology are also summarized.

Following publication of the report, the work of the RSWG will now turn to the development, demonstration, and documentation of a formalized methodology for developing and assessing the safety of Generation IV systems. Through increasing interaction with the SSCs, the group will provide consultation to help guide the development of a robust safety basis for the evolving Generation IV systems. In addition, the RSWG will continue its interaction with Proliferation Resistance and Physical Protection Working Group to further facilitate integrated consideration of safety, proliferation resistance and physical protection goals. Consistency among the two methodologies is a key objective for future work. Finally, while the report issued in 2007 deals only with reactor technology, issues associated with fuel cycle facilities and processes will be addressed in the future work of the RSWG.

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HTTR Project
JAEA, Japan

PBMR Project
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International collaboration has been an important element in the successful development of peaceful applications of nuclear energy since its very beginning. With the renewed interest of policy makers worldwide for the nuclear option, international endeavors in the field of nuclear research and development have multiplied. Several of them have objectives and scopes offering opportunities for synergies with GIF and, accordingly, GIF has established communications to ensure coordination whenever appropriate and avoid duplication of efforts that would be detrimental to members contributing to more than one of those endeavors. Brief descriptions of international programs are given below with an explanation of their relationship to GIF.

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

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was initiated by the IAEA in 2001 based on a resolution of its General Conference in 2000.
(http://www.iaea.org/worldatom/Programmes/Nuclear_Energy/NENP/NPTDS/Projects/inpro.html)
INPRO has 28 Members contributing to its program of work. 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. The project aims at bringing together technology holders and users so that they can contribute jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles (http://www.iaea.org/INPRO).

INPRO has a broad variety of missions including analyzing the sustainable development of nuclear energy, facilitating international cooperation for the deployment of new nuclear systems, and responding to the needs of developing countries interested in new nuclear systems, and developing methodologies for assessing new nuclear systems. These missions complement those of GIF, and potential synergies with GIF in the area of evaluation methodologies are apparent. Exchange of information is beneficial for both endeavors and is therefore promoted by countries and organizations participating in GIF and INPRO. Additionally the Secretariats of both projects have established mechanisms for cooperation on topics of mutual interest.

The deployment focus of INPRO and the involvement in the project of potential users of systems being developed within GIF provide valuable information to GIF members on their commercial prospects in the world marketplace. On the other hand, the technology development focus of GIF and the nuclear research programs undertaken under its auspices are valuable to informing INPRO members of the likely pace and direction of innovations.

In terms of membership, all country Members of GIF are also members of INPRO; this greatly facilitates an ongoing exchange of information between the two initiatives. Furthermore, the secretariats of GIF and INPRO have sponsored yearly meetings to exchange information on progress and help coordinate and cross fertilize the activities of the two projects.

5.2 Global Nuclear Energy Partnership (GNEP)

In 2006, the US Department of Energy introduced the Global Nuclear Energy Partnership (GNEP, http://www.gnep.energy.gov/), motivated by the realization that future deployment of nuclear energy would likely be worldwide and would further increase concerns about the fate of spent nuclear fuel and the proliferation of technologies for making special nuclear materials.

At present thirty-five countries participate in GNEP – nineteen as Partners and the remaining ones as candidate partner and observer countries. Three intergovernmental organizations (GIF, IAEA and Euratom) are observers.

The cooperation to be undertaken as part of GNEP has the following objectives, excerpted from the GNEP Statement of Principles (http://www.gnep.energy.gov/pdfs/gnepSOP_091607.pdf):

- Expand nuclear power to help meet growing energy demand in a sustainable manner and in a way that provides for safe operations of nuclear power plants and management of waste[s].
- In cooperation with the IAEA, continue to develop enhanced nuclear safeguards to effectively and efficiently monitor nuclear materials and facilities, to ensure that nuclear energy systems are used only for peaceful purposes.
- Establish international supply frameworks […] providing options for […] fostering development while reducing the risk of nuclear proliferation by creating a viable alternative to acquisition of sensitive fuel cycle technologies.
- Develop, demonstrate and in due course deploy advanced fast reactors that consume transuranic elements from recycled spent fuel.
- Promote the development of advanced, more proliferation resistant nuclear power reactors appropriate for the power grids of developing countries and regions.
- Develop and demonstrate, *inter alia*, advanced technologies for recycling spent nuclear fuel […] with the long-term goal of ceasing separation of plutonium and eventually eliminating stocks of separated civilian plutonium. […]

While GNEP encompasses a broader policy vision than the technology-focused GIF goals, both endeavors have similar goals for future nuclear systems, most notably improvement of waste management and enhancement of proliferation resistance. Cooperation between GNEP, GIF and INPRO is affirmed in the GNEP Statement of Principles.

5.3 Multinational Design Evaluation Program (MDEP)

The Multinational Design Evaluation Program (MDEP, http://www.nea.fr/mdep/welcome.html) is a multinational initiative launched by the US Nuclear Regulatory Commission to leverage the knowledge of the national regulatory authorities who will be tasked with the review of new reactor designs. 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.

Currently, ten countries participate in MDEP: 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.

MDEP is proceeding in three stages. In the first stage which started in 2005, nuclear regulators are using technical data gathered during the certification of a reactor design in one country for its certification in another, thereby avoiding duplication of work. In particular, the nuclear regulatory authorities of France and Finland are working with their US counterpart on the review and evaluation of the European Pressurized water Reactor (EPR). The second MDEP stage which was initiated mid-2006 focuses on identifying common regulatory practices and regulations that enhance the safety of new reactor designs. Ultimately, this phase is expected to lead to a convergence of codes, standards and safety goals in the participating countries. During the third stage, which is a much longer-term endeavor, the lessons learnt during the earlier stages will be used to facilitate the licensing of Generation IV reactor designs.

Exchange of information and realization of synergies between MDEP and GIF are facilitated by the fact the NEA staff providing the Technical Secretariat for MDEP during its second stage also serve as the Technical Secretariat for the GIF Risk and Safety Working Group.

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Generation IV nuclear energy systems are expected to offer significant improvements over existing systems in the areas of economics; safety and reliability; proliferation resistance and physical protection; and sustainability. The GIF Technology Roadmap evaluated over 100 system concepts, identified six with the greatest promise and outlined the R&D necessary to bring them to commercialisation in the 2030 timeframe. The Generation IV International Forum (GIF) members are collaborating on the R&D needed to develop generation IV nuclear energy systems, beyond what is currently being undertaken by industry.

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