ANNUAL REPORT

2010

GEN IV International Forum[SM]

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Message from the Chairman

I would like to say thank you to all of you who participate in the GIF and support its activities. I am pleased to inform you of the publication of the *2010 GIF Annual Report*.

First of all, I should refer to the accident at the nuclear power plant in Fukushima, Japan. This unprecedented disaster was caused by the giant earthquake and following tsunami on March 11, 2011. Safety is the most important requirement for nuclear power, and we have devoted steady effort to enhance safety. However, this accident has occurred and I personally regret it. I hope that all measures being taken by the people concerned will bring about good results as soon as possible. Already underway in many countries, we have to make serious efforts to review and enhance the safety of nuclear power plants and never let such an accident happen again.

The nuclear disaster notwithstanding, many nations have recognized the effectiveness of nuclear power as a viable answer for energy resources and environmental issues. Countries expecting high economic growth are planning to introduce nuclear power. They have engaged in international cooperation such as the international project on innovative nuclear reactors and fuel cycles (INPRO) and the international framework for nuclear energy cooperation (IFNEC) to promote the preparation required for the introduction of nuclear power. China and India, which have already introduced nuclear power, are increasing deployment with high objectives to meet their rapidly increasing electricity demands.

In these circumstances, many countries request that nuclear power plants have higher safety and reliability, as well as high proliferation resistance and the sustainability that comes with the efficient use of resources and the reduction of environmental burden. Of course, it has to be economical for commercial use. The Generation IV (Gen-IV) reactor systems that we are developing strive to meet these exact requirements and are indispensable for mankind's future.

Many nations are conducting research and development on Gen-IV systems. Monju, a prototype of the sodium fast reactor (SFR) in Japan, restarted in May 2010 for the first time in 14 years. The China experimental fast reactor (CEFR) attained its first criticality in July 2010. Further, the prototype fast breeder reactor (PFBR) in India is under construction, aiming at its first criticality in 2012. The Russian fast reactor BN-800 is expected to be critical in 2014. These reactors are expected to be undertaking vital roles for the future development of SFR technology. The construction of several demonstration-scale reactors is planned during the 2020s in different countries including the advanced sodium technological reactor for industrial demonstration (ASTRID) in France and the Japan standard fast reactor (JSFR) in Japan.

The European Union (EU) has launched an initiative to promote industrial development for next-generation nuclear technologies named the European sustainable nuclear industrial initiative (ESNII). The EU has also identified development targets for other fast reactor technologies, including the lead-cooled fast reactor (LFR) demonstrator ALFRED, and the gas-cooled fast reactor (GFR) demonstrator ALLEGRO.

The United States has ambitions to commercialize several kinds of small modular reactors including Gen-IV systems. The very-high-temperature reactor (VHTR), including the next-generation nuclear plant (NGNP) in the United States, is also being developed. Within the GIF, seven members participate in the cooperation on the VHTR.

Other than those above, R&D has continued for the supercritical-water-cooled reactor (SCWR), which is expected to have better economics due to high thermal efficiency and a simplified system, and for the molten salt reactor (MSR) which can utilize thorium as well as uranium.

There has been much progress on our activities in 2010. In March 2010, the accession to the GIF framework agreement by Russia came into force. Russia also signed the SFR system arrangement and expressed interest in all related projects. Regarding the MSR and the LFR, although preparations of their system arrangements have taken time, MOUs were concluded between France and the EU, and Japan and the EU respectively.

As regards GIF projects, the arrangements on the materials for VHTR and on the thermal-hydraulics and safety of the SCWR were signed in April and December respectively. The signing of a new SFR project arrangement for system integration and assessment is expected shortly, in addition to the four project arrangements already effective – advanced fuel, global actinide cycle international demonstration, component design and balance of plant, and safety and operation. Cooperation on each system is being actively promoted within the established frameworks. Methodology working groups are also producing fruitful results.

We also have plans for future GIF R&D activities to promote the development of systems in both the viability phase and the performance phase. With the progress of our cooperation as well as the need for securing higher safety considering the Fukushima accident, it will be necessary to establish a set of common safety criteria for Gen-IV systems. Gen-III nuclear systems have their design criteria based on accumulated experience in each country, but Gen-IV systems do not have such criteria yet. I believe GIF is able to contribute to such discussions and studies. We have just started a review for establishing the common design criteria for safety of the SFR and are planning to expand this effort to other Gen-IV systems.

GIF has cooperated with other international cooperation frameworks for nuclear power utilization such as INPRO and IFNEC by holding a joint workshop and participating in meetings. We would like to further enhance cooperation with other international organizations.

Yutaka SAGAYAMA

GIF Chairman – August 2011

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Opening the International Conference on Access to Civil Nuclear Energy in March 2010 in Paris, the French President Nicolas Sarkozy noted in his speech that the worldwide population is growing and becoming ever more prosperous. As a result, we will need 40% more energy by 2030. […] We need to save the planet and we need to abide by our stated goals in terms of climate change. For all of these reasons, we need nuclear power generation. We cannot do this solely through the use of renewable energies. We need renewable but we also need nuclear power generation if we are to protect the planet. Indeed, to achieve its scenario 450, which aims to the stabilization of greenhouse gas concentration at a level of 450 ppm, to limit the increase of the average world temperature to 2°C, the IEA *World Energy Outlook 2009* assumes an increase of the nuclear energy capacity of 378 GW in the period 2008-2030. While uranium resources can withstand this evolution,[1] it is important to find ways to improve the nuclear fuel cycle, especially aiming at reducing the quantity of final waste, to be kept in repositories. Also, the application of nuclear energy should be checked, together with electricity production, in the provision of process heat to non-nuclear industries, replacing fossil fuel as heat sources.

The Generation IV International Forum (GIF), created in 2000 to foster international collaboration at a detailed level of actual R&D, is a cooperative international endeavor, organized to develop the research necessary to test the feasibility and performance capabilities of fourth generation nuclear systems, with the goal of making such systems deployable in large numbers around 2030. Since its beginning, GIF members stated the following goals for the fourth generation of nuclear power plants when compared to previous generations:

a) improve sustainability (including effective fuel utilization and minimization of waste);

b) improve economics (competitiveness with respect to other energy sources);

c) improve safety and reliability (e.g. no need for offsite emergency response); and

d) improve proliferation resistance and physical protection.

After an in-depth analysis of the different available concepts, whatever their level of development, the Forum selected six concepts as the most promising, and decided to focus R&D on these systems:

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

Active members of the GIF are Canada, Euratom, France, Japan, People's Republic of China, Republic of Korea, Republic of South Africa, Russian Federation, Switzerland and the United States. Altogether, they represent around 90% of the world installed nuclear capacity for producing electricity, and all key technology holders. The forum is led by the policy group, where all members are represented, and currently chaired by Japan since December 2009, assisted by vice-chairs from France and United States.

1. *Uranium 2009: Resources, Production and Demand*, Joint report by OECD/Nuclear Energy Agency and the International Atomic Energy Agency.

The year 2010 has seen some important achievements and decisions regarding these six systems. For example, two sodium-cooled fast reactors (re)started this year: Monju in Japan restarted after being down for 15 years, and the China experimental fast reactor (CEFR) reached its first criticality in July. Despite the closure of PBMR, projects for new VHTR are ongoing, including the HTR-PM in China, now ready for construction, and the NGNP in United States, while the HTTR prototype in Japan achieved successfully 50 days of continuous operation at 950°C.

This fourth GIF Annual Report includes four chapters in addition to this introduction plus three appendices. Chapter 2 describes the membership and organization of GIF, the structure of its cooperative research and development arrangements as well as the status of members' participation in such arrangements. Chapter 3 summarizes GIF R&D plans, activities and achievements during 2010. 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 on the development of methodologies for assessing Generation IV systems with respect to the established goals of GIF. Chapter 4 reviews the cooperation between GIF and other international programs dealing with the development of nuclear energy. Appendix 1 provides an overview on the goals of Generation IV nuclear energy systems and an outline of the main characteristics of the six systems selected for joint development by GIF. Appendix 2 presents the objectives that have been set for the various System Steering Committees and the associated Project Management Boards for the period extending from 2010 to 2015. Finally, Appendix 3 provides a list of abbreviations and acronyms (with the corresponding definitions) which are used in this report or are relevant to GIF activities.

*The public website (**www.gen-4.org**), regularly updated, provides a complete description of the GIF, as well as technical and scientific information on Generation IV systems and methodologies.*

CHAPTER 2 GIF MEMBERSHIP, ORGANIZATION AND R&D COLLABORATIONS

2.1 GIF Membership

The Generation IV International Forum has thirteen members, as shown in Table 2-1, which are signatories of its founding document, the *GIF Charter*. Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, the United Kingdom and the United States signed the GIF Charter in July 2001. Subsequently, it was signed by Switzerland in 2002, Euratom[2] in 2003, and the People's Republic of China and the Russian Federation, both in 2006. Signatories of the Charter are expected to maintain an appropriate level of active participation in GIF collaborative projects.

Table 2-1: Parties to GIF Framework Agreement and System Arrangements as of December 2010

Member	Implementing agents	Date of signature or receipt of the instrument of accession	System arrangements (SA)			
			GFR	SCWR	SFR	VHTR
Argentina (ARG)						
Brazil (BRA)						
Canada (CAN)	Department of Natural Resources (NRCan)	02/2005		11/2006		11/2006
Euratom (EUR)	European Commission's Joint Research Centre (JRC)	02/2006	11/2006	11/2006	11/2006	11/2006
France (FRA)	Commissariat à l'énergie atomique et aux énergies alternatives (CEA)	02/2005	11/2006		02/2006	11/2006
Japan (JAP)	Agency for Natural Resources and Energy (ANRE) Japan Atomic Energy Agency (JAEA)	02/2005	11/2006	02/2007	02/2006	11/2006
People's Republic of China (CHN)	China Atomic Energy Authority (CAEA) Ministry of Science and Technology (MOST)	12/2007			03/2009	10/2008
Republic of Korea (KOR)	Ministry of Education, Science & Technology (MEST) Nation Research Foundation (NRF)	08/2005			04/2006	11/2006
Republic of South Africa (ZAF)	Department of Minerals and Energy (DME)	04/2008				
Russian Federation (RUS)	ROSATOM	12/2009			07/2010	
Switzerland (CHE)	Paul Scherrer Institute (PSI)	05/2005	11/2006			11/2006
United Kingdom (GBR)						
United States (USA)	Department of Energy (DOE)	02/2005			02/2006	11/2006

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

Among the Signatories to the charter, ten members (Canada, Euratom, France, Japan, the People's Republic of China, the Republic of Korea, the Republic of South Africa, the Russian Federation, Switzerland and the United States) have signed or acceded to the framework agreement (FA) as shown in Table 2-1. Parties to the framework agreement formally agree to participate in the development of one or more Generation IV systems selected by GIF for further R&D. Each party to the framework agreement designates one or more implementing agents to undertake the development of systems and the advancement of their underlying technologies.

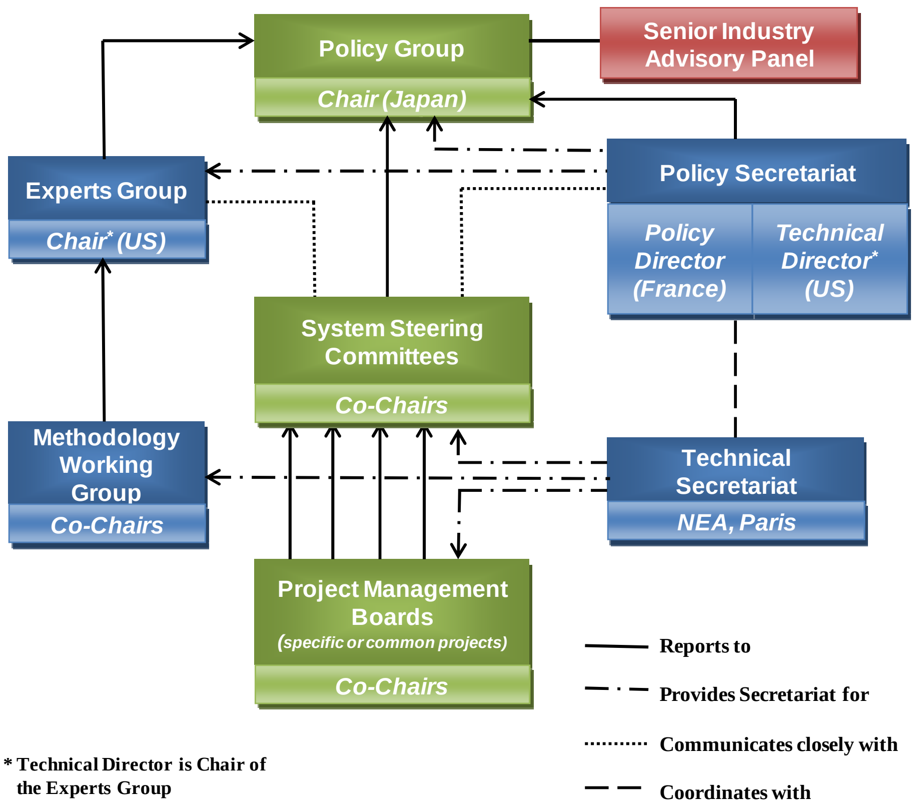
Argentina, Brazil and the United Kingdom[3] have signed the GIF charter but did not accede to or ratify the FA; accordingly, within the GIF they are designated as "non-active members".

Members interested in implementing cooperative R&D on one or more of the selected systems have signed corresponding system arrangements (SA) consistent with the provisions of the FA. The participation of GIF members in system arrangements is also shown in Table 2-1.

2.2 GIF Organization

The GIF charter provides a general framework for GIF activities and outlines its organizational structure. Figure 2-1 gives a schematic representation of the GIF governance structure and indicates the relationship among different GIF bodies which are described below.

Figure 2-1: GIF Governance Structure



3. The United Kingdom participates in GIF activities through Euratom.

As detailed in its charter and subsequent GIF policy statements, the GIF is led by the policy group (PG) which is responsible for the overall steering of the GIF cooperative efforts, the establishment of policies governing GIF activities, and interactions with third parties. Every GIF member nominates up to two representatives in the policy group. The PG usually meets two or three times each year (Figure 2-2).

Figure 2-2: Policy Group in Pretoria (October 2010)

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

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

The GIF charter and framework agreement allow for the participation of organizations from public and private sectors of non-GIF members in PAs and in the associated PMBs, but not in SSCs. Participation by organizations from non-GIF members requires unanimous approval of the corresponding system steering committee. The PG may provide recommendations to the SSC on the participation in GIF R&D projects by organizations from non-GIF members.

Three methodology working groups (MWGs) are responsible for developing and implementing methods for the assessment of Generation IV systems against GIF goals in the fields of economics, proliferation resistance and physical protection, and risk and safety. Those groups – the economic modeling working group (EMWG), the proliferation resistance and physical protection working group (PRPPWG), and the risk and safety working group (RSWG) – report to the experts group which provides guidance and periodically reviews their work plans and progress. Members of the MWGs are appointed by the policy group representatives of each 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 and technical aspects. The SIAP contributes to strategic reviews and guidance of the GIF R&D activities in order to ensure that technical issues impacting on future potential introduction of commercial Generation IV systems are taken into account. In particular, the SIAP provides guidance on taking into account investor-risk reduction and incorporating the associated challenges in system designs at an early stage of development.

The GIF secretariat is the day-to-day coordinator of GIF activities and communications. It includes two groups: the policy secretariat and the technical secretariat (TS). 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 Nuclear Energy Agency (NEA) of the organization for economic cooperation and development (OECD), supports the SSCs, PMBs and MWGs. The NEA is entirely resourced for this purpose through voluntary contributions from GIF members, either financial or in kind (e.g. providing a cost-free expert for supporting TS work).

2.3 Participation in GIF R&D Projects

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

As of 31 December 2010, system arrangements have been signed by several members for four systems (GFR, SCWR, SFR and VHTR). For the MSR and the LFR systems, memorandum of understanding were signed in October 2010 and November 2010 respectively by interested members,[4] while collaborative R&D is pursued under the auspices of provisional SSCs.

Several project arrangements (PAs) have been signed and are effective for the four systems having agreed on a SA:

- Gas-cooled fast reactor: conceptual design and safety (CD&S) PA.
- Sodium-cooled fast reactor: advanced fuel (AF) PA; global actinide cycle international demonstration (GACID) PA; component design and balance-of-plant (CDBOP) PA; and safety and operation (SO) PA.
- Supercritical-water-cooled reactor: thermal-hydraulics and safety (TH&S) PA; and materials and chemistry (M&C) PA.

4. MSR: Euratom, France/LFR: Euratom, Japan.

- Very-high-temperature reactor: fuel and fuel cycle (FFC) PA; hydrogen production (HP) PA; and materials (MAT) PA.

Other PAs are defined already and their membership agreed upon by interested parties on a provisional basis. Table 2-2 shows the list of signed arrangements and provisional cooperation within GIF as of 31 December 2010.

Table 2-2: Status of Signed Arrangements and Provisional Cooperation within GIF as of 31 December 2010

	Effective since	CAN	EUR	FRA	JPN	CHN	KOR	ZAF	RUS	CHE	USA
VHTR SA		X	X	X	X	X	X			X	X
HP PA	19-Mar-08	X	X	X	X		X			O	X
FFC PA	30-Jan-08	O	X	X	X		X				X
MAT Project	30-Apr-10	X	X	X	X	O	X	X		X	X
CMVB Project			P		P	P	P	P			P
SFR SA			X	X	X	X	X		X		X
AF PA	21-Mar-07		X	X	X		X				X
GACID PA	27-Sep-07			X	X						X
CDBOP PA	11-Oct-07			X	X		X				X
SO PA	11-Jun-09			X	X		X				X
SIA Project			P	P	P		P				P
SCWR SA		X	X		X						
M&C Project	6-Dec-10	X	X		X		O				
TH&S PA	5-Oct-09	X	X		X	O	O				
SIA Project		P	P		P	O	O				
FQ Project		P	P		P						
GFR SA			X	X	X					X	
CD&S PA	17-Dec-09		X	X						X	
FCM Project			P	P	P					P	
LFR System			P		P						O
MSR System			P	P					O		O

X = Signatory P = Provisional participant O = Observer

Project Acronyms

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

R&D activities within GIF are carried out at the project level and involve all sectors of the research community, including universities, governmental and non-governmental laboratories as well as industry, from interested GIF and non-GIF members. Indeed, beyond the formal and provisional R&D collaborations shown in Table 2-2, many institutes and laboratories cooperate with GIF projects through exchange of information and results, as indicated in Chapter 3.

3.1 Systems

The main results obtained for each of the six systems selected by GIF members for further R&D are provided in the following sections. Although the focus is on collaborative work pursued in 2010, a brief overview of the characteristics of each system is given as background for putting the R&D undertaken in perspective. Relevant key outcomes from research programs pursued by GIF members outside of the GIF collaborative framework are described, especially for systems which had not yet an established/signed System Arrangement in 2010.

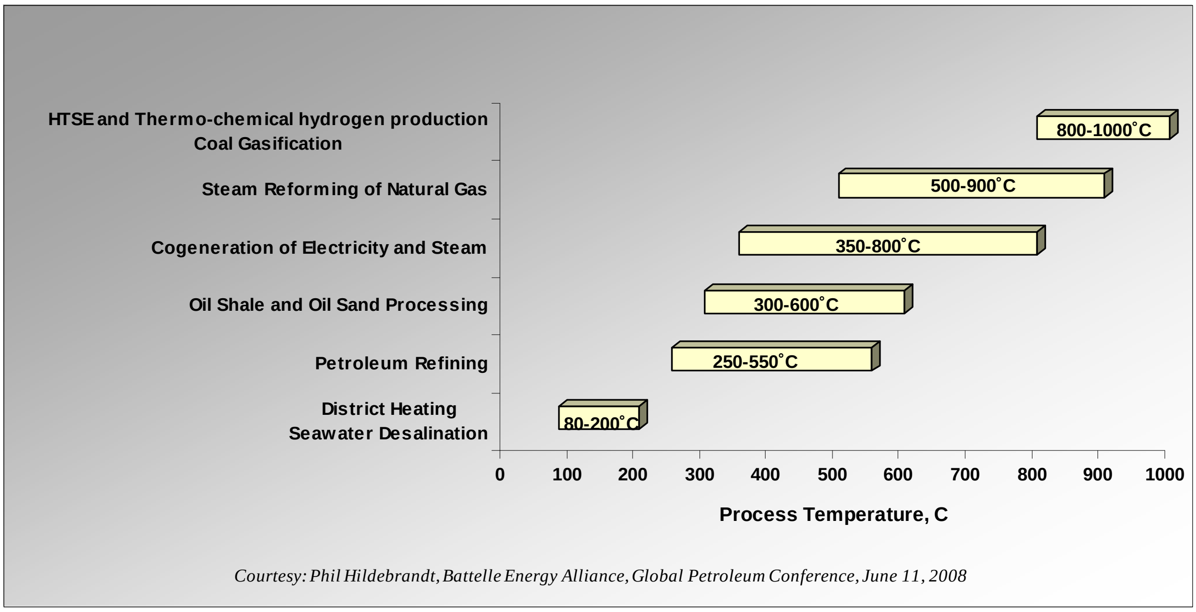
3.1.1 Very-high-temperature Reactor (VHTR)

Main characteristics of the system

The very-high-temperature reactors are the descendants of the high temperature reactors developed in the 1970s-1980s. They are characterized by a fully ceramic coated particle fuel, the use of graphite as neutron moderators, and of helium as coolant.

Use of helium as coolant allows operation at temperature at core outlet as high as 1 000°C, allowing for hydrogen production using processes with no green-house gas emission, such as thermochemical cycles (Iodine Sulfur) or high temperature steam electrolysis (HTSE). Beyond electricity generation and hydrogen production, high temperature also allows for use in other industries, substituting fossil fuel facilities to provide heat to industrial processes (Figure 3-1).

Figure 3-1: Industrial Applications vs. Temperatures



As previously noted, the basic technology for the VHTR has been established in former high-temperature gas reactors such as the US Peach Bottom and Fort Saint-Vrain plants as well as the german AVR and THTR prototypes. These reactors represent the two baseline concepts for the VHTR core: the prismatic block-type and the pebble bed-type. The fuel cycle will initially be once-through with low-enriched uranium fuel and very high fuel burn-up. Solutions need to be developed to adequately manage the back-end of the fuel cycle and the potential for a closed fuel cycle also needs to be fully established. Although various fuel designs are considered within the VHTR systems, all concepts exhibit extensive similarities allowing for a coherent R&D approach, as the coated-particle fuel form is the common denominator for all. This fuel consists of small particles of nuclear material, surrounded by porous carbon buffer, and coated with three layers: pyro-carbon/silicon carbide/pyro-carbon. This coating represents the first barrier against fission products release.

Former reactors were operated at temperature lower than 800°C (high temperature reactors). The available high-temperature alloys used for heat exchangers and metallic components determine the current temperature range of VHTR (~800-950°C). The final target for GIF VHTR has been set at 1 000°C or above, which imply the development of innovative materials like new super alloys, ceramics and compounds. This is especially needed for some non electric applications, where this very high temperature at the core outlet is required to fulfill VHTR mission of providing industry with very high temperature process heat.

In the current projects of VHTR, the electric power conversion unit is an indirect Rankine cycle applying the latest technology of conventional power plants, as this technology is available. However, direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles are perceived as longer term options.

Experimental reactors HTTR (Japan, 30 MWth) and HTR-10 (China, 10 MWth) support the advanced reactor concept development for VHTR. They provide important information for the demonstration and analysis of safety and operational features of VHTRs. This allows improving the analytical tools for the design and licensing of commercial-size demonstration VHTRs. The HTTR, in particular, will provide a platform for coupling advanced hydrogen production technologies with a nuclear heat source at a temperature level up to 950°C.

The technology is being advanced through near and medium-term projects, such as HTR-PM, NGNP, GT-MHR, NHDD, and GTHTR300C, led by several plant vendors and national laboratories respectively in the People's Republic of China, United States, Republic of Korea and Japan. The construction of a two-module HTR with pebble bed core (HTR-PM) has been started in China (Figure 3-2). Each module will have a power of 250 MWth. The coolant gas temperature will be 750°C, which represents the current state-of-the-art for materials. High quality steam of 566°C will be fed into a common steam header.

Status of cooperation

The VHTR system arrangement (SA) was signed in November 2006 by Canada, Euratom, France, Japan, the Republic of Korea, Switzerland and the United States. In October 2008, the People's Republic of China formally signed the VHTR SA during the policy group meeting held in Beijing. The Republic of South Africa, which has expressed high interest in the VHTR, formally acceded to the GIF framework agreement in 2008, and is expected to sign the VHTR SA.

The fuel and fuel cycle project arrangement (PA) became effective on 30 January 2008, with implementing agents from Euratom, France, Japan, the Republic of Korea and the United States.

The materials PA, which addresses graphite, metals, ceramics and composites, was signed by implementing agents from Canada, Euratom, France, Japan, Republic of Korea, South Africa,[5] Switzerland and the United States by 16 September 2009, and is effective since 30 April 2010. China initiated the process for joining the PMB in 2010.

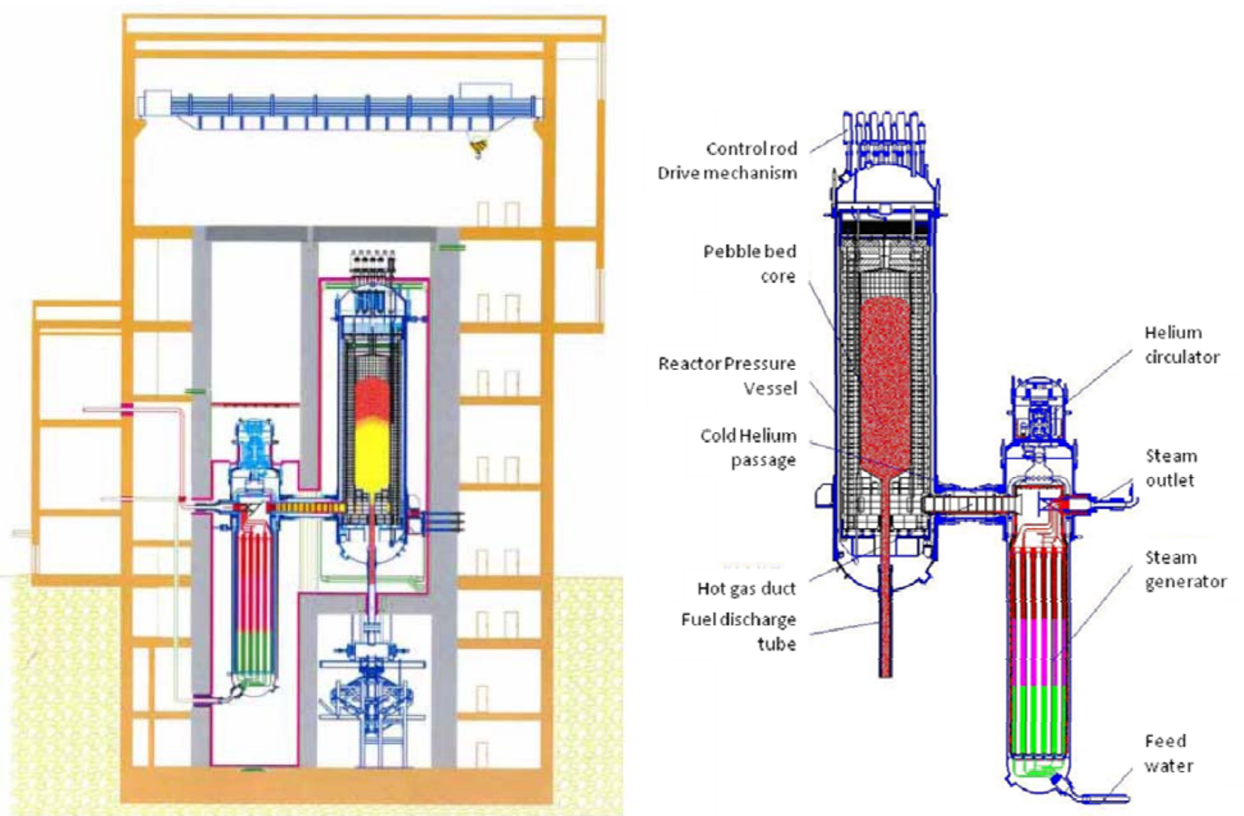
5. The provisions of the GIF framework agreement, under Article V, allow a SSC to approve other entities from the public or private sectors to be signatories to a PA subject to the unanimous approval of the SSC. Accordingly, the SSC voted unanimously on 2 October 2008, to approve direct participation of PBMR Pty Ltd in the materials PA.

The hydrogen production PA became effective on 19 March 2008 with implementing agents from Canada, Euratom, France, Japan, the Republic of Korea and the United States. In 2010, China expressed its wish to join the PMB.

The computational methods, validation and benchmarking PA will be finalized soon, and should be ready for signature in the near future.

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

Figure 3-2: HTR-PM Reactor Building/Primary Circuit



R&D objectives

Even if VHTR development is mainly driven by the achievement of very-high-temperatures providing higher thermal efficiency for new applications, other important topics are driving forces to current R&D: demonstration of reliable inherent safety features, high fuel burn-up (150-200 GWd/tHM) and "very" long operational lifetime (more than 60 years), with potential for conflicts among those challenging R&D goals.

The VHTR system research plan (SRP) describes the research and development program to establish the basic technology of the VHTR system. As such, it is intended to cover the needs of the viability and performance phases of the development plan described in the Generation IV technology roadmap. While the SRP is structured into six projects; only three projects are now effective, and one should be soon ready for signature by members:

1. Fuel and fuel cycle (FFC) investigations are focusing on the performance of the TRISO coated particles, which are the basic fuel concept for the VHTR. R&D aims to increase the understanding of

standard design (UO_2 kernels with SiC/PyC coating) and examine the use of uranium-oxicarbide 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 involves fuel characterization, post irradiation examination, safety testing, fission product release evaluation, as well as assessment of chemical and thermo-mechanical materials properties in representative service and accident conditions. R&D also examines spent-fuel treatment and disposal, including used-graphite management, as well as the deep-burn of plutonium and minor actinides (MA) in support of a closed cycle.

2. Materials (MAT) development and qualification, design codes and standards, as well as manufacturing methodologies, are essential for the VHTR system development. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in the operating environments. For core coolant outlet temperatures up to around 900°C, it is envisioned to use existing materials; however, the goal of 1 000°C, including safe operation under off-normal conditions and involving corrosive process fluids, requires the development and qualification of new materials. Improved multi-scale modeling is needed to support inelastic finite element design analyses. Structural materials are considered in three categories: graphite for core structures, fuel matrix, etc.; very/medium-high-temperature metals; and ceramics and composites. A materials handbook is being developed to efficiently manage VHTR data, facilitate international R&D coordination and support modeling to predict damage and lifetime assessment.

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

4. Computational methods validation and benchmarks (CMVB) 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, upset and accident conditions. Code validation will be carried out through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported by HTTR[6] and HTR-10 tests or by past high-temperature reactor data (e.g. AVR, THTR and Fort Saint-Vrain). Improved computational methods will also facilitate the elimination of unnecessary design conservatisms and improve construction cost estimates.

Despite it is not currently implemented, components development needs to be addressed for the key reactor systems (core structures, absorber rods, core barrel, pressure vessel, etc.) and for the energy conversion or coupling processes like steam generators, heat exchangers, hot ducts, valves, instrumentation and turbomachinery. Some components will require advances in manufacturing and on-site construction techniques, including new welding and post-weld heat treatment techniques. Such components will also need to be tested in dedicated large scale helium test loops, capable of simulating normal and off-normal events. The project on components should address development needs that are in part common to those of the gas-cooled fast reactor (GFR), so that common R&D could be envisioned for specific requirements, when identified.

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6. HTTR: high-temperature test reactor/HTR-10: high temperature gas-cooled reactor 10 MW/AVR: arbeitsgemeinschaft versuchsreakto /THTR: thorium high-temperature nuclear reactor.

Design, safety and system integration is necessary to guide the R&D to meet the needs of different VHTR baseline concepts and new applications such as cogeneration and hydrogen production. Near- and medium-term projects should provide information on their designs to identify potentials for further technology and economic improvements. At the moment, this topic is directly addressed by the system steering committee.

Milestones

The major milestones defined in VHTR system research plan are:

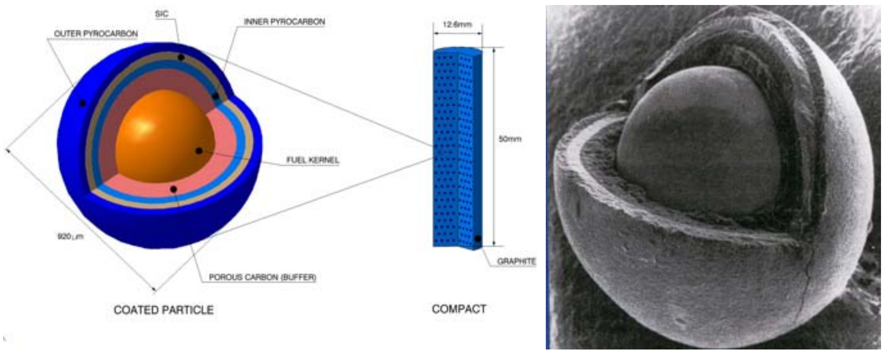
- Viability stage/preliminary design and safety analysis: 2010.
- Performance stage/final design and safety analysis: 2015.
- Demonstration stage/construction and preliminary testing: 2020.

Main activities and outcomes

Fuel & fuel cycle

The main achievement in 2010 is the succesful operation of HTTR, in Japan, for 50 days at core outlet temperature of 950°C. No fuel failure has been detected during this period, and only very low fission products content was measured in the coolant. This result confirm the very high quality of the TRISO fuel.

Figure 3-3: VHTR Fuel – TRISO Particle



Several irradiation programs are ongoing. In particular, the AGR-2 initiated irradiation in the advanced test reactor in June 2010. This program includes fuel provided by US, France and South Africa. The HFR-EU1 irradiation, which consist of 3 GLE4 pebbles produced for AVR and 2 pebbles produced by INET, has been completed in 2010. The European irradiations PYCASSO-I and –II (PYrocarbon irradiation for creep and shrinkage/swelling on objects) are now complete, with very good performance observed. Post irradiation examinations (PIE) are planned under the new 7th EU framework program ARCHER program, follow-up of the RAPHAEL program (6th EU FP), with X-ray tomography as key feature. PMB members also contributed to the round robin test of characterization of ZrO2 surrogate kernel coated particle samples organized under the IAEA CRP6 (advances in HTGR fuel technology), in particular with a benchmark of quality control techniques applied to samples supplied by USA, Korea and South Africa. The contribution also included extensive normal and accident condition benchmark for TRISO fuel performance models. Work on the back end of the fuel cycle and the transmutation potential of VHTRs continues in the EU (CARBOWASTE, PUMA, RAPHAEL)[7] and the US (Deep-Burn).

7. CARBOWASTE: treatment and disposal of irradiated graphite and other carbonaceous waste/PUMA: plutonium and minor actinides management by gas-cooled reactors/RAPHAEL: reactor for process heat, hydrogen and electricity generation.

Materials

The important date for the PMB in 2010 was 30 April, when it entered officially into force. As of 2010, this project arrangement, dealing with materials for the helium cooled systems which share most of their requirements except for core material (VHTR and GFR), has been joined by seven signatories of the framework arrangement (Canada, Euratom, France, Japan, Korea, Switzerland and United States), plus one representative of an eigth member (PBMR from South Africa) as agreed by the system steering committee. In 2010, China also initiated the process of joining the project arrangement, with offers for input to graphite and to metals activities.

In 2010, activities focused on near and medium term projects needs, i.e. graphite and high temperature metallic alloys. Characterization of selected graphite grades are performed and shared by the different members. Graphite irradiations are complete [INNOGRAPH program under RAPHAEL (6^{th} EU FP)], ongoing (AGC1 program in the advanced test reactor, in United States), or planned [low dose irradiation under the ARCHER project (7^{th} EU FP)]. Post irradiation examination of INNOGRAPH samples is planned in 2011, within the ARCHER program. Regarding metallic alloys, main effort is put on high temperature nickel-base alloy and on 9Cr steel, especially on the creep and creep-fatigue behavior in operating conditions, as well as corrosion behavior. In the near/medium term VHTR projects, targeting temperature below 900°C, metallic alloys are considered as main option for control rods, instead of ceramics which are intended for future projects at temperature above 1 000°C. Ceramics are also still of interest as thermal insulation materials and for gas fast reactor fuel cladding. PMB results are shared through the materials handbook, database which already gather all deliverables of the project, but is also intended to collect data resulting from characterization by the members. Discussions concerning format and content (mandatory information required for analysis) of data inputs into the materials handbook are ongoing, in order to enable a smooth transfer of data. Also to be noted, the regular participation of a representative from the American society of mechanical engineers (ASME) at the PMB meetings allows for exchange between code-related research needs and R&D groups. This is expected to help tune the research activities coordinated in the PMB according to current needs.

Hydrogen production

Hydrogen production project arrangement was signed originally by Canada, Euratom, France, Japan, Korea and United States. Since 2010, China is also candidate for joining the PMB, expressing interest especially in the sulfur-iodine thermochemical cycle, and the high temperature steam electrolysis.

Main activities of the PMB are dealing with thermochemical cycles (iodine-sulfur, and copper-chlorine), and with high temperature steam electrolysis. Iodine-sulfur process is mainly driven by Japanese (HTTR-IS) and Korean (NHDD) programs. Among results from 2010 about IS process, it is to be noted progress regarding improvement of the different steps involved (electrodialysis for iodine section or new catalyst development for sulfur section for example), as well as in the development of components which could be used in the aggressive operating conditions of the process. Feasibility of high temperature steam electrolysis coupled to nuclear power is now established. However, longevity of components remains one of the key issues associated to this process. In 2010, significant progress has been achieved with lifetime characteristics of cells. In parallel, design and development of components required for implementing the process continue. The third process most studied in 2010 is the copper-chlorine thermochemical cycle, which could be operated at lower temperature, in agreement with the temperature targeted for other Generation IV systems such as SCWR. Significant milestones were reached in 2010, such as demonstration of electrolyser for CuCl/HCl and thermal decomposition of copper-oxychloride.

Computational methods validation and benchmark

Due to the evolution regarding national programs, the list of provisional signatories to the computational methods validation and benchmark (CMVB) PA evolved. Provisional members are now Canada, China, Euratom, Korea, Japan and United States. South Africa could also join the PMB when they sign the system arrangement and the situation of PBMR is clarified.

2010 was devoted to the finalization of the project plan. CMVB project plan objective is to ensure that the numerical models used for reactor system analysis are capable of calculating the reactor system behavior at normal operational conditions and for operational transients and accident scenarios. Computational tools are used in areas such as thermal-hydraulics, structural mechanics, core physics, chemical transport, and may also be used together via some form of coupling to calculate scenarios that require multiphysics solutions. While some numerical models are presently under development (e.g. pebble-bed and prismatic reactor physics), the development of many numerical models is now considered ready for validation. This validation must be accomplished using the classical approaches that have been accepted by the nuclear community over the past few decades. Most of the CMVB R&D is focused on (a) identifying the key phenomena, i.e. performing phenomena identification and ranking studies, (b) identifying the data that may be available within the CMVB member organizations to be used for performing validation calculations, (c) defining the standards that validation data sets must achieve before the data sets may be qualified for use in validation matrices, and (d) performing validation studies using data sets released to CMVB members for that purpose. The experimental data sets range from those describing basic phenomena to integrated experiments including presently operational experiments such as the 30 MWth HTTR, HTR-10 or vintage experiments such as AVR or plant data from Fort Saint-Vrain.

3.1.2 Sodium-cooled Fast Reactor (SFR)

Main characteristics of the system

The sodium-cooled fast reactor 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 SFR closed fuel cycle enables regeneration of fissile fuel and facilitates management of high-level waste, in particular the plutonium and minor actinides. The fast neutron spectrum also greatly extends the uranium resources compared to thermal reactors. However, this requires that recycled fuels be developed and qualified for use. Important safety features of the Generation IV system include a long thermal response time, a reasonable margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the power conversion system. Water/steam and supercritical carbon dioxide are considered as working fluids for the power conversion system to achieve high performance in terms of thermal efficiency, safety and reliability. With innovations to reduce capital cost, the SFR will be economically competitive in future electricity markets.

In addition, the SFR is considered to be the nearest-term deployable system for actinide management. Fast reactors hold a unique role in the actinide management mission because they operate with high energy neutrons that are more effective at fissioning transuranic actinides. The main characteristics of the SFR for actinide management mission are:

- Consumption of minor actinides in a closed fuel cycle, thus reducing the radiotoxicity and heat load which facilitates waste disposal and geologic isolation.
- Enhanced utilization of uranium resources through efficient management of fissile materials and multi-recycle.
- High level of safety achieved through inherent and passive means that accommodate transients and bounding events with significant safety margins.

Plant size options under consideration range from small, 50 to 300 MWe, modular reactors to larger plants up to 1 500 MWe. The outlet temperature is 500-550°C for the different options, which affords the use of the materials developed and proven in prior fast reactor programs.

Also, much of the basic technology for the SFR has been established in former fast reactor programs, and is being confirmed by the Phenix end-of-life tests in France, the restart of Monju in Japan, the lifetime extension of BN-600 and construction of BN-800 in Russia and PFBR[8] in India, and the startup of the China experimental fast reactor.

The reactor unit can be arranged in a pool layout or a compact loop layout. Three options are considered: a large size (600 to 1 500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially minor actinides, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors as shown in Figure 3-4; an intermediate-to-large size (300 to 1 500 MWe) pool-type reactor with oxide or metal fuel reactor as shown in Figure 3-5; and a small size (50 to 150 MWe) modular-type reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor as shown in Figure 3-6. The two primary fuel recycle technology options are (1) advanced aqueous and (2) pyrometallurgical processing. A variety of fuel options are being considered for the SFR, with mixed oxide the lead candidate for advanced aqueous recycle and mixed metal alloy the lead candidate for pyrometallurgical processing.

The SFR is an attractive energy source for nations that desire to make the best use of limited nuclear fuel resources and manage nuclear waste by closing the fuel cycle. If deployed widely, the SFR can reduce the intensity of CO_2 emissions.

Figure 3-4: Loop-configuration SFR

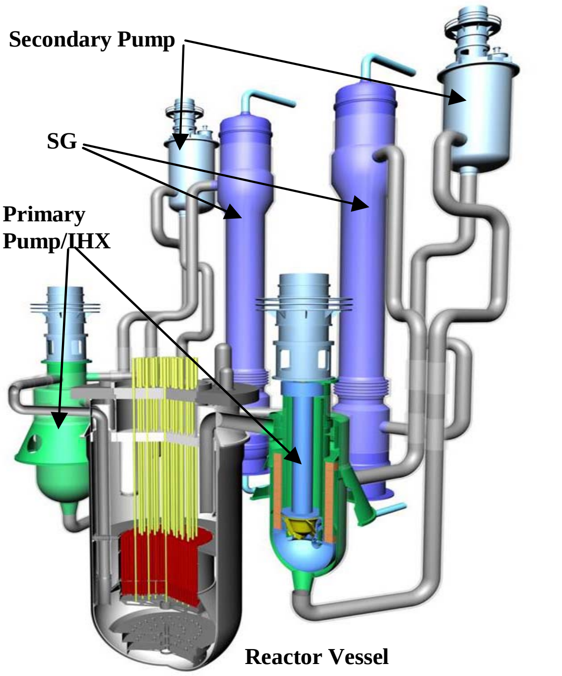
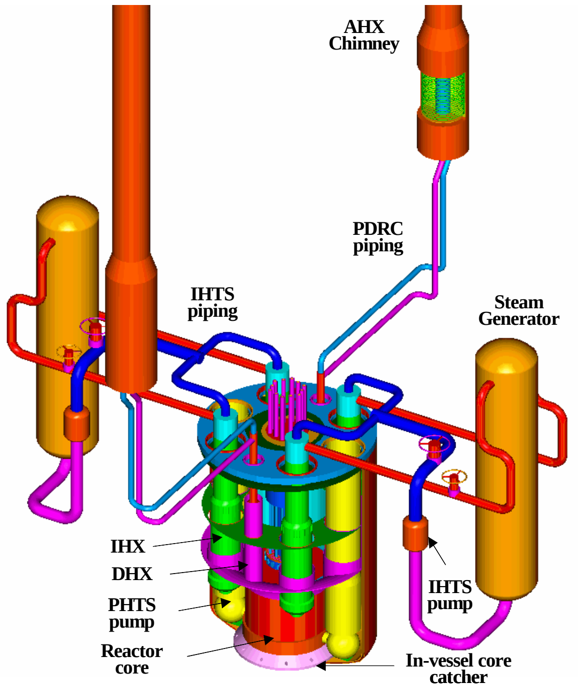
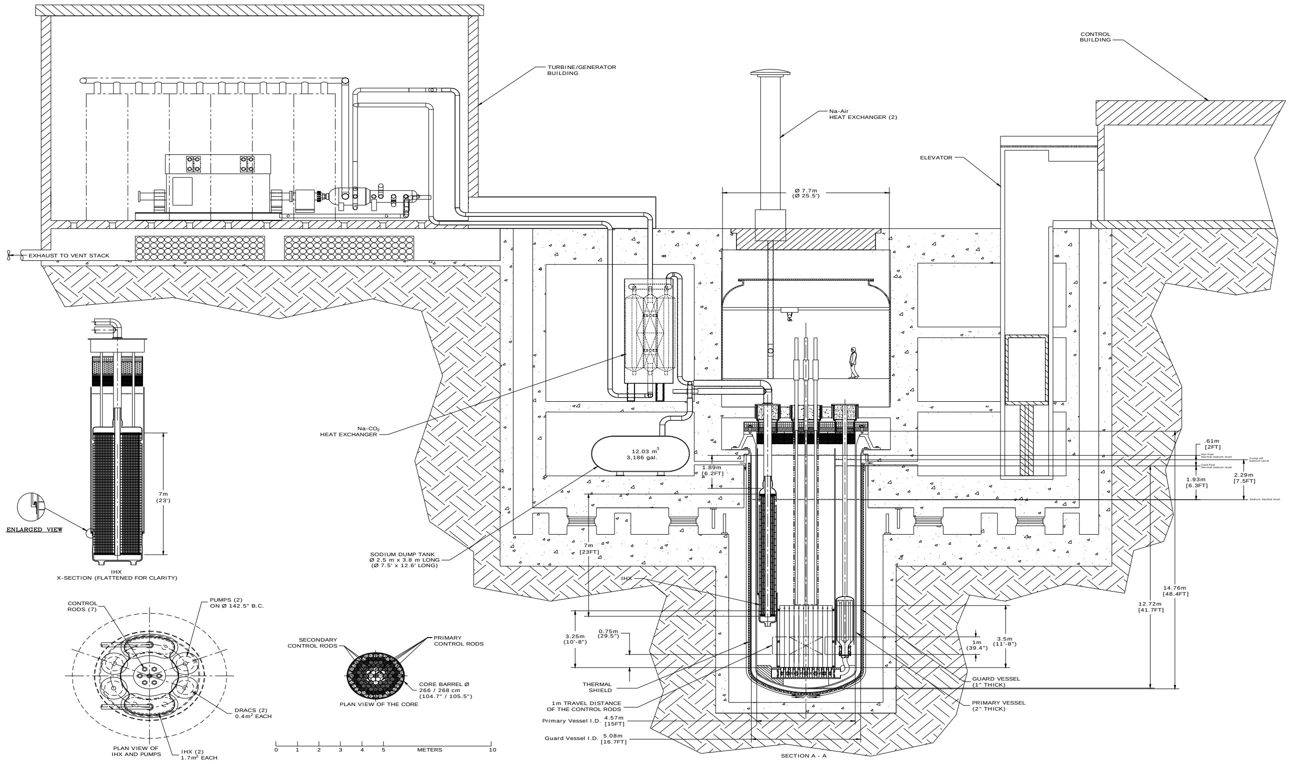


Figure 3-5: Pool-configuration SFR



8. PFBR : prototype fast breeder reactor.

Figure 3-6: Small Modular SFR Configuration



Status of cooperation

The system arrangement (SA) for the international research and development of the SFR nuclear energy system was signed in February 2006. In 2010, Russia signed the SA and the present official members of the SA are:

- The Commissariat à l'énergie atomique of France.
- The Department of Energy of the United States.
- The Joint Research Centre of Euratom.
- The Japan Atomic Energy Agency of Japan.
- The Ministry of Education, Science and Technology of the Republic of Korea.
- The China National Nuclear Corporation of the People's Republic of China.
- The ROSATOM of the Russian Federation.

Three project arrangements were signed in 2007 for advanced fuel, component design and balance-of-plant and global actinide cycle international demonstration. The project arrangement for safety and operation was signed in 2009. The project arrangement for system integration and arrangement is in its final stage awaiting an approval by the GIF policy group.

R&D objectives

The SFR development approach builds on technologies already used for SFRs that have successfully been built and operated in France, Germany, Japan, the Russian Federation, the United Kingdom and the United States. As a benefit 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, the research activities within GIF have been arranged by the SFR SA signatories into five projects. The scope and objectives of the R&D to be carried out in these five projects are summarized below.

System integration and assessment (SIA) project

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

Safety and operation (SO) project

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

Advanced fuel (AF) project

Fuel related research aims at developing high burn-up MA bearing fuels as well as claddings and wrappers withstanding high neutron doses and temperatures. It includes research on remote fuel fabrication techniques for fuels that contain minor actinides and possibly traces of fission products as well as performances under irradiation of fuels, claddings and wrappers. Candidates under consideration are: oxide, metal, nitride and carbide (since 2008) for fuels, and Ferritic/Martensitic & ODS steels for core materials.

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

Research on component design and balance-of-plant covers experimental and analytical evaluation of advanced in-service inspection and repair technologies including leak-before-break assessment, and development of alternative energy conversion systems, e.g., using Brayton cycles. Such a system, if shown to be viable, would reduce the cost for electricity generation significantly as compared with the Rankine steam cycle owing to its compactness. In addition, the significance of the experience that has been gained from SFR operation and upgrading is recognized.

Global actinide cycle international demonstration (GACID) project

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

Milestones

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

- SO Project
  - R&D for safety

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

- R&D for reactor operation and technology testing

  2008-2011 Tasks related to SIA project

  - Phenix end-of-life program
  - Thermal-hydraulics/general system
  - Feedback from the decommissioning of liquid metal fast reactors

  2008-2012 Tasks related to CDBOP project

  - Development of in-service inspection techniques for future SFR drawing from existing reactor experience
  - Sodium chemistry
  - Sodium technology

- AF Project

2006-2007	Preliminary evaluation of advanced fuels
2007-2010	Evaluation of MA-bearing fuels
2011-2015	High-burn-up fuel behavior evaluation
2016-	Demonstration and application of the selected advanced fuel

- CDBOP Project

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

- GACID Project

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

Main activities and outcomes

Safety and operation project

Various safety provisions and systems were investigated and their performances were assessed in order to evaluate whether the design meets the safety requirements. Three different SFR core options, i.e. oxide, carbide, and metal cores, were studied focusing on the role of reactivity feedback coefficients during unprotected accidental transients to avoid sodium boiling, clad and structure damage, and fuel melting. The study revealed a drastic reduction in the sodium void effect is necessary to avoid sodium boiling in the case of an unprotected loss of primary flow when oxide cores are used. It is also found that metal or carbide fuels with Na bonding are able to meet the design limits for both unprotected primary and secondary flow transients. Several new designs of oxide fuel assemblies are being studied to improve inherent performance levels. The mechanism of the feedback reactivity on power operation was also investigated and the effect of irradiation behavior of fresh fuel on power reactivity coefficient was confirmed using the measured data in the experimental fast reactor Joyo.

A detailed design of test loop for the experimental investigation of the performance of decay heat removal circuit was completed. Data of heat transfer and pressure drop characteristics for different types of heat exchangers will be generated. For the investigation of innovative safety provisions against core melting accident consequences, studies on the sacrificial materials for core catcher and fusible shutdown device were carried out. The effectiveness of design measures for the elimination of recriticality by the early discharge of molten fuel and post-accident material redistribution was assessed based on the EAGLE experiment data.

The applicability of system analysis codes such as CATHARE and MARS-LMR for the safety assessment of the advanced SFR designs is being investigated. Regarding the reactor operation and technology testing, studies for the validation of computational tools and the feedback of operational experiences were conducted. The results from Phenix end-of-life asymmetric tests were applied to validate system code models. A refined fuel subassembly thermal-hydraulic model was also validated using the instrumented data from the EBR-II shutdown heat removal test (SHRT) experimental demonstrations.

The experiences for decommissioning of Rapsodie, Phenix and Superphenix were summarized and documented for feedback to the future SFR design. Specifications and feedback for sodium quality control were also analyzed from Phenix and Monju operational experiences. Tritium release or buildup, activated corrosion products and associated dosimetry, and transportation of radioactive elements in the cooling circuits were assessed.

Advanced fuel project

Fuels under consideration are mixed uranium-plutonium based fuels: oxide, metal, nitride and carbide (since 2008) as SFR driver fuel with MA incorporation up to a few percent in accordance with the so-called homogeneous MA recycling in nuclear systems. Fuel investigations have been enlarged in 2009 to include the heterogeneous route for MA transmutation, for which MA are concentrated in dedicated fuels located at the core periphery, by the request of SIA project.

A first technical evaluation based on historical experience, knowledge on fast fuel development, as well as specific fuel tests currently being conducted on MA bearing fuels, has pointed out that both oxide and metal fuels emerge as primary options to meet quickly the goals. Regarding core materials, promising candidates are Ferritic/Martensitic and ODS steels. In the year 2010, fuel behavior of MOX and MA bearing oxide fuel was analyzed by using the code CEDAR-3. Thermo-mechanical behavior of carbide fuel of plate type was evaluated for SFR core design. Irradiation and post-irradiation examinations of MA bearing metal, oxide, nitride and carbide fuels were continued. $(U,Am)O_{2-x}$ targets and metal fuel slugs were fabricated and characterized. Research on MA bearing fuel fabrication in hot cell by remote operation was continued. For the cladding development, mechanical properties of Ferritic/Martensitic cladding materials were measured. Ferritic/Martensitic cladding tube fabrication and preparation of fuel pins with ODS cladding for irradiation in Joyo was continued.

Component design and balance-of-plant project

In 2010, work progressed on in-service inspection technologies, summarizing and reporting of repair experience, high temperature leak before break assessment, and S-CO_2 Brayton cycle advanced energy conversion.

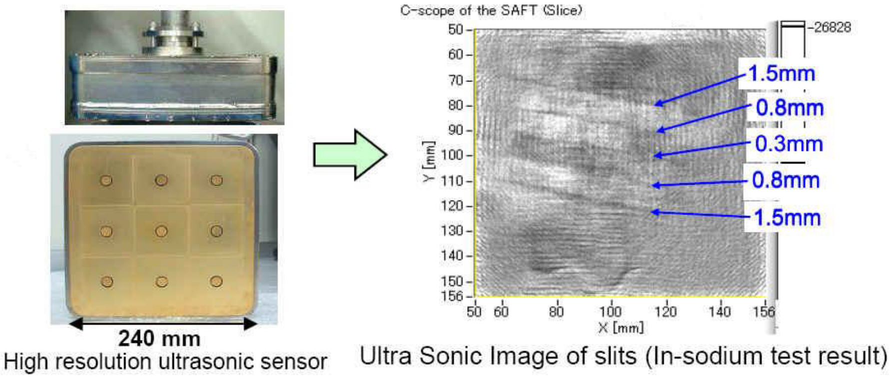
Developments of ultrasonic inspection and control of in-vessel structures from outside of the reactor vessel and the plate type waveguide sensor system continue to achieve better performances. The wetting test of AISI 304L for ultrasonic testing for sensor operability was conducted in addition to the ultrasonic controls of structures immersed in sodium to observe through multiple metal layers. The under sodium viewing sensors using piezoelectric elements (0.5 sec/frame, 2 mm resolution) and using optical diaphragm type sensor (0.3 mm resolution) were tested and good performances were obtained (Figure 3-7). The 10 m long plate type ultrasonic waveguide sensor was tested in both water and sodium environments. The wetting problem of the sensor in sodium remains as one of crucial issues to be addressed.

Significant progress was made in the small-scale demonstration of a Brayton cycle energy conversion system in which supercritical CO_2 is utilized as the working fluid. CEA plant dynamics code, CATHARE, has been developed for transient analysis of ASTRID,[9] with cooperation of ANL. The small

9. Advanced sodium technological reactor for industrial demonstration.

scale demonstration test of S-CO_2 Brayton cycle was continued and plugging due to precipitation of sodium oxide from oxygen contamination of sodium inside of small cooled printed circuit heat exchanger (PCHE) sodium channels was investigated utilizing previous results from tests in ANL small sodium test loop. Comparison of compressor modeling in plant dynamic code and G-PASS/CO2 codes with test data showed good agreement. S-CO_2 compact heat exchanger tests employing the new configuration (aerofoil) of the flow plate were conducted to show lower pressure drop while the heat exchange efficiency was the same as the zigzag configuration. The interactions between sodium and S-CO_2 were studied among CEA, JAEA, and KAERI and the surface reaction formula with temperature ranges was proposed from the test results. As for the potential plugging tests, the cases of piping rupture and small leak from the pipe hole were studied, and the 316FR showed good corrosion resistance between two materials (12 Cr and 316 FR) corrosion tests. Also corrosion behavior of two categories of steels (Ferritic/Martensitic steel and austenitic steel) were tested in S-CO_2 environment and it was found that the austenitic steel 253MA has the most promising corrosion behavior pending further studies.

Figure 3-7: Ultrasonic Viewing Sensor



Global actinide cycle international demonstration project

Activities performed in common by the members included evaluation of MA-bearing fuel material properties, analysis and evaluation of irradiated fuel data, and preliminary program planning for bundle-scale MA-bearing fuel assembly irradiation demonstration in Monju.

3.1.3 Supercritical-water-cooled Reactor (SCWR)

Main characteristics of the system

The supercritical-water-cooled reactors (SCWR) is a high temperature, high pressure water-cooled reactor that operates above the thermodynamic critical point (374°C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be lumped into two main categories: pressure vessel concepts proposed first by Japan and more recently by a Euratom partnership – the European high performance light water reactor (HPLWR); and pressure tube concepts proposed by Canada, generically called CANDU-SCWR. Other than the specifics of the core design, these concepts are taking many common options into consideration (e.g., outlet temperatures, fuel based on UO_2, thermal neutron spectra, steam cycle options, materials, etc.). Therefore, the R&D needs for each reactor type are similar which enables collaborative research to be pursued.

The main advantage of the SCWR is improved economics because of the higher thermodynamic efficiency and the potential for plant simplification. Improvements in the areas of safety, sustainability, and proliferation resistance and physical protection (PR&PP) are also possible and are being pursued by considering several design options using thermal and fast spectra, including the use of advanced fuel cycles.

Status of cooperation

There are currently four project management boards (PMBs) within the SCWR system: 1) system integration and assessment (provisional), 2) materials and chemistry, 3) thermal-hydraulics and safety, and 4) fuel qualification testing (provisional). Table 2-2 lists the members and shows the status of these PMBs.

R&D objectives

The following critical-path R&D projects have been identified in the SCWR system research plan:

- System integration and assessment – definition of a reference design, based on the pressure tube and pressure vessel designs, that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance.
- Thermal-hydraulics and safety – significant gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed. The design-basis accidents for a SCWR have some similarities with conventional water reactors, but the difference in thermal-hydraulic behavior and large changes in fluid properties around the critical point compared to water at lower temperatures and pressures need to be better understood.
- Materials and chemistry – selection of key materials for use in in-core and out-core components of both pressure tube and pressure vessel designs. Selection of a reference water chemistry which minimizes materials degradation and corrosion product transport will also be sought based on materials compatibility and radiolysis.
- Fuel qualification test – an important collaborative R&D project is to design and construct an in-reactor fuel test loop to qualify the reference fuel design. As a SCWR has never been operated before, such generic testing is considered to be mandatory before a prototype reactor can be licensed.

Milestones

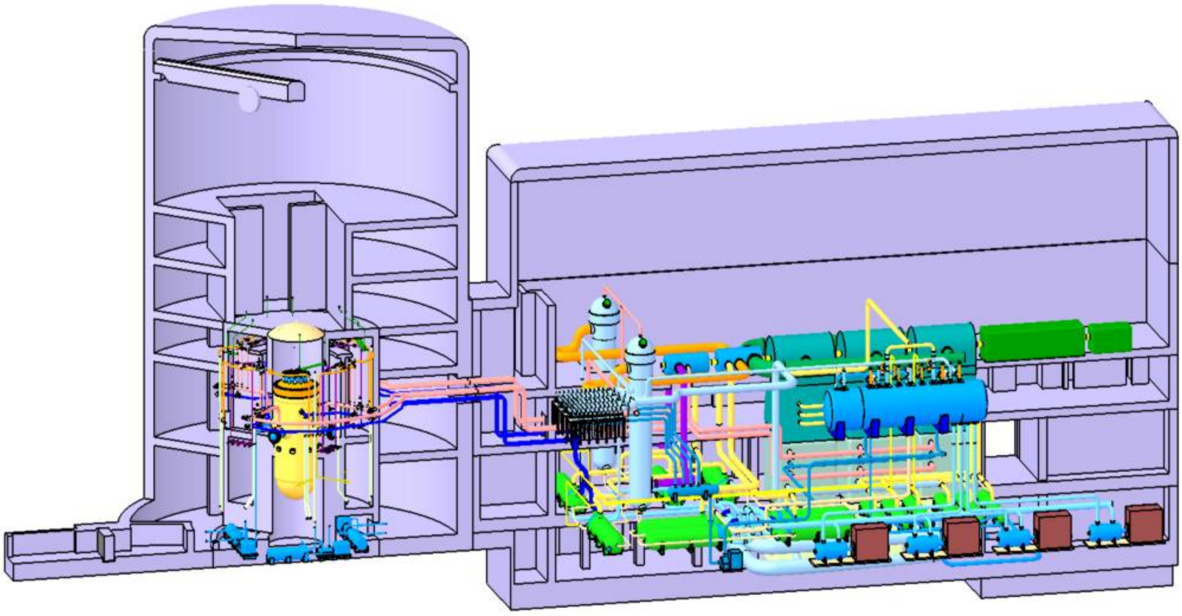
2009	Signing of the thermal-hydraulics and safety project arrangement
2010	Signing of the material and chemistry project arrangement
2010	Assessment of the HPLWR concept with respect to Generation IV criteria
2010	Assessment of the JSCWR (Japanese SCWR) concept with respect to Generation IV criteria
2011	Signing of the fuel qualification testing project arrangement
2011	Completion of the round-robin material tests
2011	Providing water chemistry specification to support long-term testing of candidate materials
2012	Out-of-pile 4-rod bundle sub-assembly testing for thermal-hydraulic validation
2015	In-pile 4-rod bundle sub-assembly testing for fuel qualification
2020	Essential R&D work completed
2020s	Construction and operation of a prototype reactor (maybe outside the GIF activities)
2030s	Construction and operation of commercial SCWR plants (outside the GIF activities)

Main activities and outcomes

System integration and assessment (Canada, EU, Japan)

In Europe, the conceptual design of the HPLWR has been completed and assessed with respect to the criteria of the Generation IV International Forum. This particular SCWR concept features a thermal reactor core with a thermal power of 2 300 MW, a core outlet temperature of 500°C and a supercritical core inlet pressure of 25 MPa. Key design features are a coolant heat-up in 3 steps with intermediate coolant mixing to minimize peak cladding temperature below 650°C. The compact containment design with only 25 m inner height includes automatic depressurization systems, a pressure suppression pool, as well as 4 redundant high and low pressure coolant injection systems with residual heat removal. The once though steam cycle produces a net electric power of 1 000 MW, resulting in a net efficiency of 43.5%. Figure 3-8 illustrates the plant layout with its reactor and turbine building.

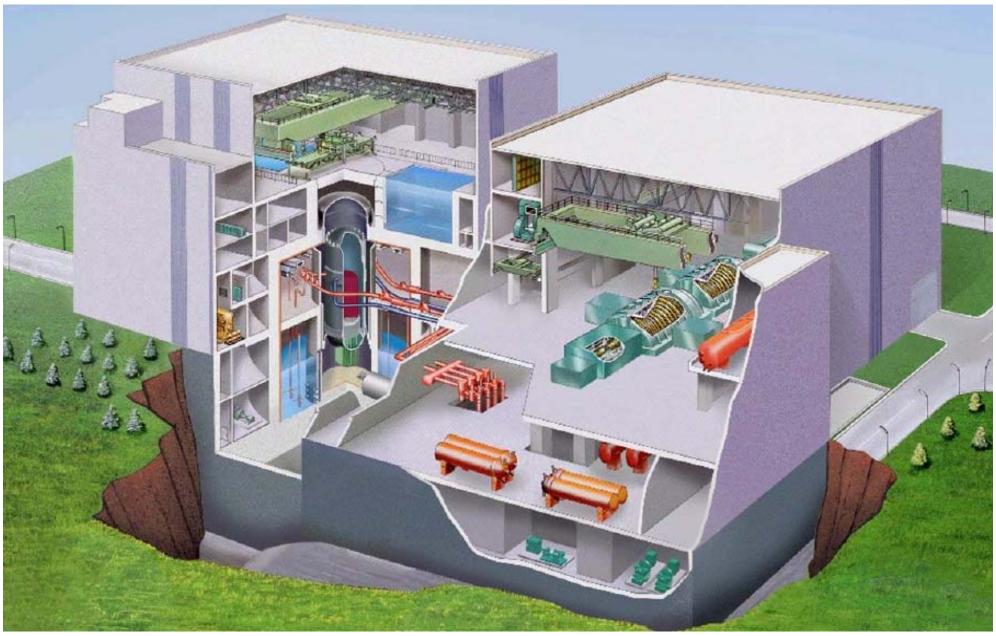
Figure 3-8: Layout of the HPLWR Nuclear Power Plant with 1 000 MW Net Electric Power



The overall assessment of this concept confirmed that the reactor core will meet the design criteria of maximum cladding temperatures and maximum fuel temperatures at design conditions, even including uncertainties and allowances for operation. Safety analyses performed do not give any indication that the core melting frequency could be higher than in current LWRs due to the intrinsic characteristics of this concept. The expected economic advantages are obvious in the plant erection costs, being around 20% cheaper than current LWR, whereas the expected fuel costs do not show significant cost savings yet. Assessment of the proliferation resistance shows at least no disadvantage compared with current LWR. The thermal design concept is not considered for a sustainable fuel cycle, but recycled MOX fuel could be used optionally.

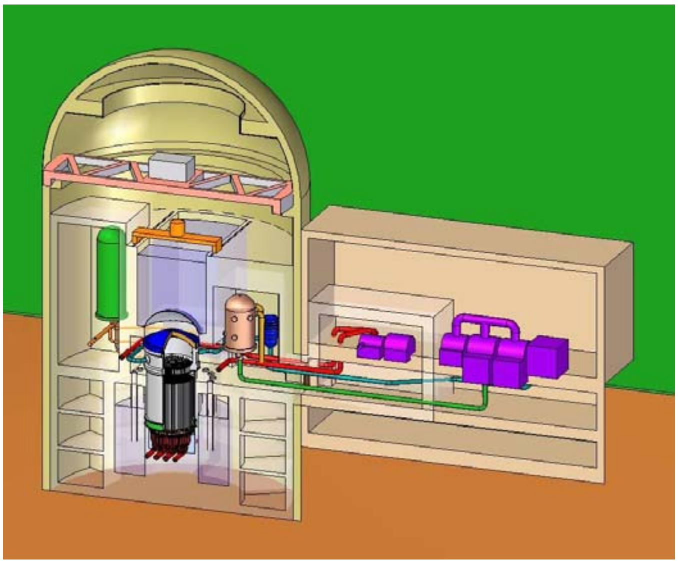
In Japan, the concept of the Japanese supercritical-water-cooled reactor (JSCWR) has been developed and assessed with respect to the criteria of the Generation IV International Forum. The electrical output is assumed to range from 600 MW to 1 700 MW to fulfill user's requirements, and a reference value of 1 700 MW has been selected. Main characteristics of this concept are a thermal neutron spectrum using light water as moderator and coolant, a pressure vessel type, once through reactor, and a direct Rankine cycle turbine system. Figure 3-9 shows a bird's eye view of the power plant. Detailed core design and safety system analyses confirm the viability of the concept. The expected economic advantages agree reasonably well with those of the European plant concept mentioned above, and the proliferation resistance is expected to be as good. A textbook on super light water reactors and super fast reactors has been issued by Oka *et al.*, summarizing technology and design studies by the University of Tokyo.

Figure 3-9: Bird's Eye View of JSCWR Power Plant



The development of the CANDU-SCWR concept has been focusing on the core design since the balance of plant has been well established for the supercritical water fossil power plants. This concept features a thermal neutron spectrum core and maintains the CANDU modular design with fuel channels separating the light water coolant from the heavy water moderator. Operating conditions (design) are set at the pressure of 25 MPa and the outlet coolant temperature of 625°C, matching closely the advanced high-pressure turbine design currently implemented in the fossil power plant. The high outlet temperature would lead to an estimated thermal efficiency of 48%. Further enhancement of the thermal efficiency can be achieved with the reheat option and co-generation capability. The thermal power of the concept can be varied to meet the need and has been set at 2 540 MWth for reference purpose. This has resulted in the electric power of about 1 200 MWe. The supercritical steam is led directly to the high-pressure turbine, eliminating the need of the steam generators (plant simplification and cost saving). Figure 3-10 illustrates the pre-conceptual plant layout of the CANDU SCWR.

Figure 3-10: Layout of the Pre-Conceptual CANDU-SCWR



A passive moderator cooling system is connected to the low-pressure calandria vessel providing continuous cooling of the moderator during normal operation and postulated accident scenarios. Decay heat from the fuel is transferred through the high efficiency fuel channel to the moderator during the long-term cooling phase of the large-break loss-of-coolant accident scenario. The fuel and clad temperatures can be maintained below the melting point, potentially without the need of operator interference for a relatively extended period.

A fuel matrix of thorium and plutonium is currently being considered for the CANDU-SCWR. Worldwide, thorium is more abundant than uranium and plutonium can be extracted from existing inventories of spent fuel. This would enhance sustainability and proliferation resistance.

Materials and chemistry (Canada, EU, Japan)

The project arrangement “materials and chemistry” has been signed in 2010 and materials testing continued with the goal of collecting the data required to specify key materials, in particular the fuel cladding. There were close interactions in 2010 between the M&C PMB and the newly formed fuel qualification PMB.

New materials test facilities came on-line in 2010, for example, a new SCW loop in JRC IE (Netherlands) that includes a newly developed pneumatic bellows-based loading device for stress corrosion cracking testing, and the SCW in-pile corrosion test loop at UJV (Czech Republic), which is undergoing commissioning, including the connection of the active channel to the loop and raising of the sample area temperature to 550°C (max. design temperature 600°C). Longer term out-of-pile operation will continue in 2011.

The body of corrosion data of various alloy classes in SCW continued to grow, with increased emphasis on longer-term testing. In Japan, long term general corrosion testing (25 MPa, 600˚C, 8 ppm dissolved oxygen, <0.1µS/cm conductivity, target test time 5 000 h) of type 310S (low carbon) and modified type 310S stainless steels (SS) was carried out in 2010. These are the current candidate Japanese for SCWR fuel cladding alloys, as well as being strong candidate fuel cladding alloys for fuel qualification testing. SCW oxidation experiments performed at CIEMAT (Spain) showed lower weight gain for 316 SS and Alloy 600 (0.12 mg/dm^2/day (mdd)) than F/M T91 (1.8 mdd) at 400ºC, 25 MPa, 8 ppm oxygen, while PM2000 showed a weight gain of 0.15 mdd at 500ºC, 25 MPa, 10 ppb oxygen. Oxide dispersion strengthened (ODS) alloys such as PM2000 continue to be of interest; Canada made significant progress in assembling the infrastructure required for the development of ODS alloys, and testing of ODS alloys is also planned in the new SCW loop in JRC IE. Canada presented the results of a large parametric study of the effects of temperature, pressure, water chemistry and surface finish on corrosion of 304 SS in SCW, and work continues in Canada to better understand the mechanisms of corrosion in SCW.

The importance of surface finish, highlighted in the 2009 Annual Report, continued to be an active area of investigation in 2010. For example, investigations by VTT (Finland) and AEKI (Hungary) of the effects of surface finish (cold work) and surface modification by ion implantation (He+ and N+) on corrosion in SCW (650°C, 25 MPa, up to 2 000 hours) suggested that a machined surface has the best corrosion resistance, with the protection lasting up to 3 000 h. Investigation of the use of ceramic or metallic coatings to improve corrosion resistance of key components continues; Canada reported on the development and testing of various coatings, including ceramic coating of P91 and Zircaloy (Zr-2.5%Nb), and the use of NiCrAl(Y) coatings.

The specification of a chemistry control strategy continues to be a main focus, with the two key chemistry issues continuing to be water radiolysis and corrosion product transport. In Canada, both experiment and modeling are being used to develop an improved understanding of water radiolysis in SCW. The existing Monte Carlo model has been benchmarked against a recently released state-of-the-art assessment of all existing sub-critical water radiolysis data. Molecular dynamics simulations were carried out at different densities and temperatures to obtain a detailed picture of the heterogeneous molecular structure of SCW, needed to determine how this structure influences radiation energy deposition and subsequent radiolysis reactions.

On-going work in Canada to determine metal oxides solubilities includes a reassessment of magnetite solubility data and the measurement of the solubility of molybdenum oxide (Mo is a common alloying element in steels). Initial results, modeling fission product transport using strontium salts, show that neutral species are important at moderate concentrations at 350°C; their solubility in SCW is sufficient to allow transport out-of-core. A theoretical study of the effects of small amount of inorganic material on SCWR thermal hydraulics showed that the peak values for isothermal heat capacity and isobaric compressibility changed 20-22% at 658.2K upon increasing the concentration from 0 to 300 ppm; this could have an effect on the location of a deteriorated heat transfer region.

Thermal hydraulics and safety (Canada, EU, Japan)

Thermal hydraulics and safety project management board (PMB) members (Canada, EU and Japan) have been working on tasks identified in the 2010 annual work plan. Progress of each member was presented at PMB meetings held in 2010 June and October. A collaborative task between members has been established to perform benchmarking exercise of thermal-hydraulic tools (such as subchannel and computational fluid dynamic codes) against Freon heat transfer data obtained with a 7-rod bundle (to be contributed by Japan).

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 subcritical conditions. This established characteristic is also required in the analysis of a postulated large-break loss-of-coolant accident event.

In Canada, extensive databases on supercritical heat transfer have been compiled for tubes and bundle subassemblies in water and non-aqueous fluids. The water heat-transfer database for tubes contains over 24 000 data points, and is one of the largest compilations in the community. These data are being applied in developing a look-up table for heat transfer coefficient covering sub- and super-critical conditions. The supercritical heat transfer test with water flow in annuli has been extended for an unheated shroud of increased diameter to 20 mm (from 16 mm in the original test). Heat transfer is more efficient for large shroud diameter than small shroud diameter at the same pressure, mass flux, and heat flux. The impact of shroud diameter on heat transfer is more significant at high mass flux and low heat flux conditions than at low mass flux and high heat flux conditions. The data are ideal for validation of supercritical heat-transfer correlation and established the effect of flow area on supercritical heat transfer.

A heat-transfer test facility has been constructed using carbon dioxide as coolant at supercritical conditions. It is being commissioned using a vertical tube. A 3-rod bundle has been constructed for testing in this facility to provide detailed surface-temperature distributions at supercritical flow conditions. Separate test facilities using water and non-aqueous fluids as coolant are being constructed for heat transfer, stability, and critical-flow experiments.

In Europe, the HPLWR phase 2 project has been completed. A total of seven deliverables have been produced for GIF from the thermal-hydraulics and safety (TH&S) project in 2010. The safety concept initially proposed for the HPLWR has been further developed. In particular, the main safety functions and appropriate strategies for accident control have been identified, and the key parameters for the operation of the systems have been selected. In order to evaluate the feasibility of this still preliminary concept, safety analyses have been performed using a variety of system and coupled codes. The transient analyses performed addressed a variety of initiating events, including anticipated transients as well as accidents.

Although within this work no code could be demonstrated to be capable to simulate all transients, the simulations performed show that for each class of transients at least one of the computational tools used in this project has been adequate for preliminary assessment of the safety concept of the HPLWR. In spite of the limitations by the uncertainties in the validity of certain models, the analyses have shown that the proposed systems can be expected to be capable to provide all the safety functions. In particular, the analyses showed that the safety systems can effectively limit overheating of the core under the most severe conditions, such as loss of coolant accidents and loss of flow transients

In the design of the HPLWR fuel assembly, the fuel rods are arranged in a square lattice and a helical wire-wrap spacer has been adopted to improve mixing between the different sub-channels. The thermal-hydraulics and heat transfer characteristics in the HPLWR fuel assembly have been studied within the TH&S project using computational fluid dynamics (CFD). CFD models and guidelines have been developed, validated and applied to derive a suitable heat transfer correlation for the HPLWR fuel assembly. This correlation takes into account the effect of the wire-wrap spacer and the effect of the geometry of the HPLWR fuel assembly.

In Japan, thermal-hydraulics tests at supercritical pressure conditions with water and Freon have been done to obtain heat transfer, using tube and bundle. Japan has been conducted supercritical Freon tests for more than 10 years and have been conducting supercritical-pressure water tests for 5 years.

To provide high-precision heat transfer and hydraulics resistance correlations of supercritical water, which are necessary for the conceptual design of the SCWR core and fuel, database was constructed from literature survey and previous research results. Aforementioned Japanese test data and other data collected from literatures and through GIF collaboration have been utilized to establish the database and correlations. The most suitable correlation applied for circular tubes was selected based on the database and the range of application and predictive accuracy were defined.

A thermal-hydraulics analysis code based on large eddy simulation (LES) has also been developed to give detailed information of thermal-hydraulics phenomena of supercritical water in a fuel bundle.

Fuel qualification test

A fuel qualification test facility, required for licensing of a nuclear facility operated with supercritical water, is planned to be installed in the LVR-15 research reactor in Rez, Czech Republic. The provisional project management board has been working in 2010 on a joint project arrangement. The fuel qualification test is planned under evaporator conditions and is addressed with the following concept. A pressure tube with 57 mm outer diameter and 9 mm wall thickness is placed instead of a fuel assembly in the LVR-15 reactor. It contains 4 fuel rods with 8 mm diameter and 9.44 mm pitch, like the HPLWR assembly concept, inside a square assembly box. The rod length is limited to 600 mm to match with the core height of the research reactor. With a ^{235}U enrichment of almost 20%, these 4 fuel rods can reach a fissile power of more than 50 kW. A recuperator inside the pressure tube, situated above the fuel rods, is boosting the feed-water temperature of 300°C to typical evaporator conditions. A cooler in the top section of the pressure tube acts as the heat sink to remove the fissile and gamma power, and thus keeps the coolant temperature at the outlet of the pressure tube below the inlet temperature. A single recirculation pump drives the primary loop running at around 25 MPa system pressure.

During the year 2010, a bilateral, collaborative project between Euratom and China, supporting these activities, has been prepared and submitted for financing to the European Commission and to the Chinese Atomic Energy Agency (CAEA), respectively. European partners will contribute with the research reactor LVR-15, design and analyses and cladding material autoclave tests, while Chinese partners offer an out-of-pile electrically heated pre-qualification facility, which is being commissioned at the Shanghai Jiaotong University. The Chinese partners also offer to share their analyses and material tests with the European partners.

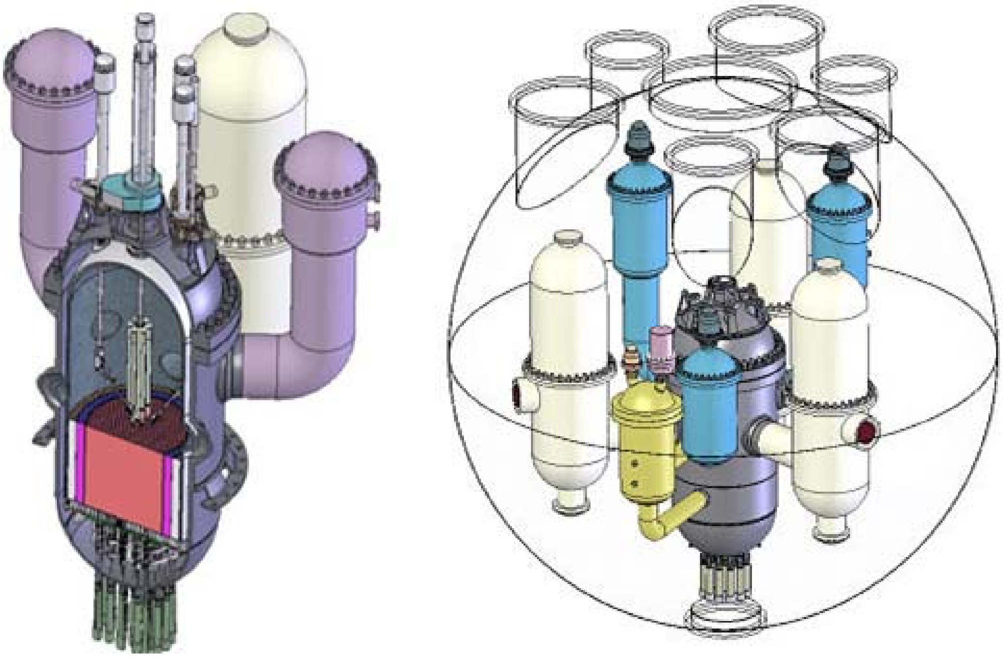
3.1.4 Gas-cooled Fast Reactor (GFR)

Main characteristics of the system

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

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

Figure 3-11: GFR Reference Design

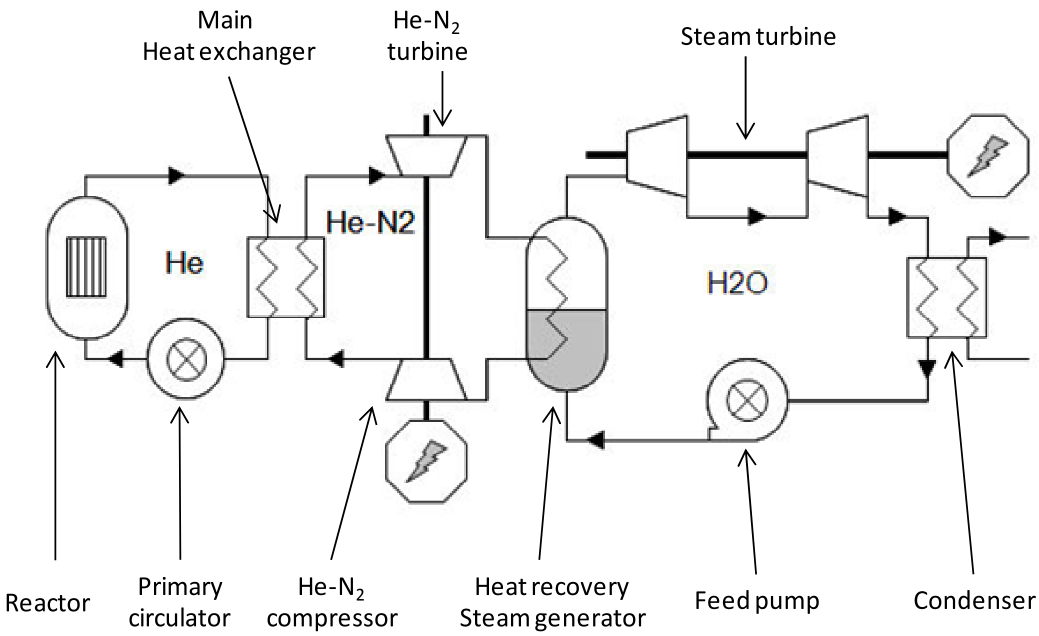


GFR - reactor, decay heat loops, main heat exchangers and fuel handling equipment

GFR – spherical guard vessel

The reference design for GFR is based around a 2 400 MWth reactor core contained within a steel pressure vessel. The core consists of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fuelled pins contained within a ceramic hex-tube. The favored material at the moment for the pin clad and hex-tubes is silicon carbide fiber reinforced silicon carbide. Figure 3-11 shows the reactor core located within its fabricated steel pressure vessel surrounded by main heat exchangers and decay heat removal loops. The whole of the primary circuit is contained within a secondary pressure boundary, the guard containment. The coolant is helium and the core outlet temperature will be of the order of 850°C. Heat exchangers transfer the heat from the primary helium coolant to a secondary gas cycle (Figure 3-12) containing a helium-nitrogen mixture which, in turn drives a closed cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plant so represents an established technology, with the only difference in the GFR case being the use of a closed cycle gas-turbine.

Figure 3-12: GFR Indirect Combined Cycle Power Conversion System



The proposed experimental reactor ALLEGRO (formerly ETDR) could become the first gas cooled fast reactor to be constructed. Being a small experimental reactor (80 MWth), the objectives of ALLEGRO are to demonstrate the viability and to qualify specific GFR technologies such as the fuel, the fuel elements and specific safety systems in particular, the decay heat removal function, together with demonstrating that these features can be integrated successfully into a representative system. So far, ALLEGRO development has been driven by the French national program with significant contributions from Euratom and Switzerland. In 2010 a memorandum of understanding was signed between the Czech Republic, Slovakia and Hungary as partners to support each other in bidding for one of them to host ALLEGRO, with assurances that the two other partners would provide technical and administrative support to the successful host nation.

Status of cooperation

The system arrangement was signed at the end of 2006 by Euratom, France, Japan and Switzerland. It is to be noted that, while France and Japan have been very active in the development of the GFR concept, providing regarding conceptual design, safety assessment and fuel development in the previous years, in 2010 research priorities were re-focused on sodium cooled fast reactors, which led to reducing effort on the GFR system. However, France is also part of the GoFastR project (7th EU framework program), which represent the main contribution from Euratom to the GIF GFR system development.

Two projects were discussed at the origin of the system arrangement, dealing with conceptual design & safety, and fuel and core materials. The conceptual design & safety project arrangement was signed in 2009 by Euratom, France and Switzerland, and is effective as of 17 December 2009. The fuel and core material project arrangement is now ready for signature, and should become effective in 2011.

R&D objectives

As presented above, the GFR system can take advantage of the ongoing R&D within GIF, especially regarding the out-of-core high temperature components and technology. However, there remain some significant technology gaps which demand a more revolutionary approach. These technology gaps are specific to GFR and must be addressed to demonstrate the technical (and commercial) viability of the reactor:

- Fuel forms suitable for simultaneous high temperature and high power density operation with tolerance of fault conditions.
- Development of core materials with superior resistance to fast-neutron fluence under very-high-temperature conditions with good structural, ageing and fission product retention capabilities.
- Core design, achieving a core that is self-sustaining in fissile material but, preferably, without the use of heterogeneous fertile "breeder" blankets to increase proliferation resistance and with the capability to burn minor actinides to improve sustainability.
- Safety systems, including highly reliable decay heat removal systems that must cope with high core power density and the lack of any significant thermal inertia in the core or the coolant provided by the moderator in thermal reactor designs or the liquid metal coolant in other fast reactor systems.
- Fuel cycle technology, including spent-fuel treatment and refabrication for recycling uranium, plutonium and minor actinides.

In this context, the main goals of the conceptual design & safety (CD&S) project are:

- Definition of a GFR reference conceptual design and operating parameters (meeting requirements, already presented in previous reports, on breeding, MA transmutation, Pu mass, efficiency, availability and safety objectives).
- Identification and study of alternative design features (e.g. lower temperatures, pre-stressed concrete pressure vessel, diverse decay heat removal systems).
- Definition of appropriate safety architecture for the reference GFR system and its alternatives.
- Definition of the ALLEGRO conceptual design and its safety architecture, in coherence with that of the GFR.
- Development and validation of computational tools needed to analyze performance and operating transients (design basis accidents and beyond).

The goals of the fuel and core materials (FCM) project are to investigate fuel element design and qualification, material for cladding, and dense fuel material:

- Regarding fuel design, with at least 50% of fissile phase inside the fuel element, pin-type fuel has been finally selected to enhance high power density.
- For clad, standard alloys cannot reach the foreseen temperature. Refractory materials have to be envisaged (metals and ceramic composite), while ODS alloy can be applied for lower temperature GFR core concepts.
- For achieving a high power density and a high temperature, dense fuels with good thermal conductivity are required. Carbide and nitride appear more attractive than oxide. However, oxide is a backup because of extensive experience feedback.

For the development of this innovative fuel element, the R&D activities performed within the FCM project include fuel element design, in-core materials studies (clad materials and fissile phase), fuel fabrication and irradiation program.

Main activities and outcomes

Optimization of core performance and safety characteristics

Designing and optimizing a reactor core is a rather long and complex process as it involves all together neutronics, thermal-hydraulics and fuel thermomechanics. In order to tentatively solve faster such a multiphysics problem, CEA has developed a new approach which aims at optimizing simultaneously core performance parameters (core volume, in-cycle Pu inventory, fuel burn-up etc.) and core safety characteristics (neutronics coefficients, supply pumping power, transient response time etc). Schematically the method is described as follows:

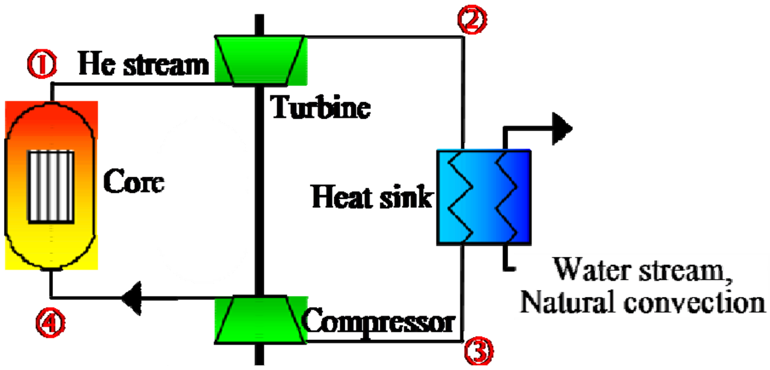
- simplified models are built using reference CEA codes (PARIS for neutronics, METEOR for fuel thermomechanics) and polynomial interpolations derived from physical analytical considerations;
- safety aspects are considered using analytical indicators (decay heat removal by natural convection, thermal inertia of the core, etc…) and a global feedback neutronics coefficient methodology;
- a multi-criteria genetic algorithm explores a large domain of core parameters and searches for improved performance and safety characteristics: for a given set of specifications (thermal power, inlet coolant temperature, He pressure), 10 optimization variables are found sufficient to determine all the core design features (power density, fuel temperature, pin diameter, etc…).

This new methodology leads to less accurate, but quickly optimized, core design features, and proves they are the best that fulfill all the requirements. It has been used to check the reference core design (pin type, He bonded, UPuC, SiCfSiC cladding), and it is presently used to perform the preliminary design of a core concept using an innovative fuel leading to higher performances (increased burn-up, lower Pu inventory).

Decay heat removal – Brayton-cycle-driven system for operation at low pressure

A design study for a Brayton cycle machine has been carried out by PSI. This system would constitute a dedicated, standalone decay heat removal (DHR) device for the Generation IV gas-cooled fast reactor (GFR) (Figure 3-13). In comparison to the DHR reference strategy developed by the French Commissariat à l'Énergie Atomique during the GFR pre-conceptual design phase (which was completed at the end of 2007), the salient feature of this alternative device would be to combine the energetic autonomy of the natural convection process – which is foreseen for operation at high and medium pressures – with the efficiency of the forced convection process which is foreseen for operation down to very low pressures.

Figure 3-13: Schematic of the Dedicated Brayton Cycle for DHR

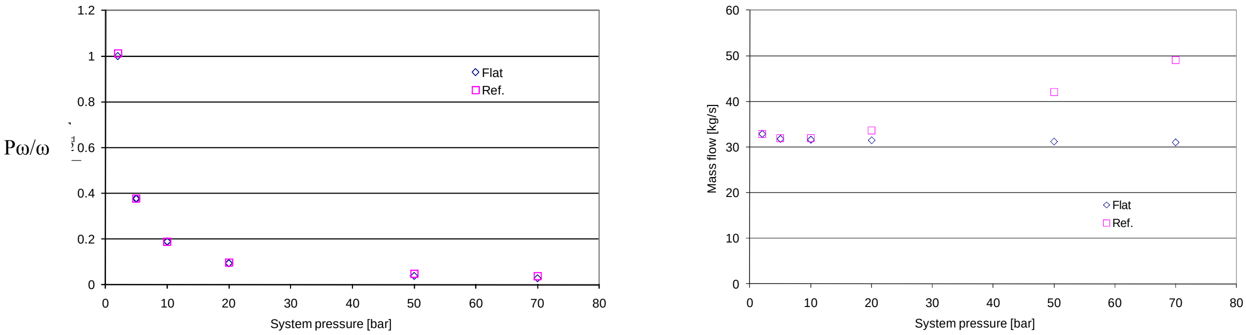


An analytical model based on simplified thermodynamic and aerodynamic equations was developed to highlight design choices. Two different machine designs are analyzed: a Brayton-loop turbo-machine working with helium, and a second one working with nitrogen, since nitrogen is the heavy gas foreseen to be injected into the primary system to enhance the natural convection under loss-of-coolant-accident (LOCA) conditions.

Simulations of the steady-state and transient behavior of the proposed device have then been carried out using the CATHARE code. These serve to confirm the insights obtained from usage of the "Brayton scoping" model, e.g. that the turbo-machine conveniently accelerates during the depressurization process to tend towards a steady rotational speed value, the speed rise being inversely proportional to the experienced pressure drop as shown in Figure 3-14. In addition, it has been found that the mass flow provided by the turbo-machine stays rather constant for a wide range of pressure.

Figure 3-14: Impact of a Pressure Change on Brayton-cycle

(helium case, Flat: without natural convection, Ref: including natural convection)



Steady-state turbo-machine speed

Steady-state mass flow rate

Finally, CATHARE simulations have been performed for complete DHR scenarios for the GFR, involving loss-of-coolant-accidents (LOCAs) in conjunction with loss of back-up-pressure (LOBP). Thereby, it has been shown that, in each of the investigated cases, incorporation of the Brayton-loop turbo-machine with nitrogen indeed leads to fuel temperatures remaining considerably below Category 4 accident limits of 1 600°C.

As an example, a slow depressurization event is investigated here, with a 3 cm LOCA break assumed in the cold duct. The containment break (LOBP) is also assumed to be 3 cm. Table 3-1 shows the event sequence calculated by CATHARE for this case, the LOCA and LOBP occurring simultaneously at the beginning of the transient (t = 2 00 s).

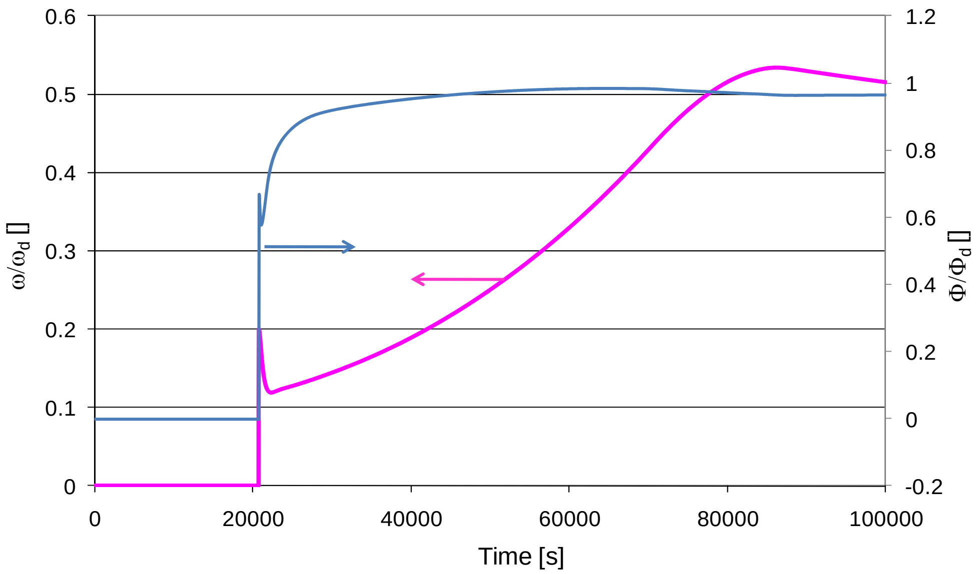
Table 3-1: Event Sequence for the Small-break LOCA and Simultaneous LOBP

Event	Time (s)
Start of transient (LOCA + LOBP)	200
SCRAM + blower trip (upper plenum temp > 107% of nominal value)	491
Main loop closure (cold duct mass flow < 3%)	550
DHR loop opening (main loop closure + 10s)	560
Gas injection from reservoir (vessel pressure <30 bar)	1755
Turbo-machine start up + Brayton loop opening (vessel pressure <10 bar)	20765
Turbo-machine decoupling from motor (start + 60s)	20825

It can be seen (Figure 3-15) that, after the turbo-machine start with the motor at nominal speed, the machine "follows" the depressurization while experiencing a smooth increase of its rotational speed. The

CATHARE simulations do not indicate any surge or stalling of the turbo-machine at any time during the transient.

Figure 3-15: Small-break LOCA + Simultaneous LOBP - Turbo-machine Reduced Rotational Speed and Turbine Flow Factor



Furthermore, it can be seen on Figure 3-16 that the machine start-up with the motor leads to a peak in the mass flow rate, which then stabilizes at about 40 kg/s. This mass flow peak translates into a temporary decrease of fuel temperature, which then goes up again to stabilize below a still convenient value of 1 000°C, i.e. much below the Category 4 accident limit of 1 600°C (Figure 3-17).

Figure 3-16: Small-break LOCA + Simultaneous LOBP - Core and Brayton Loop Mass Flow Rates

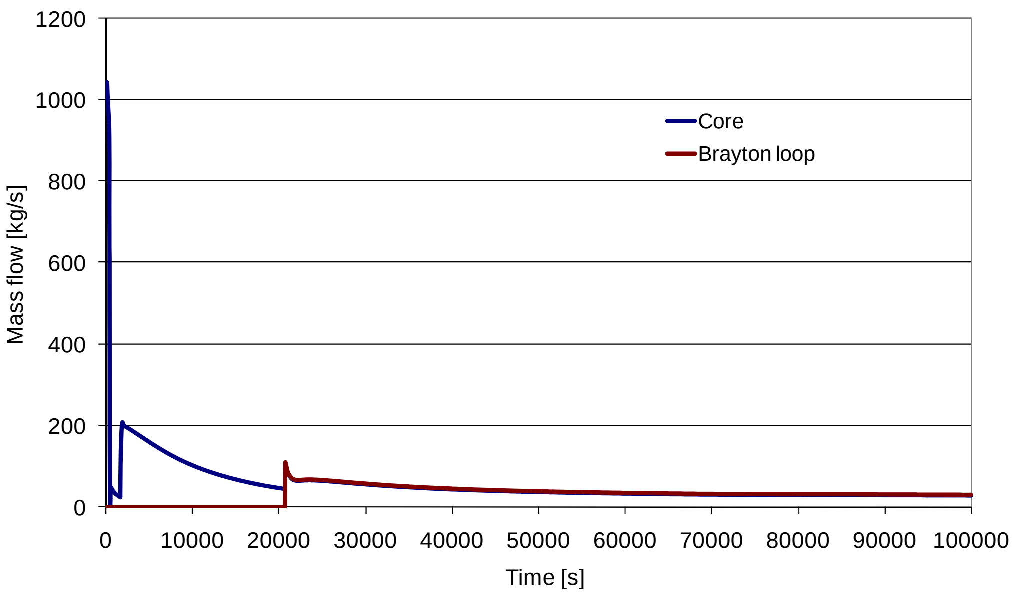
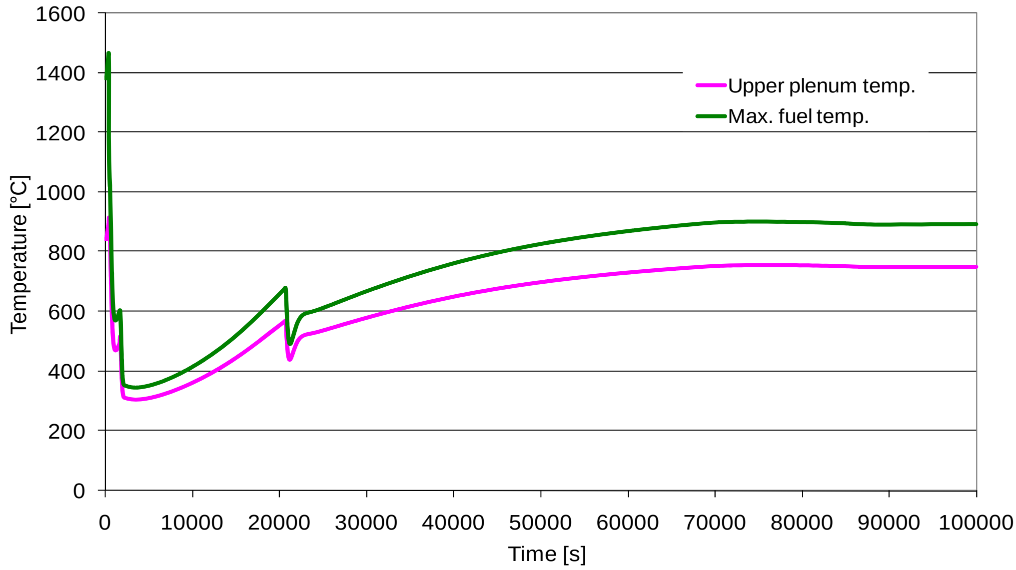


Figure 3-17: Small-break LOCA + Simultaneous LOBP - Upper Plenum and Maximum Fuel Temperatures



The above work has been submitted for publication in *Nuclear Engineering and Design*.

The GoFastR Collaborative Project

The European Union's 7th framework program project GoFastR was launched during March 2010. Twenty-two organizations plus one associate from 11 Euratom member states contribute to this project. The structure of GoFastR echoes that of the GIF GFR System, with work packages 1.1 to 1.5 aligning with those within the GIF GFR CD&S project and the tasks within work package 2 aligning with the work packages of the GIF GFR FCM project. Additional work packages with GoFastR are directed towards engaging with TSOs and licensing authorities (WP7) and managing education and training activities within the project (WP6).

GoFastR is a three-year project with the aim contributing to the establishment of the viability of GFR as a safe, sustainable, economic and proliferation resistant nuclear energy system.

The achievements of the first 9 months of the project have been to establish reference work-horse designs for GFR, ALLEGRO and their associated fuel concepts. The specifications for transient analysis benchmark exercises have been produced. The initial findings of scenario studies about the impact of penetration of GFR into a nuclear park have been prepared.

3.1.5 Lead-cooled Fast Reactor (LFR)

Main characteristics of the system

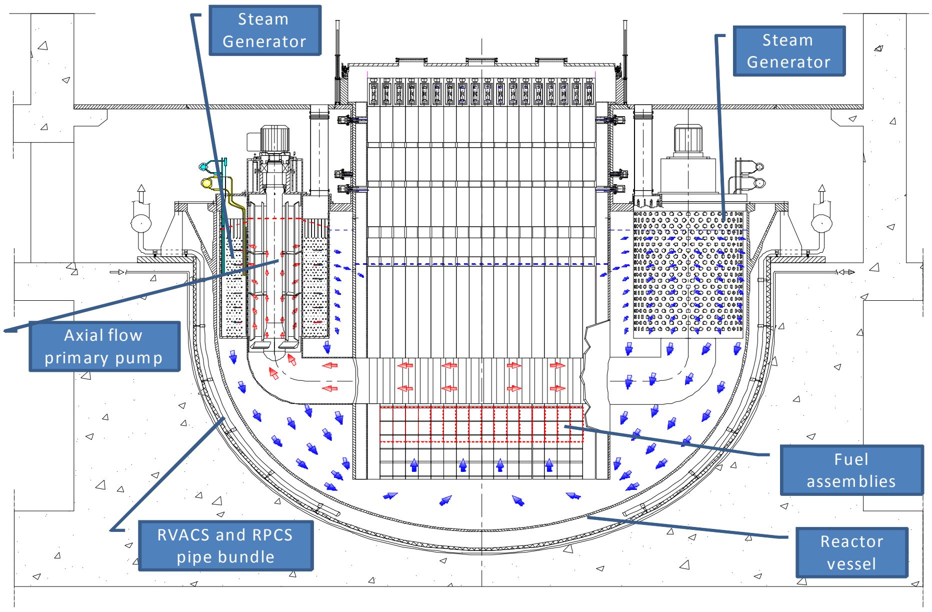
The LFR features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of minor actinides, both self-generated and from reprocessing of spent fuel from light water reactors (LWR), and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of a relatively inert coolant. It has the potential to provide for the electricity needs of remote or isolated sites or to serve as large inter-connected power stations.

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

- the European lead-cooled system (ELSY).
- the small secure transportable autonomous reactor (SSTAR).

The ELSY reference design (Figure 3-18) is a 600 MWe reactor cooled by pure lead (Cinotti, *et al.*, 2008). This concept has been under development since September 2006 within the 6th Euratom framework program. The ELSY project is being performed by a consortium consisting of seventeen organizations from Europe. ELSY aims to demonstrate the possibility of designing a competitive and safe fast critical reactor using simple engineered technical features while fully complying with the mission identified in the GIF Roadmap of minor actinides burning capability.

Figure 3-18: ELSY Configuration



The reference design for the SSTAR is a 20 MWe natural circulation reactor concept with a small transportable reactor vessel (Figure 3-19). 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 nuclear fuel self-sufficiency, autonomous load following, simplicity of operation, reliability, transportability, and a high degree of passive safety. Conversion of the core thermal power into electricity at a high plant efficiency of 44% is accomplished by utilizing a supercritical carbon dioxide Brayton cycle power converter.

Typical design parameters of the SSTAR and ELSY concepts are summarized in Table 3-2.

Figure 3-19: SSTAR Pre-conceptual Design and Operating Parameters

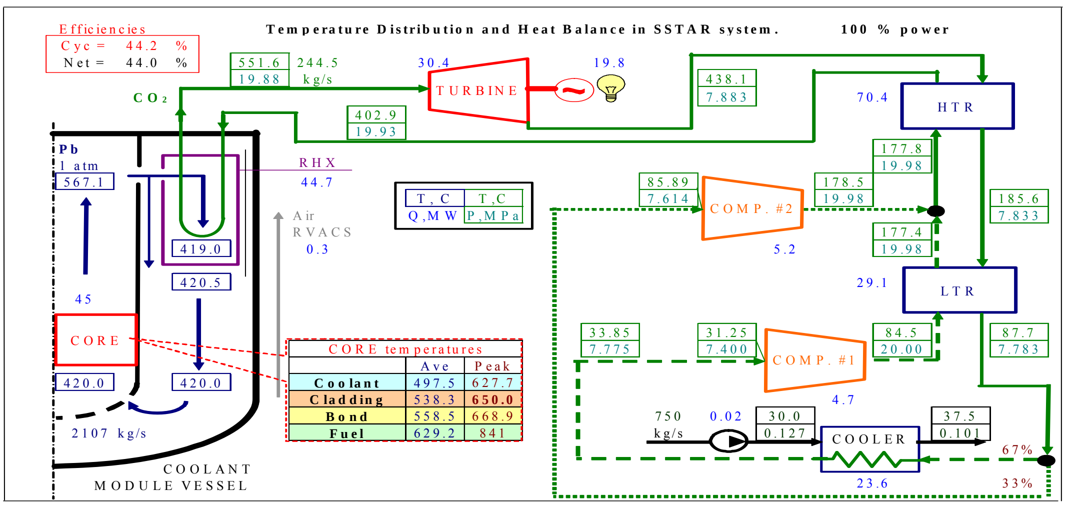


Table 3-2: Key Design Parameters of GIF LFR Concepts

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

Status of cooperation

The cooperation on LFR within GIF was initiated in October 2004, and the first formal meeting of the provisional system steering committee (PSSC) was held in March 2005. Subsequently, the PSSC held periodic meetings, with participation of representatives from Euratom, Japan, the United States and experts from the Republic of Korea to prepare a draft system research plan (SRP). The draft SRP was completed at the PSSC meeting held in Genoa, Italy, in May, 2010. The draft was further reviewed by the expert group and the SRP was finalized in October, 2010.

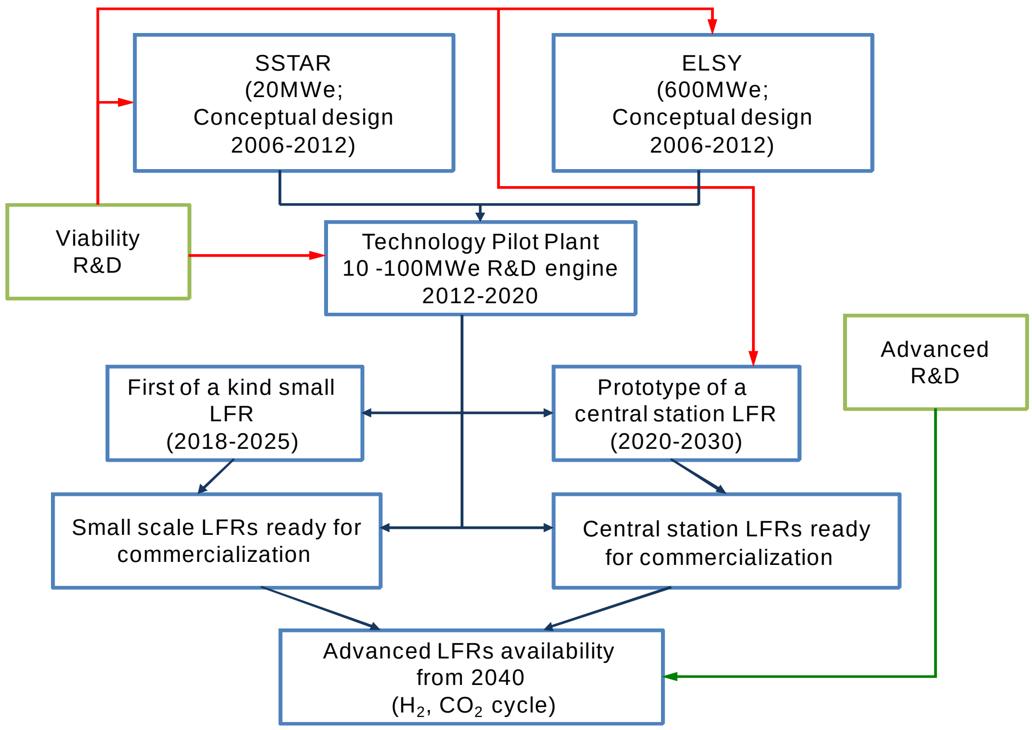
In 2009 discussions were held on the mode of cooperation on LFR and MSR R&Ds in GIF. The policy group took the decision to set up a memorandum of understanding (MOU) for both the LFR and MSR systems. This MOU would provide a more flexible structure for R&D cooperation on those systems in the GIF framework for the mid-term. In November 2010, the memorandum of understanding (MoU) for collaboration on the LFR system was signed by the signatories of JRC, for Euratom and of the Centre for Research into innovative nuclear energy systems from the Tokyo Institute of Technology, for Japan. USA will remain as observers. This MOU is effective as of 22 November 2010.

R&D objectives and milestones

The SRP for the LFR is based on the use of molten lead as the reference coolant and lead-bismuth as the back-up option. The preliminary evaluation of the concepts included in the plan covers their performance in the areas of sustainability, economics, safety and reliability, proliferation resistance and physical protection. Given the R&D needs for fuel, materials, and corrosion control, the LFR system is expected to require a two-step industrial deployment: reactors operating at relatively low primary coolant temperature and low power density by 2025; and high-performance reactors by 2040.

Figure 3-20 illustrates the basic approach recommended in the SRP. It portrays the dual track viability research program with convergence to a single, combined technology pilot plant leading to the eventual deployment of both types of systems.

Figure 3-20: Conceptual Framework for the LFR R&D



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

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

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

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

The needed research activities are identified and described in the SRP. It is expected that coordinated efforts can be organized in four major areas and formalized as projects (C.F. Smith *et al*., 2009): system integration and assessment; lead technology and materials; system and component design and fuel development. The goals and activities of these four R&D projects are summarized below.

System integration and assessment (SIA) project

The ultimate goal of the SIA project, in support to the LFR SSC, is to ensure the feasibility of the LFR system to meet with the GIF objectives for each track defined in the SRP taking into account schedule and cost. The LFR SIA activities are carried through an iterative process aimed at ensuring that R&D projects, either individually or together satisfactorily address the GIF's criteria of safety, economy, sustainability, proliferation resistance and physical protection. The LFR SIA activities will also promote communications and dialogue among R&D PMBs.

System and component design project

System design activities are conducted in the following areas: preliminary design of a central station LFR, preliminary design of a small scale plant, design of the technology pilot plant (TPP), safety approach, component development, balance of plant and hydrogen generation.

Fuel development project

The LFR fuel development project is a continuing long term process consisting of tasks designed to meet progressively more ambitious requirements. It includes efforts in the areas of core materials development, fuel fabrication, fuel irradiation and tests aimed at fuel qualification.

In the near term, an essential goal is to confirm that at least some technical solutions exist so that fuel can be provided in an early time frame that is suitable for the demonstration reactor system. This "fuel for the Demo" milestone achievement will provide the assurance, at the demonstration stage, of the feasibility of a safe and competitive LFR for electricity production.

In the mid-term, it is necessary to confirm the possibility of using advanced minor actinide (MA) bearing fuel at levels representative of the specified equilibrium fuel cycle in order to assure minimization of long-lived nuclear waste. The second goal is to confirm the possibility of achieving higher fuel burn-up when compared with that reached in current liquid metal reactors.

In the long term, it is important to confirm the potential for industrial deployment of advanced MA-bearing fuels and the possibility of using fuels that can withstand high temperatures to exploit the advantage of the high boiling temperature of lead in order to increase plant efficiency for electric energy generation and provide the possibility of high temperature heat production. This "advanced high temperature fuel" milestone achievement will demonstrate the sustainable, multipurpose capability of the LFR technology.

Lead technology and materials project

In the near term, because the development of new materials is a very time consuming process, it is necessary to maximize the use of available materials thereby limiting material qualification activities to their qualification in the new environment. To establish reactor feasibility, it is necessary to provide a technologically viable structural material capable of withstanding the rather corrosive/erosive operating conditions of a LFR.

In the mid and long term, the high boiling point of lead is convenient for a high temperature operation of the reactor extending the LFR mission towards higher efficiency in energy generation and hydrogen production. Those missions require the development of new materials both for mechanical components and fuel cladding. The development of that material will be time consuming and will be carried out with a flexible schedule depending on investments and technological achievements. Peculiar is the development of a fuel cladding resistant to high neutron doses (for increased fuel burn-up) and at high temperature (for increased coolant temperature and power density).

Main activities and outcomes

Following the conclusion of the ELSY project in February 2010, the LFR design activity has continued forward under the next 7th framework program of Euratom with the lead-cooled European advanced demonstration reactor (LEADER) project. LEADER started its activities in April 2010 and is intended to confirm the innovations embodied in ELSY, to identify complementary solutions, to complete the assessment of an industrial LFR and to perform a preliminary design of a DEMO, the prospective facility that will validate the technical solutions of the industrial reactor. The main reference parameters of the DEMO, called ALFRED (advanced lead fast reactor European demonstrator), have been defined and a design development strategy agreed between the partners. ALFRED power has been set to 300 MWth, using pure lead as coolant, both primary and secondary cycles are identical to the cycles already defined in the ELSY project, while the reference design of some specific components has been changed with the aim to shorten the timing of construction phase.

Detailed design of the European technology pilot plant (Myrrha, to be realized in Mol, Belgium) started in 2009 and continued in 2010 with the central design team (CDT) project in the frame of the 7th framework program of Euratom. Myrrha, originally conceived as an accelerator driven system (ADS), operated in sub-critical conditions, recently extended its original objective to include also a critical mode of operation. As a consequence, although maintaining its main scope of being an irradiation facility, it will serve as a pilot plant for both lead technology applications such as ADS and LFR. Using lead-bismuth as coolant, Myrrha is characterized by lower temperatures and no electric energy production and will be an important first step toward the LFR DEMO. The Belgian Government has approved funding of the facility in spring 2010 up to 40% of the expected full cost, close to one billion Euros. Additional funding is expected from European as well as non-European countries with the aim to foster worldwide efforts on the technology.

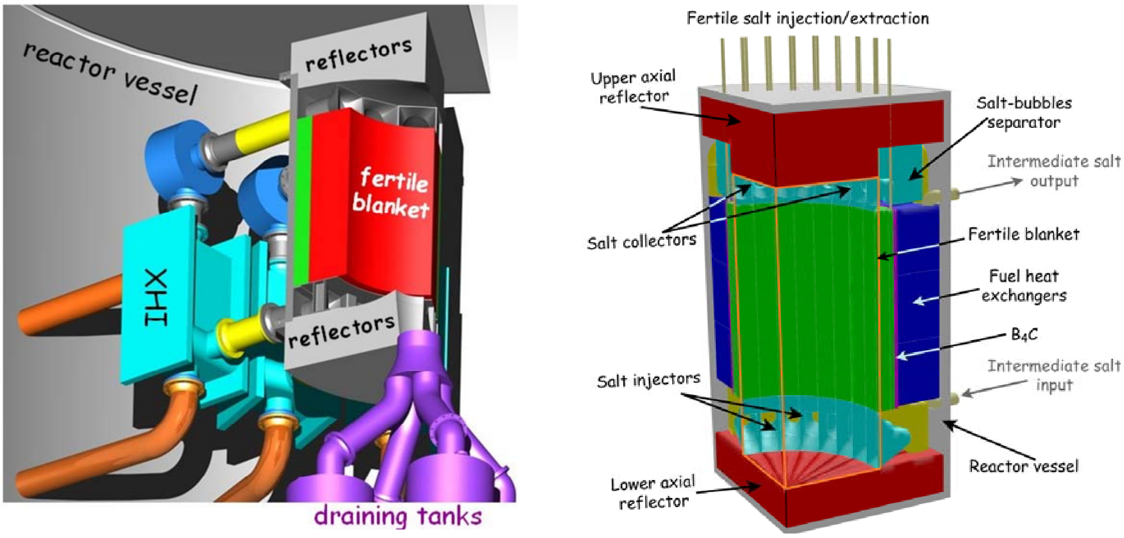
3.1.6 Molten Salt Reactor (MSR)

Main characteristics of the system

In a molten salt reactor (MSR), the fuel is dissolved in a fluoride salt coolant. Previously, MSRs were mainly considered as thermal-neutron-spectrum graphite-moderated concepts. Since 2005 R&D has focused on the development of fast-spectrum MSR concepts (MSFR) combining the generic assets of fast neutron reactors (extended resource utilization, waste minimization) with those relating to molten salt fluorides as fluid fuel and coolant (favorable thermal-hydraulic properties, high boiling temperature, optical transparency). In addition, MSFRs exhibit large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors. MSFR systems have been recognized as a long term alternative to solid-fuelled fast-neutron systems with unique favorable features (negative feedback coefficients, smaller fissile inventory, easy in-service inspection, simplified fuel cycle, etc.).

Figure 3-21 illustrates the 3 000 MWth (~1 350 MWe) MSFR reference concept. The core, fertile blankets and primary circuit (including intermediate heat exchangers) can be arranged as a pool-type reactor with a high degree of compactness (5 m diameter vessel). The fuel salt flows upward in the core until it reaches an extraction area which leads to salt-bubble separators through salt collectors. The salt then flows downward in the fuel heat exchangers and the pumps before finally re-entering the bottom of the core through injectors. The fuel salt runs through the total cycle in around 3-4 seconds, depending on the salt flow velocity. The total fuel salt volume is distributed half in the core and half in the external fuel circuit (salt collectors, salt-bubble separators, fuel heat exchangers, pumps, salt injectors and pipes). This external fuel circuit is broken up into 16 identical modules distributed around the core, outside the fertile blanket.

Figure 3-21: MSFR Pre-conceptual Design



In the MSFR, the liquid fuel processing is part of the reactor where a small side stream of the molten salt is processed for fission product removal and then returned to the reactor. Because of this design characteristic compared to classical solid-fuel reactors, the MSFR can therefore operate with widely varying fuel compositions.

Apart from MSR systems, use of liquid salt in other advanced reactor concepts is also studied. Fluoride-cooled high-temperature reactors (FHRs) combine the use of liquid fluoride salt coolants (like MSRs), pool type cores and vessel configurations in common with many sodium-cooled reactor designs, and fuel coated particles similar to high temperature gas-cooled reactors. The two most developed FHR designs are the 1 200 MWe advanced high temperature reactor (AHTR) that employs prismatic fuel elements and the 410 MWe Pebble Bed advanced high temperature reactor (PB-AHTR).

Status of cooperation

The decision for setting up a provisional system steering committee (PSSC) for the MSR with Euratom, France, the Russian Federation and United States was taken by the GIF policy group in May 2004. A memorandum of understanding (MOU) has been signed on 6 October 2010, by France and JRC, on behalf of Euratom, United States and Russian Federation will remain as observers, even if Russia considers signing the MOU in the medium term. In February 2010, the MOSART (molten salt actinide recycler & transmuter) and MSFR (molten salt fast reactor) white papers were presented to the GIF proliferation resistance & physical protection working group. In April 2010, MSR safety concerns, issues and benefits were presented to the GIF risk and safety working group.

In 2010 two meetings of the MSR PSSC were held, the first one in Paris on March and a second one in Oak Ridge National Laboratory (ORNL) on September. The ORNL meeting was coupled with the fluoride salt-cooled high-temperature reactor workshop organized by ORNL and provided the opportunity for SSC members to have an exhaustive view of USA efforts and plans to develop this type of molten salt reactors. At the end of the SSC meeting an open session was organized opened to observers from institution not involved in GIF MSR activities. During this last session, progress and future plans in MSR R&D programs were presented.

Partners of the MSR PSSC are involved in the Euratom-funded ISTC-3749 project[10], which started in February 2009 with official support from France (CEA, CNRS, EDF), Germany (FZK), Czech Republic (NRI), United States (ORNL), Euratom (JRC-ITU) and IAEA.

In 2009 a new MSR European project proposal was submitted to the 3rd call of the 7th framework program as a joint Euratom-Rosatom project. This EVOL (evaluation and viability of liquid fuel fast reactor systems) project has been approved in 2010 and the kickoff meeting planned for February 2011. A complementary ROSATOM project named MARS (minor actinides recycling in molten salt) project between Russian research organizations (RIAR, KI, VNIITF and IHTE) will be carried out in parallel. Their common objective is to propose a conceptual design of MSFR by 2012 as the best system configuration – resulting from physical, chemical and material studies – for the reactor core, the reprocessing unit and wastes conditioning. It is intended to deepen the demonstration that the MSFR system can satisfy the goals of Generation IV in terms of sustainability (Th breeder), non proliferation (integrated fuel cycle, multi-recycling of actinides), resource savings (closed Th/U fuel cycle, no uranium enrichment), safety (no reactivity reserve, strongly negative feedback coefficient) and waste management (actinide burner).

R&D objectives

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

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

The MSFR system operated with thorium fuel cycle. Although its potential has been assessed, specific technological challenges remain and the safety approach has to be established.

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

In addition, opportunities offered by liquid salts as intermediate fluid for heat transport in other systems (SFR, LFR, VHTR) are investigated. Liquid salts offer two important advantages: small equipment size, because of the high volumetric heat capacity of the salts; and absence of chemical exothermal reactions between the reactor, intermediate loop and power cycle coolants.

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

These issues have been the basis for the implementation of projects. Following experts group recommendations, six projects have been proposed:

Materials and components (selected as the first priority project plan)
System design and operation
Safety and safety system
Liquid salt chemistry and properties
Fuel and fuel cycle
System integration and assessment

10. "Experimental study on critical issues of nuclear energy systems employing liquid salt fluorides".

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

Milestones

The MSR PSSC re-evaluated the milestones mentioned in the original GIF Technology Roadmap, owing to the peculiar and more innovative position of MSR among other Generation IV systems, leading to the following milestones:

Up to 2011	Scoping and screening phase
2012-2017	Viability phase
2018-2025	Performance phase
2031	FHR Prototype operational
After 2035	MSFR Prototype demonstration phases (final design, construction and operation of prototypes) has also been discussed, envisioning a MSFR prototype.

Main activities and outcomes

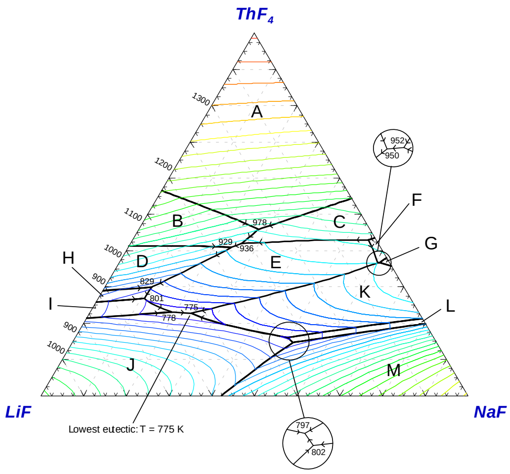
Significant progress was achieved in 2010, including:

1. Development and tests of a numerical tool dedicated to the calculation of the whole MSFR system evolution through the coupling of neutronic and reprocessing simulation codes. This tool may be used to evaluate the extraction capacities of fission products and their location in the whole system (reactor and reprocessing unit), to be used as basis for any safety and radioprotection assessment of the reactor.
2. Assessment on a reprocessing scheme adapted to the MSFR. The scheme takes advantage of a very favorable particularity of MSFR requirements which is a reduced salt flow to be re-process.
3. Thermal simulations of the core of the MSFR have been undertaken to evaluate the thermal evolution of the fuel salt in the core and the output temperature homogeneity at the top of the core.
4. The definitive design of the French molten salt loop (FFFER – forced fluoride flow for experimental research) is decided. After the salt tank, the construction of the loop part is under progress as well as the salt mixture fabrication
5. At the institute for transuranium elements (ITU) the activities on the determination and modeling of the properties of molten fluoride salts have been continued. In 2010 the thermodynamic model of the LiF-NaF-ThF_4-UF_4 system, based on the experimental studies from ORNL in the 1960s, has been completed.
6. For several years, the CEA has been collaborating with Ecole Centrale de Paris and the Université de Toulouse to develop and characterize a specific alloy (Ni-25W-6Cr) for application as a structural material in nuclear systems using molten fluorides.
7. First *annual ISTC 3749* report has been published. The project focuses on the study of the key properties for molten salt mixtures including LiF, NaF, BeF2, ThF4, BaF2 and CaF2 as main constituents. Among the reported work: i) measurement of LiF-ThF4 viscosity, ii) check measurements of thermal conductivity and heat capacity, and iii) verification of electro-analytical technique for redox potential evaluation in salt melts.
8. The Czech Republic reactor LR-0 will be used for the validation of FHR neutronics models (reactivity coefficient variation with temperature). The United States is currently negotiating to supply NRI with molten salt reactor experiment coolant salt containing isotopically separated FLiBe to enable the critical tests.

Some of these topics are further discussed below.

The thermodynamic model of the LiF-NaF-ThF_4-UF_4 system, based on the experimental studies from ORNL in the 1960s, was completed (Figure 3-22), showing that the melting temperature of the salt can be lowered by about 60 K by using the binary LiF-NaF solvent for the actinide tetrafluorides instead of pure LiF. Moreover, experimental studies of the LiF-CaF_2-ThF_4 phase diagram have been undertaken to study the potential of the LiF-CaF_2 solvent and to verify the thermodynamic model developed for that system. ITU has furthermore continued to expand its experimental facilities in the field of molten fluoride salts. A Raman spectrometer has been purchased and a set up for measurements of the Raman spectra of molten salts has been realized and is being tested. This set-up will allow the determination of the local structure of the actinides in the fluoride salts.

Figure 3-22: Calculated LiF-NaF-ThF_4 Pseudo-ternary Phase Diagram with Fixed Concentration of UF_4 Set to 2.55 mol\%.



Primary phase fields: (A) $(U,Th)F_4$ (s.s.); (B) $Li(Th,U)_4F_{17}$ (s.s.); (C) $Na(Th,U)_2F_9$ (s.s.); (D) $Li(Th,U)_2F_9$(s.s.); (E) $(Li,Na)_7Th_6F_{31}$ (s.s.); (F) $Na_7(Th,U)_6F_{31}$(s.s.); (G) $Na_3Th_2F_{11}$-$Na_5U_3F_{17}$ (s.s.); (H) $Li_7(Th,U)_6F_{31}$ (s.s.); (I) $Li_3(Th,U)F_7$ (s.s.); (J) (Li,Na)F(s.s.); (K) $Na_2(Th,U)F_6$ (s.s.); (L) $Na_7(Th,U)_2F_{15}$ (s.s.); (M) (Li,Na)F(s.s.) (Beneš, et al. J. Nucl. Mater. 404 (2010) 186.)

The reference material (Ni-25W-6Cr) was manufactured as a solid solution alloy with a homogeneous microstructure and an average grain size of 500µm. In 2010, the alloy general corrosion resistance was assessed using immersion tests in LiF-NaF eutectic at 750°C for 350 and 900 hours. The immersed specimens were characterized using SEM observations and EDX spectroscopy. Material damage is significant after exposure to LiF-NaF at 750°C with uniform corrosion morphology along the specimen length. Two specific regions were identified:

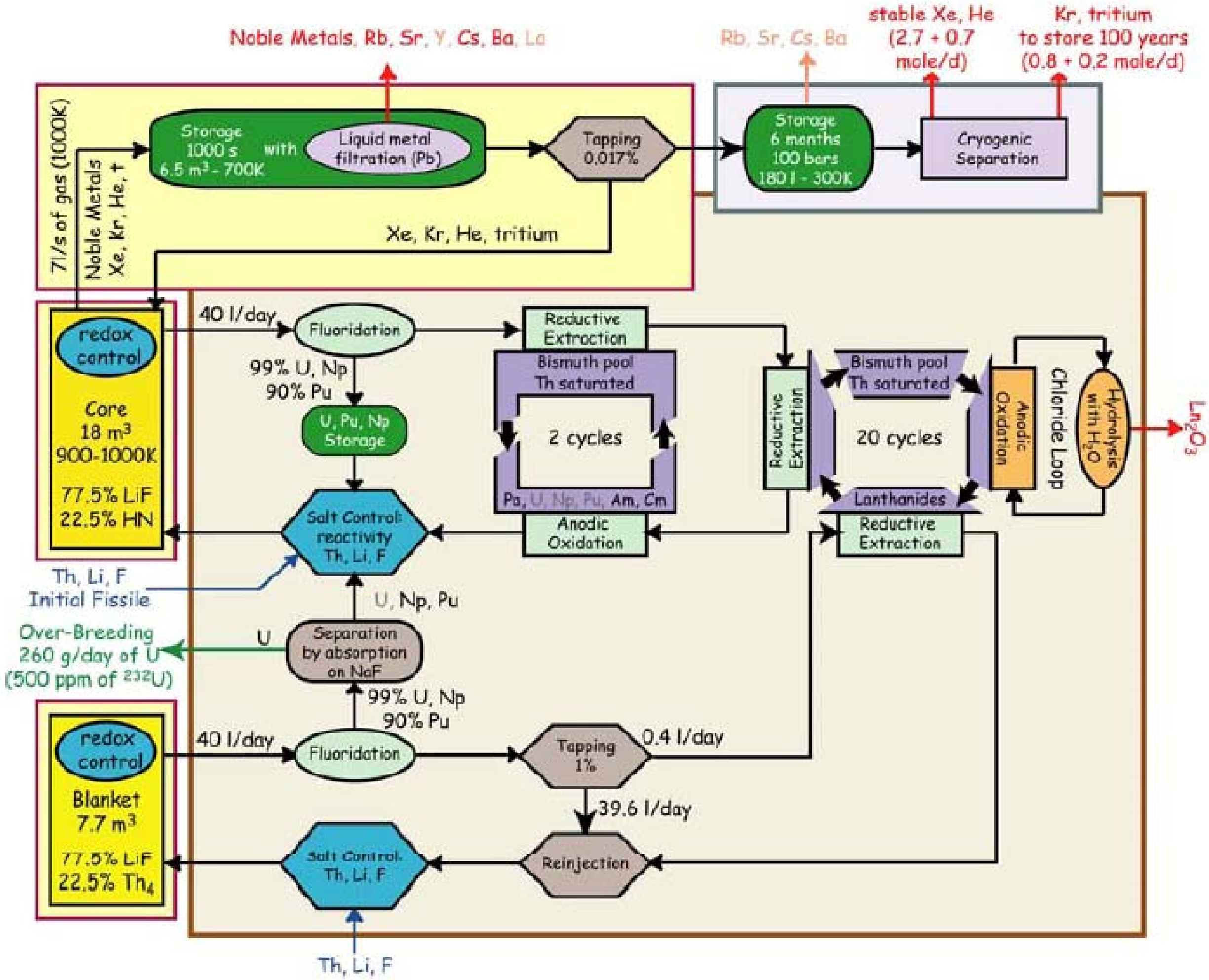
A region strongly depleted in Cr and W which exhibits many voids. Observations at a higher magnification and localized analyses verified that the suspected pores were actually unfilled. It is assumed that this closed porosity was formed by condensation of the vacancies injected during the selective chromium oxidation at the alloy surface.

A region where the Cr and W concentrations are roughly constant and slightly inferior to the bulk values which shows a modified microstructure with smaller grains.

Assessment on a reprocessing scheme adapted to the MSFR has been carried out, extrapolated from the molten salt breeder reactor (MSBR, from ORNL, US) process (volatilization, reductive extraction and metallic transfer). The scheme takes advantage of a very favorable particularity of MSFR requirements which is a reduced salt flow to be reprocessed. The flowsheet was modeled and its performances assessed. Particular attention was paid to the type and waste stream that would be generated by the treatment. The study shows that the greater part of the scheme designed for the MSBR can be adapted to MSFR. The calculations showed that the process is thermodynamically feasible and the required performances achievable. Around 98% of the initial content of Ac and Th return to the reactor with less than 25% of the initial lanthanides. However the process is not able to eliminate alkaline and alkaline earth metals and the process efficiency in separating Zr is questionable and should be investigated.

The on-site salt management of the MSFR combines a salt control unit, an on-line gaseous extraction system and an offline lanthanide extraction component by pyrochemistry. This salt reprocessing scheme is presented in Figure 3-23. The only continuous salt chemistry process (yellow part on top left of the figure) is the gaseous extraction system. First, helium bubbles are injected at the lower part of the core to trap the non-soluble fission products (noble metals) dispersed in the flowing liquid as well as the gaseous fission products. A liquid/gas phase separation is then performed on the salt flowing out of the core to extract gaseous species and dragged condensed particles. Following this “physical” process of purification, a small part of the gas is withdrawn to let the fission products decay, and the remaining part of gas is sent back to the lower part of the core. The salt properties and composition are monitored through the on-line chemistry control and adjustment unit. A fraction of salt is periodically withdrawn and reprocessed off-line in order to extract the lanthanides before it is sent back to the core. In this separate batch reprocessing unit 99% of uranium (including ^{233}U) and neptunium and 90% of plutonium are extracted by fluorination and are immediately reintroduced in the core. The remaining actinides are then quickly extracted together with protactinium and also sent back to the core. Finally, the lanthanides are separated from the salt through a second reductive extraction and sent to waste disposal.

Figure 3-23: Overall Scheme of the Fuel Salt Management



Bubbling treatment requires the insertion of an injector and a liquid gas separator in the salt circuit between the core and heat exchangers, as shown in Figure 3-24. In order to begin the conception of the bubbling components for reactor scale, an experimental project was launched in France, based on the construction of a molten salt loop, the forced fluoride flow for experimental research (FFFER) project. Studies dedicated to bubbling cleanup process have led to the conception of a liquid-gas separator with satisfying efficiency when measured on water mock up. The volumic gas rate domain investigated is between 0.02 to 0.5%. Specific features determined on the mock up are reported on the metallic separator internal design. The whole design of the loop is now complete. It is presented in Figure 3-25 without any instrumentation, heating or insulation. The loop tank separation system comprises two parts in parallel, a metallic valve and a cold plug. Tests of the cold plug have yet to be realized before final assembly. The FFFER loop will be operated with LiF-NaF-KF salt. The salt tank can contain up to 100 liters but the loop circuit is designed for running with a volume ranging from 50 to 80 liters.

Figure 3-24: Schema Giving the Position of the Bubbling System Components

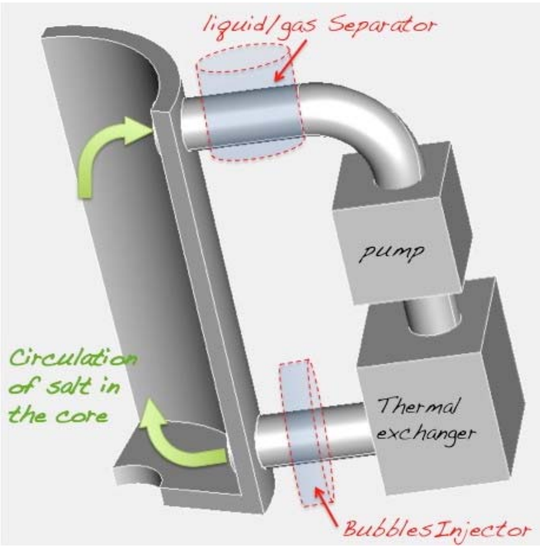
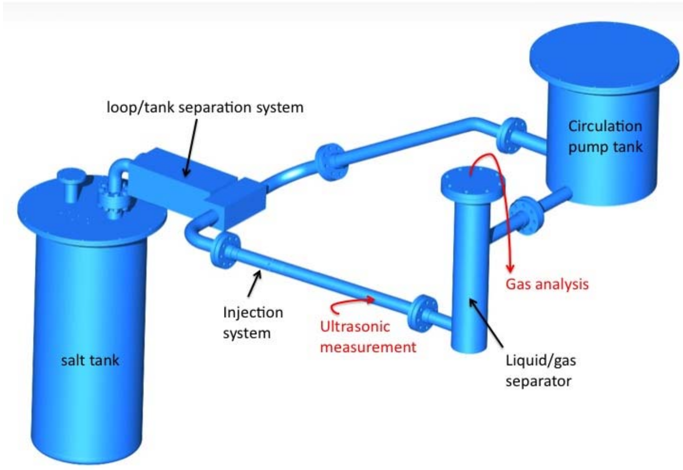


Figure 3-25: FFFER Molten Salt Loop Design, without Instrumentation, Heating, Insulation and Gas Supply



3.2 Assessment Methodologies

The three methodology working groups (MWGs) of GIF – economic modeling (EMWG), proliferation resistance and physical protection (PRPPWG), and risk and safety (RSWG) – were established between late 2002 and early 2005. Their overall objective is to design and implement methodologies for evaluating the GIF systems against the goals defined in the *Technology Roadmap for Generation IV Nuclear Energy Systems* (GIF, 2002) in terms of economics, proliferation resistance and physical protection, and safety.

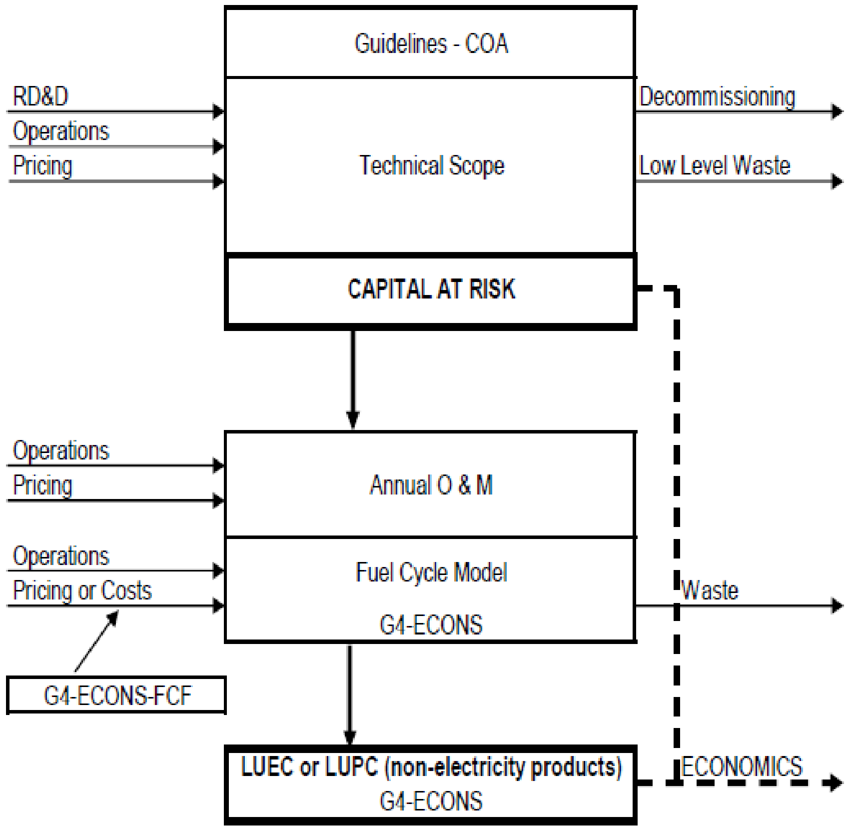
3.2.1 Economic Assessment Methodology

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

The methodology developed by the EMWG (Figure 3-26) 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),
- to have a level of financial risk comparable to other energy projects (i.e., to involve similar total capital investment and capital at risk).

Figure 3-26: Structure of the GIF Cost Estimating Methodology

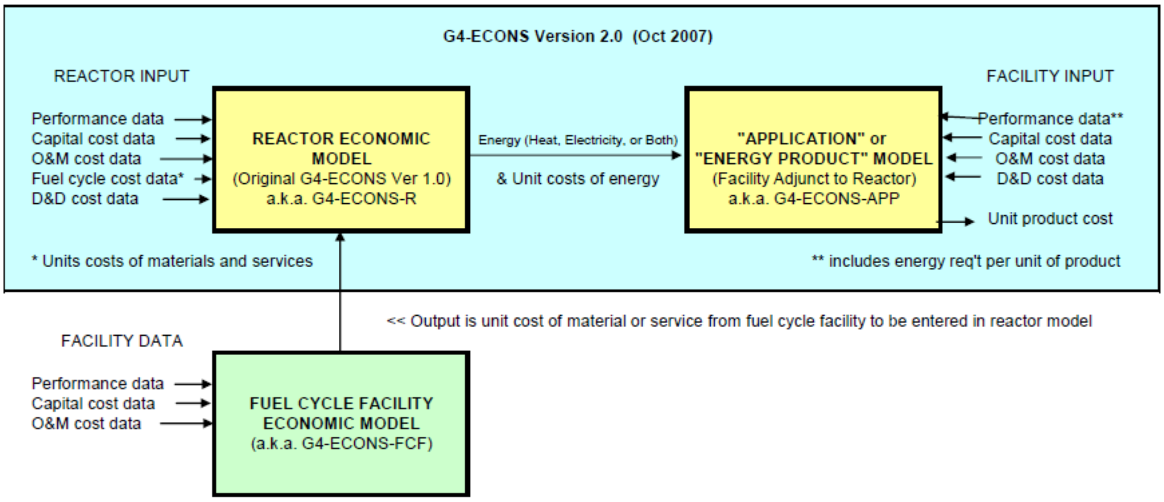


The methodology produced by EMWG consists of:

- Cost estimating guidelines for Generation IV nuclear energy systems, Rev. 4 (GIF/EMWG/2007/004).
- G4ECONS software package (Figure 3-27).
- Users manual for G4ECONS Version 2.0 (GIF/EMWG/2007/005).

Sample calculations have been performed using the cost estimating guidelines and the G4ECONS software for both Generation III and Generation IV systems to demonstrate its validity.

Figure 3-27: Overall G4-Econs Modeling System



Several papers demonstrating implementation of the GIF cost estimating methodology were presented by EMWG members at the GLOBAL 2009 Conference held in Paris in September. The EMWG also participated in the concurrent GIF Symposium and presented a paper over viewing the methodology and its applications.

In 2008, the EMWG, with the agreement of the GIF experts and policy groups, released the methodology for public as well as GIF application. A CD is available from OECD/NEA containing the complete methodology. To date, over 60 copies of the methodology CD have been provided to those organizations requesting its use. In addition to GIF groups, the software has been requested by various IAEA groups and several Universities.

In 2009, the EMWG developed a standard Training Presentation. The training presentation is modularized so as to be useful for presentation from a management level to a detailed user's level. EMWG members are prepared to give this presentation to GIF groups as requested.

In 2010, the EWMG continued with improvement of the G4ECONS software to better facilitate the analysis of heterogeneous fuel cycles which may be proposed for fast reactor systems and particularly for interest in actinide management applications. Several studies were begun to demonstrate an approach for estimating the cost of actinide management services. Applications of the GIF methodology by other groups and institutions were reviewed to gain feedback and experience which may be helpful to GIF groups in the future.

The EMWG continues to monitor the use of the methodology and encourages feedback on its use and possible improvement. Interactions with the experts group, the policy group and the senior industry advisory panel on economic and cost matters continue as requested.

3.2.2 Proliferation Resistance and Physical Protection Assessment Methodology

The proliferation resistance and physical protection working group (PRPPWG) is one of the methodology groups created by GIF to carry out horizontal activities of interest to all the system steering committees (SSCs) which are pursuing R&D on each of the six GIF nuclear systems. The PRPPWG has developed a methodology which provides designers and policy makers with a formal comprehensive approach to assess the proliferation resistance and physical protection performance of GEN IV nuclear systems. This PR&PP methodology is presented in a document entitled *Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems Revision 5,* which was released for general distribution in 2006 (GIF/PRPPWG/2006/005, www.gen-4.org/Technology/horizontal/PRPPEM.pdf).

The PRPPWG has used the methodology in several case studies to illustrate its possible applications and the types of results it permits to obtain. In particular, the example sodium fast reactor (ESFR) case study, was completed in 2009 (GIF/PRPPWG/2009/002) and, after its review by the GIF experts and policy group, released for general distribution in 2010. It is available on the public GIF website (http://www.gen-4.org/Technology/horizontal/proliferation.htm).

The main objective of the case study was to exercise the PR&PP methodology on an illustrative example of Generation IV systems and to show how its results may assist decision makers in comparing design options from PR&PP viewpoints. The study combined a practical approach based upon the data available at an early stage of system design development and a rigorous framework providing robust and dependable results which can be traced, reproduced and explained to policy makers.

In 2010, the PRPPWG focused its activities on:

- Enhancing the PR&PP methodology taking advantage of feedback from the experience gained through the ESFR case study and other applications;
- Developing an updated version of the methodology document (GIF/PRPPWG/2006/005), published in 2006, with emphasis on guidance to users; and
- Continuing collaborative work with system steering committees to prepare a report on PR&PP aspects of the six GIF systems.

In addition, the group continued to publicize its methodology through presentations in national and international forums and publications in scientific journals. Other activities included continued contacts and discussions with experts in the field, and collaboration with national and other international programs addressing PR&PP issues.

The work on methodological aspects focused mainly on measures and metrics (M&M) and expert elicitation (EE).

The definitions of measures and metrics (M&M) recommended in the approach have been refined and better guidance was developed to help users in choosing adequate M&M in any specific case study. Furthermore, recommendations were developed to assist users, analysts and policy makers in interpreting the results of PR&PP assessment.

Expert elicitation is a process used to draw information from knowledgeable people when an assessment is needed but physically-based data are limited or open to interpretation. Although a formal EE is not required in the PR&PP methodology, the use of EE is helpful in providing a systematic, credible and transparent qualitative analysis, and inputs for quantitative analyses. In 2010, a subgroup of the PRPPWG developed a white paper on EE which served as a basis to increase the awareness of the Group on the benefit of the process and to elaborate on its use in the revised version of the methodology document (Rev. 6) issued in September 2010.

The main outcomes from the work carried out on M&M and EE are integrated in the revised and updated version of methodology document entitled *Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems Revision 6* which was issued as working document at the end of September 2010. The draft document is being reviewed by the group and it is planned to release it to the GIF experts group by mid-2011. Like Rev. 5 of the document, Rev. 6 is intended eventually for release to a broad audience within and outside GIF, after its clearance by the policy group.

Collaborative work with SSCs was pursued through workshops, conference calls and exchange of information by electronic mail. The overall objective of interaction with SSCs is to raise the awareness of research teams about proliferation resistance and physical protection aspects of the design concepts under development and to provide a framework for incorporating PR&PP characteristics into the design process. Integrating PR&PP issues and concerns at an early stage of system design, in an approach similar to that adopted for nuclear safety, should enhance the efficiency and cost/effectiveness of measures taken to improve proliferation resistance and physical protection of GEN-IV systems.

The main goals pursued in 2010 in the field of collaboration with SSCs were: to finalize the six system white papers (SWPs) which were conceived and drafted in preliminary forms in 2009; to harmonize and compile the SWPs within a document on PR&PP aspects of GIF nuclear systems; and to initiate preliminary reflections on possible joint studies with SSCs on assessment of PR&PP aspects of a GIF nuclear system at an early stage of design concept.

The SWPs provide an overview on technology characteristics and status of design development for each system, covering the various design options under consideration by each SSC. They highlight PR&PP relevant aspects, concerns and issues raised as well as the approaches which were adopted or are being considered to address PR&PP challenges. Also, they elaborate on R&D needs and programs included in the research plans in the field of proliferation resistance and physical protection.

The group supported the preparation of the SWPs by developing a template for the paper, and providing assistance to the respective authors upon request. The SSCs collected the required information and issued successive drafts which were reviewed by members of the PRPPWG and then revised by their respective authors. This iterative process contributed to a better understanding of the PR&PP issues and of the importance of integrating PR&PP concerns in the system design at an early stage.

A workshop for SSC representatives was held on 25 January 2010 in Bologna, Italy, in connection with the 20th meeting of the group. Each SSC representative presented their respective SWP and responded to question raised by members of the group and representatives of other SSCs. The main outcomes from the workshop were a consensus on the overall framework of the SWPs and their table of contents, and an agreement on the time table for completing and issuing jointly a document on PR&PP aspects of GIF nuclear systems.

Following the schedule agreed upon, SWPs were cleared by the SSCs for integration in the document on PR&PP aspects of GIF systems by mid-2010. The co-chairs and technical director of the group, assisted by the secretariat, developed an introductory section to the document covering its overall objectives and scope, and providing a brief review of cross-cutting issues relevant for all systems. They compiled the six SWPs, and carried out a preliminary technical editing for checking the consistency of terminology and harmonizing in so far as feasible the style and length of the contributions from SSCs. A draft document completed in December 2010 is under review by the group and by SSC co-chairs. It is planned to submit to the experts group the final draft cleared by the SSCs and the group by mid-2011. After clearance by the experts and policy groups, the PRPPWG plan to polish the document for publication and distribution to a broad audience within and outside GIF.

Members of the PRPPWG participated in a large number of international events, including the institute of nuclear materials management (INMM) 51st annual meeting in July 2010 in Baltimore, MD USA, the American nuclear society annual meeting, and several conferences organized by the IAEA, to present the work of PRPPWG, its methodology and its results. Specific sessions of international meetings dedicated to the PR&PP methodology and its applications provided opportunities to discuss with other experts and get feedback on its perceived benefits and drawbacks.

In collaboration with the editor of nuclear technology (NT), a special edition on PR&PP issues was prepared. The special edition will contain a series of papers authored mainly by members of the group, including some papers based upon presentations made at the Global 2009 conference, but also by external experts working on different approaches to PR&PP assessment. The articles have been submitted to NT for peer review and the special edition is expected to be published in the first half of 2011.

Collaboration with the international project on innovative nuclear reactors and fuel cycles (IAEA/INPRO) was pursued in 2010. Since the maturation of both the GIF PR&PP and INPRO PR assessment methodologies, there has been recognition of possible areas of coordination between the two approaches. Several members of each group have met since 2008 to explore these interfaces and propose a path forward that takes advantage of any efficiency and synergies that arise from coordination. It is acknowledged that both methodologies should answer the same general questions from the various users of PR assessments, and that both should address the same general measures used in assessment. At the same time, it is acknowledged that each methodology serves a slightly different purpose and audience, and accordingly differs somewhat in scope.

A practical example of coordination between GIF PRPPWG and INPRO was recently demonstrated by the INPRO collaborative project on "proliferation resistance: acquisition/diversion pathway analysis" (PRADA), which successfully integrated the core of the PR&PP methodology into the implementation of its user requirement #4, which assesses robustness of proliferation pathway barriers. INPRO intends to update its PR methodology with this development.

Within GIF, collaboration with the risk and safety working group (RSWG) was strengthened as the RSWG progressed on the formalization of its methodology. Topics for further discussion between the two groups were identified including: establishment of an integrated framework encompassing the RSWG and PRPPWG methodologies; and identification of synergies and complementarities in the two approaches and evaluations. Provided that SSCs would support such a study, the two groups could undertake a pilot demonstration of applying the RSWG and PRPPWG approaches simultaneously to a GIF system at an early stage of design concept.

3.2.3 Risk and Safety Assessment Methodology

In accordance with its terms of reference, the primary objective of the risk and safety working group (RSWG) is to promote a harmonized approach on safety, risk, and regulatory issues in the development of Generation IV systems.

After its initial meeting in 2005, the early work of the RSWG focused largely on identification of high-level safety goals, articulation of a cohesive safety philosophy and discussion of design principles, together with attributes and characteristics that may help ensure the optimal safety of Generation IV systems. The first product of the RSWG, finalized in 2008, was a report entitled "*Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems*" which addresses the safety-related attributes and characteristics that should be reflected in Generation IV nuclear systems.

During 2010 the work of the RSWG focused on the finalization of the methodology presented in 2009, the integrated safety assessment methodology (ISAM), for use throughout the Generation IV technology development cycle. ISAM was presented to the Generation IV system steering committees (see below) and to the experts group. It is envisioned that ISAM will be used in three principal ways:

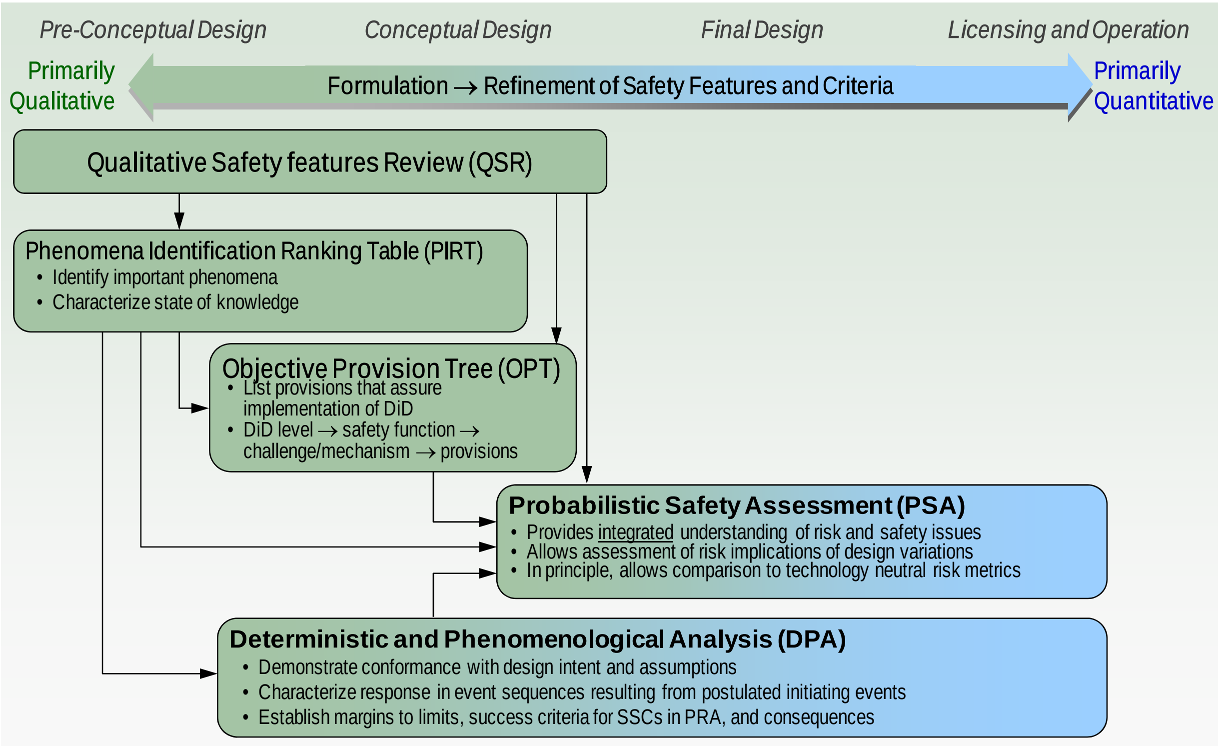
- Throughout the concept development and design phases with insights derived from ISAM serving to actively drive the course of the design evolution. In this application ISAM is used to develop a more detailed understanding of design vulnerabilities, and resulting contributions to risk. Based on this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

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

The integrated methodology consists of five distinct analytical tools and stages which are structured around the last one, the probabilistic safety assessment (Figure 3-28). The tools/stages are the following:

- Qualitative safety requirements/characteristic review (QSR);
- Phenomena identification and ranking table (PIRT);
- Objective provision tree (OPT);
- Deterministic and phenomenological analyses (DPA);
- Probabilistic safety assessment (PSA).

Figure 3-28: Proposed Gen-IV Nuclear Systems Integrated System Assessment Methodology (ISAM)



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

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

As the ISAM concept has matured, the RSWG has initiated interactions with both internal and external entities involved with the activities of the GIF. These include interactions with the systems steering committees (SSCs) and the PR&PP working group, the multinational design evaluation programme (MDEP) and the international project on innovative nuclear reactors and fuel cycles (INPRO). Some of these interactions are briefly described below.

SSCs: a workshop was held in April 2010 attended by representatives of the six SSC where the ISAM was presented as well as the main safety topics relevant for each of the systems. Several SSCs have started using part of the ISAM to clarify their research needs. Following a strategy analogous to the one adopted by the PR&PP, a *White Paper* template was established to launch the interaction with SSCs. The key objective of this document is to provide a sort of "table of contents" of what should be the work for the safety assessment of a given system. Within the document, some tasks are suggested to be under the responsibility of the SSC and the management boards and some others would be under the responsibility of the RSWG. Once finalized, the white paper could provide the foundation for future work of the RSWG.

PR&PP: close interaction between the PRPPWG and the RSWG is recommended by the experts group. Among the members of the RSWG, there is the conviction that part of the approach promoted by the RSWG (ISAM), to support the design and the assessment, can be used as a starting point to converge towards a strategy that would ensure greater consistency to address concerns of both safety and security.

MDEP: an RSWG meeting was the opportunity for the chair of the steering technical committee of the multinational design evaluation programme (MDEP) to express his interest in GIF activities, recognizing that there are a number of common interest areas, like "cooperation on safety goals, severe accidents and operating experience feedback to new reactors." It is now recognized that if nuclear regulators may not be, systematically, part of RSWG, close contacts should be maintained.

INPRO: specific action is engaged to interact with the IAEA/INPRO team in order to discuss the suggestions for improvements of the INPRO methodology.

3.3 Selected Bibliography

J. Starflinger, T. Schulenberg, P. Marsault, D. Bittermann, E. Laurien, C. Maraczy, H. Anglart, J.-A. Lycklama, M. Andreani, M. Ruzickova, T. Vanttola, A. Kiss, M. Rohde, R. Novotny, High Performance Light Water Reactor Phase 2, Public Final Report – Assessment of the HPLWR Concept, 6th framework program, Contract No. FI60-036230, 2010, www.hplwr.eu

Y. Oka, S. Koshizuka, Y. Ishiwatari, A. Yamaji, Super Light Water Reactors and Super Fast Reactors, Springer, ISBN 978-1-4419-6034-4 (2010).

Sienicki, J. J., *et al*., (2008), Status of Development of the Small Secure Transportable Autonomous.

Reactor (SSTAR) for Worldwide Sustainable Nuclear Energy Supply, presented at ICAPP, 13-18 May 2007, Nice, France.

Cinotti, L., *et al.,* (2008), *The ELSY Project,* presented at PHYSOR, 14-19 September 2008, Interlaken, Switzerland.

Alemberti A., *et al.,* (2009), "*European Lead-cooled Fast Reactor*", presented at FISA 2009, Prague 22-24 June 2009.

Cinotti L., *et al.,* (2009), "*Lead-cooled Fast Reactor (LFR) Overview and perspectives*", presented at GIF Symposium 2009, Paris, France September 9-10, 2009.

Smith C. F., *et al.*, (2009), "*Lead-cooled Fast Reactor (LFR) Ongoing R&D and Key Issues*", presented at GIF Symposium 2009, Paris, France September 9-10, 2009.

Sekimoto, H., *et al.*, (2008), *A design and Safety Features of a Small CANDE Fast Reactor*, presented at ICAPP, 8-12 June 2008, Anaheim, CA, United States.

D. Heuer, E. Merle-Lucotte, M. Allibert, X. Doligez, V. Ghetta (CNRS), "Simulations Tools and New Developments of the Molten Salt Fast Reactor", Contribution A0115, Proceedings of the European Nuclear Conference (ENC) 2010, Barcelona, Spain (2010)).

H. Boussier (CEA), D. Heuer (CNRS) "Estudio de sistema de un proceso de tratamiento-reciclado piroquímico del combustible de un reactor de sales fundidas" Annual meeting of Spanish Nuclear Society SNE'10 2010 Santiago de Compostela.

Xavier Doligez (CNRS) "Influence du retraitement physico-chimique du sel combustible sur le comportement du MSFR et sur le dimensionnement de son unité de retraitement" Thèse de doctorat, Grenoble INP, France (2010).

S. Fabre "Comportement de métaux et alliages en milieux fluorures fondus" Ph D defended on the 25th of September, 2009 at Université de Toulouse.

C. Cabet, "Molten salt corrosion", invited oral talk, Nuclear Fuels and Structural Materials workshop (embedded in the ANS annual meeting), San Diego, CA, USA, June 13-17, 2010.

Delpech S., Cabet C., Slim C., Picard G., "Materials for molten fluorides", Materials Today 13, No. 12 (2010) 34-41.

Fabre S., Cabet C., Cassayre L., Chamelot P., Delepech S., Finne J., Massot L., Noel D., "Use of electrochemical techniques to study the corrosion of metals in model fluoride melts" submitted to Journal of Nuclear Materials .

ISTC Project # 3749 Annual Report Experimental study on critical issues of nuclear energy systems employing liquid salt fluorides (From February 01, 2009 until February 01, 2010), International Scientific Centre, Moscow, RF, 2010.

Feynberg O., Neutronic and fuel cycle consideration: from single stream to two fluid Th-U molten salt system, In Proc. of 1st international workshop of 7th Euroatom program ACSEPT project, Lisbon, Portugal, March, 30 - April 02, 2010.

Ignatiev V., Feynberg O., The Molten Salt Reactor In P&T: Overview and Perspectives, In Proc. Actinide and Fission Product Partitioning and Transmutation, 11th Information Exchange Meeting, San Francisco, USA, 1-4 November 2010.

Ignatiev V., Afonichkin V., Feinberg O., Subbotine, V., Surenkov A., Experimental study on critical issues of nuclear energy systems employing liquid salt fluorides, Book of abstracts of Nuclear Materials 2010 an international conference in association with Journal of Nuclear Materials, 4-7 October 2010, Karlsruhe, Germany.

Afonichkin, A.Bovet, V.Shishkin, Salts purification and voltammetric study of U(IV) to U(III) electro reduction in LiF-ThF4 melt, *ibid*.

V.Khokhlov, I.Korzun, V.Dokutovich, E.Filatov, Heat Capacity and Thermal Conductivity of Molten Ternary Lithium, Sodium, Potassium, and Zirconium Fluorides Mixtures, *ibid*.

A.Merzlyakov, V.Ignatiev, S.Abalin, Viscosity of molten lithium, thorium and beryllium fluorides mixtures, *ibid*.

A. Zagnitko, V. Ignatiev, Experimental study on reductive extraction of LaF3 and NdF3 from molten fluoride salts, *ibid*.

A.Surenkov, V.Ignatiev, I.Gnidoi, V.Uglov, V.Sagaradze, A.Shestakova Combined materials compatibility: Te corrosion of Ni-based alloys in molten salt fluorides, *ibid*.

R.Fazylov, J.Dolinsky, E.Shestakova, Yu.Zuev, Hydrogen heavy isotopes permeation studies in Ni-base alloys, *ibid*.

Ignatiev V. and Surenkov A., Alloys compatibility with selected Molten Salt Environments for Molten Salt Fast Reactor Breeders and Burners, Book of abstracts OECD Nuclear Energy Agency, 2nd International Workshop on Structural Materials for Innovative Nuclear Systems, KAERI, Daejeon, Republic of Korea, 31 Aug. – 3 Sep., 2010, p. 33.

Ignatiev V. and Surenkov A., Alloys Compatibility in Molten Salt Fluorides: Kurchatov Institute Related Experience, In Proc. ANS 2010 meeting (Nuclear Fuels and Structural Materials 2010 embedded topical), San Diego, USA, June 13-17, 2010.

Ignatiev V. and Surenkov A., Reactor Coolants: Material Performance in Molten Salts, Comprehensive nuclear materials (NMAT), Article 98, Elsevier Ltd, Oxford, UK, 2011.

A. Zagnitko and D. Chuvilin, Removal of Nanoaerosol During the Bubbling of the Salt Melt of Beryllium and Lithium Fluorides for the Preparation of Reactor Radioisotopes, Russian Journal of Physical Chemistry, 2010, Vol. 84, No. 6, pp. 1009–1014.

CHAPTER 4 COLLABORATIONS WITH OTHER INTERNATIONAL PROGRAMS

The key drivers of the global energy policy (increasing energy demand due to population and economic growth in large developing countries, reduction of greenhouse gas emissions, and concern regarding the security of energy supply) has renewed interest of policymakers in the peaceful application of nuclear energy and had triggered many multinational initiatives in this field. However, exchange of information among those initiatives is a prerequisite to ensure their global effectiveness. The GIF has been very attentive since its inception to collaboration with other projects, especially as regards major international endeavors aiming at the development of advanced nuclear energy systems and, more broadly, at enhancing the contribution of the nuclear option to sustainable energy supply. As GIF activities in the field of R&D on advanced systems are progressing, GIF members place a high priority on strengthening cooperation with other international projects which have complementary objectives and scopes.

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

The following sections describe briefly the interactions of GIF with the three international projects which are the most relevant for GIF activities at present – the international project on innovative nuclear reactors and fuel cycles (INPRO), the global nuclear energy partnership (GNEP), and the multinational design evaluation program (MDEP).

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

The INPRO (www.iaea.org/INPRO/) initiative started in 2000 under the auspices of the IAEA which ensures its management. It provides a forum for discussions between experts and policy makers on the development and deployment of innovative nuclear energy systems (INS). INPRO brings together technology holders and users to consider jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles. 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.

As of December 2010, INPRO membership consists of 31 IAEA members States plus the European Commission participating in its various collaborative projects as well as in joint programs of work in different fields such as methodologies for evaluating innovative nuclear systems and user requirements for those systems. All countries that are members of GIF are also members of INPRO. Therefore, the flow of information between INPRO and GIF is straightforward and its effectiveness relies mainly on representatives of countries participating in both endeavors. However it is important to note that the results of INPRO’s activities are being made available to all IAEA Member States, while GIF projects are aimed at producing Intellectual Property.

The missions and activities of INPRO are broader than those of GIF but in many areas the two projects have complementary roles offering potential for creating fruitful synergies. In particular, items of common interest which were identified include safety, non-proliferation, economics, fuel cycle implications of GIF systems, small and medium sized reactors, and thorium utilization. The areas where exchanges and cooperation between GIF and INPRO are the most relevant are methodologies and user requirements. The comparison and eventual harmonization of methodological approaches adopted have been identified by members of both projects as a key element for cooperation. With regard to user requirements, INPRO can provide GIF technology holders/developers with insights on the needs of future technology users. Collaboration between GIF and INPRO is ongoing also within selected research projects of common interest.

In 2010, the collaboration between GIF and INPRO was pursued through participation of members of the IAEA/INPRO team in the GIF methodology working groups meetings and activities as well as in the GIF policy and experts group meetings. The GIF/INPRO interface meeting held in Vienna on 1–3 March 2010 was the opportunity to discuss main outcomes of the collaboration, mainly dealing with methodologies, as well as to check for new topics for collaboration such as non electric applications.

A joint GIF/INPRO "Workshop on Operational and Safety Aspects of Sodium Cooled Fast Reactors" was conducted in Vienna on 23-25 June 2010. Items addressed during the workshop included the review of safety related SFR experimental and operational experience; the SFR safety approaches, review processes, and approaches to licensing; and the substantiation of safety approaches and safety design goals (ongoing and planned R&D programs). In addition to GIF country members, the meeting welcomed representatives from India and their contribution to the discussions.

4.2 Global Nuclear Energy Partnership (GNEP)/International Framework for Nuclear Energy Cooperation (IFNEC)

The global nuclear energy partnership (GNEP) initiative was launched in 2006 by the U.S. government to serve as a forum to support the development of the peaceful use of nuclear energy in a safe and secure manner.

The GNEP steering group, meeting in Ghana on 16-17 June 2010, approved the evolution of this partnership, changing the name from GNEP to international framework for nuclear energy Cooperation (IFNEC). The IFNEC statement of mission, adopted by the steering group, specify that "*the international framework for nuclear energy cooperation provides a forum for cooperation among participating states to explore mutually beneficial approaches to ensure the use of nuclear energy for peaceful purposes proceeds in a manner that is efficient and meets the highest standards of safety, security and non-proliferation. Participating states would not give up any rights and voluntarily engage to share the effort and gain the benefits of economical, peaceful nuclear energy*".

As of December 2010, IFNEC membership consists of 29 partner countries, 30 observer countries and three permanent observer intergovernmental organizations – the IAEA, the GIF and Euratom.

IFNEC organization is copied from GNEP's. It employs a three-tier hierarchy, led by a ministerial-level executive committee and seconded by a steering group which provides guidance to the two working groups, the infrastructure development working group (IDWG) and the reliable nuclear fuel services working group (RNFSWG).

- The objective of the IDWG is to facilitate the development of the infrastructure needed for the use of clean, sustainable, nuclear energy worldwide in a safe and secure manner, while at the same time reducing the risk of nuclear proliferation. The IDWG addresses infrastructure issues of concern to participating GNEP countries, including human resources development, nuclear plant financing, small modular reactors and radioactive waste management. The IDWG has also performed assessments of the nuclear power-related infrastructure of member countries considering nuclear power for the first time and has created an on-line Resource Library consisting of information on global infrastructure development references, programs, tools, and other resources.
- The RNFSWG addresses issues of assurance of the front- and back-end of the nuclear fuel cycle. Specifically, it is examining needs and potential solutions in the areas of lessons learned and resource requirements, assurances a country should seek as sufficient for nuclear fuel supply, and approaches for selecting back-end fuel cycle options.

While IFNEC encompasses a broader policy vision than GIF, which focuses on technology progress through collaboration within specific R&D projects, both endeavors have similar goals for future nuclear systems, most notably high priority put on safe, secure and sustainable use of nuclear energy, but also improvement of waste management and enhancement of proliferation resistance. In its capacity as an IFNEC permanent observer, GIF participates in IFNEC meetings at all three levels of the hierarchy. GIF has also taken an active role in the discussions organized within the working groups, including taking the lead in involving specialist organizations in the IDWG's activities to facilitate awareness of and access to available information and identify opportunities for joint efforts.

4.3 Multinational Design Evaluation Programme (MDEP)

MDEP is a "multinational initiative taken by national safety authorities to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities who will be tasked with the review of new reactor power plant designs".

According to its terms of reference (http://www.oecd-nea.org/mdep/mdep_ToR.pdf), the governing bodies of MDEP are the policy group and the steering technical committee, which consist of representative from the national safety authorities from the 10 members: 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 these countries 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 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.

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

In order to achieve its long-term goals, MDEP will focus first on cooperation and convergence of regulatory practices that will eventually develop into convergence of regulatory requirements. Regarding this issue, the terms of reference of MDEP state that the steering technical committee "will interact as needed with GIF and INPRO to ensure effective communication and alignment with activities in similar areas." Indeed, an MDEP representative attended the GIF policy group meeting in Vancouver in 2010, where he presented the activities ongoing within this program. Discussions have shown that there are a number of topics of common interest for both groups, such as the evaluation of the similarities and differences in the scope of review for severe accidents, the comparison of top level safety goals, and the comparison of operating experience in reviews for new reactors. As a conclusion, progress towards harmonized regulatory practices and requirements for Generation IV reactor designs will be a natural outcome from the work to be undertaken within MDEP. Obvious synergies exist between GIF activities on risk and safety approach and the MDEP program of work.[11] Therefore, a continued exchange of information will be established between the two projects, each of them benefiting from relevant progress and findings of the other.

11. NEA providing technical secretariat for both MDEP and GIF facilitates exchange of information and realization of synergies between these programs.

A.1.1 Technology Goals of GIF

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

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

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

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

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

A.1.2 GIF Systems

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

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

Table A.1.1 Overview of Generation IV Systems

System	Neutron spectrum	Coolant	Outlet Temperature °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	500-550	closed	50-150 300-1 500 600-1 500
SCWR (supercritical-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	480-570	closed	20-180 300-1 200 600-1 000
MSR (molten salt reactor)	thermal/fast	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 a closed fuel cycle for fuel breeding and/or actinide management. The reactor may be arranged in a pool layout or a compact loop layout. The reactor-size options which are under consideration range from small (50 to 150 MWe) modular reactors to larger reactors (300 to 1 500 MWe). The two primary fuel recycle technology options are advanced aqueous and pyrometallurgical processing. A variety of fuel options are being considered for the SFR, with mixed oxide preferred for advanced aqueous recycle and mixed metal alloy preferred for pyrometallurgical processing. Owing to the significant past experience accumulated with sodium cooled reactors in several countries, the deployment of SFR systems is targeted for 2020.

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

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

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

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

(Published for the GIF Symposium in Paris – September 2009)

The GIF Symposium has the objective to give a global view on ongoing activities within the initiative. At the same time, the "Outlook" document illustrates the foreseen path forward. The following text provides a summary of agreed priority objectives for the different systems in order to help focusing and streamlining the GIF R&D activities during the next five years, consistent with GIF objectives.

These priority objectives result from an analysis based on the following steps:

1) Review of the potential of the system.

2) Development target for the effective use of its potential.

3) Review of the current stage of development and analysis of technology options, with a view to down selection.

4) Assessment of key R&D issues and priority requirements.

These steps are discussed in the "Outlook" document. The summary presented below is essentially related to step 4) and provides for each system some key R&D priorities.

Very-high-temperature Reactor (VHTR)

The VHTR has a long-term vision for operating with core-outlet temperatures in excess of 900ºC and a long-term goal of achieving an outlet temperature of 1 000°C. At the same time, the VHTR benefits from a large number of national programs that are aimed at nearer-term development and construction of prototype gas-cooled reactors that have adopted core-outlet temperatures in the range of 750°C to 850°C. The overall plan for the VHTR within Generation IV is to complete its viability phase by 2010, and to be well underway with the optimization of its design features and operating parameters within the next five years.

Core outlet temperatures

Objective:

- Further assess the range of candidate applications for VHTRs with the core outlet temperatures and unit power required, as well as the associated time line.

Domains of application and priorities

Objectives:

- Spur the interest of industries to use VHTRs to produce high temperature process heat in various industrial applications, thereby displacing fossil fuels and reducing the production of greenhouse gases.
- Make progress towards resolving feasibility issues (processes, technologies) and more reliably assessing performance.
- Update the definition of priority R&D needs.

Hydrogen production

Objectives:

- Make progress towards resolving feasibility issues (processes, technologies) and more reliably assessing performance of hydrogen production processes.
- Update the definition of priority R&D needs and pre-industrial demonstration projects.

Materials for the core and cooling systems

Objectives:

- Make progress towards resolving feasibility issues of high temperature design, including the qualification of heat resisting materials and manufacturing issues for key components of the core and the cooling systems (pressure vessel, intermediate heat exchangers).
- Update the definition of priority R&D needs.

TRISO fuel particles

Objective:

- Establish performance margins of the uranium-dioxide (UO_2) and uranium-oxicarbide (UCO) coated particle fuels and establish fission product source terms.

Sodium-cooled Fast Reactor (SFR)

The SFR has a long term vision for highly sustainable reactors requiring its development in several important technical directions. At the same time, the SFR benefits from the worldwide operational experience of several sodium-cooled reactors and from a number of national programs aiming at nearer-term restart, development and construction of prototype Generation IV reactors. The overall plan for the SFR within Generation IV is to be well underway with the optimization of its design features and operating parameters within the next five years, and possibly to complete its performance phase by 2015.

Advanced fuels

In this area, after the identification of the advanced fuel options, major R&D efforts will be focused on fabrication feasibility and irradiation behavior of minor-actinide-bearing fuels. A preliminary selection of advanced fuel(s) should be made.

The assessment of the high burn-up capability of advanced fuel(s) and materials should follow.

Objectives:

- Make preliminary selection of advanced fuels.
- Define priority irradiations beyond the global actinide cycle international demonstration (GACID) project.
- Progress towards the resolution of feasibility issues regarding actinide recycling.
- Verify that milestones of the GACID project are realistic.

Safety approach

Objectives:

- Progress towards converging safety approaches.
- Revisit re-criticality and potentially positive reactivity coefficient issues, to compare approaches and seek for consensus.
- Assess, among other approaches, the effectiveness of inner-duct structures to mitigate severe accidents while enhancing fuel discharges without the formation of large molten-fuel pool. This assessment may benefit from analyses and conclusions of the EAGLE (experimental acquisition of generalized logic to eliminate re-criticalities) experiment if they can be shared with the international community.

In-service inspection

Research and development of in-service inspection approaches is following three parallel paths each of which is highly innovative in its own right. Significant improvements or breakthroughs in the ability to perform in-service inspection of in-vessel sodium components may result from this ongoing work.

Objectives:

- Draw conclusions from related R&D work and set priorities for the future.
- Progress towards resolving in-service inspection and repair feasibility issues.

Phenix, Monju and possibly CEFR and BN–800 tests

Objective:

- Summarize lessons learned from planned experiments and start-up.

Energy conversion systems

In this field R&D activities cover development and demonstration of sodium-CO_2 Brayton cycle advanced energy conversion systems including: the development and performance testing of compact heat exchangers; development and testing of small-scale sodium-CO_2 turbo-machinery and a complete integrated cycle; sodium-CO_2 interaction testing; CO_2 oxidation and carburization tests; and the analysis of system behavior for SFRs incorporating the sodium-CO_2 Brayton cycle.

Objectives:

- Draw conclusions from related R&D work and define priority research for the future.
- Make progress towards resolving feasibility issues on alternative energy conversion systems with gas or supercritical CO_2.

Materials, codes and standards

Objective:

- Develop of codes and standards for high temperature application (for example RCC-MR published by AFCEN is available and has been used for construction of PFBR).

Supercritical-water-cooled Reactor (SCWR)

The SCWR has a long-term vision for water reactors that requires significant development in a number of technical areas. At the same time, the SCWR benefits from the resurgence of interest worldwide in water reactors as well as an established technology for supercritical water power cycle equipment in the fossil power industry. The overall plan for the SCWR within Generation IV is to complete its viability phase research by about 2010 and to operate a prototype fueled-loop by around 2015, thereby preparing for construction of a prototype reactor sometime after 2020.

Feasibility of meeting GIF goals

The SCWR builds on a strong technical foundation from two advanced technologies: advanced Gen III+ water-cooled reactors; and advanced supercritical fossil power plants. The work performed to date does not show any issues regarding the viability of merging these two well-known technologies. However, the feasibility of meeting GIF goals and the estimation of the extent to which GIF metrics can be improved require significant R&D.

Objectives:

- Improve knowledge base to enable optimized designs and accurate assessments against GIF goals.
- Continue R&D needed to design and build a prototype.
- Continue conceptual designs of the various SCWR versions, including fast and thermal neutron spectrum designs using pressure tube and pressure vessel technologies.

Critical-path R&D

Two critical-path R&D projects have been identified and are currently underway: materials and chemistry; and thermo-hydraulic phenomena, safety, stability and methods development.

Materials and chemistry

Objectives:

- Test key materials for both in-core and out-core components.
- Investigate a reference water chemistry taking into consideration materials compatibility and radiolysis behavior.

Basic thermal-hydraulic phenomena, safety, stability and methods development

Objectives:

- Continue investigating key areas such as heat transfer, stability and critical flow at supercritical conditions.
- Understand better the different thermal-hydraulic behavior and large changes in properties around the critical point compared to water at lower temperatures and pressures although the design-basis accidents for the SCWR will have similarities with conventional water-cooled reactors.

In addition, non-critical-path R&D areas will continue for specific designs in the areas of advanced fuels and fuel cycles (e.g. using thorium in the pressure-tube design and development of the fast-core and mixed-core options for the pressure-vessel design), and hydrogen production.

Gas-cooled Fast Reactor (GFR)

The GFR has a long-term vision for highly sustainable reactors that requires significant development in a number of technical areas. Unlike the SFR, the GFR does not benefit from operational experience worldwide and will require more time to develop. However, the GFR may benefit from its similarities with the VHTR, such as the use of helium coolant and refractory materials to access high temperatures and provide process heat. The overall plan for the GFR within Generation IV is to be well underway with the viability research within the next few years and to be completed by 2012.

Fuel

Work in this field focuses on assessment of multilayer SiC clad carbide fuel pins.

Objectives:

- Identify and demonstrate suitable technologies for pin fuels (low-swelling mixed-carbide fuel, multilayer composite SiC cladding for fuel pins).
- Update irradiation experiments in BR2, and identify other priority R&D needs (e.g. fabrication and behavior at extreme temperature).

Experimental demonstration design

The ALLEGRO experimental prototype is an option within the “European Strategic Research Agenda”.

Objectives:

- Update and improve the definition of the experimental prototype ALLEGRO intended to demonstrate GFR key principles and technologies and to offer multi-purpose services such as fast-neutron irradiations and high temperature heat supply.
- Document ALLEGRO so as to support a decision around 2012 of proceeding towards detailed design studies and implementation.

Safety

GFR conceptual studies and operating transient analyses are priority R&D areas.

Objectives:

- Demonstrate the safety in case of depressurization accident;
- Study the phenomenology of severe accidents in core with ceramic cladding and structures;
- Confirm GFR safety through further accidental-transient analyses, assessments of innovative design features, and documentation of severe accidents analyses. Especially:
  - assess the merits of a pre-stressed concrete primary pressure boundary; and
  - proceed with tests of GFR fuel samples in extreme-temperature conditions.
- Further update the definition of priority R&D needs.

Lead-cooled Fast Reactor (LFR)

The LFR features a fast-neutron spectrum and cooling by an inert liquid metal operating at atmospheric pressure and relatively high temperatures. The main missions include the production of electricity, process heat, and hydrogen, and actinide management aiming at long-term fuel sustainability. The LFR has development needs in the areas of fuels, material performance, and corrosion control. The overall plan for the LFR is to be well underway with the development of its materials, design features, and operating parameters within the next five years.

Heavy liquid metal technology (coolant, materials, components)

Work in this field focuses on progress towards resolving issues related to the feasibility of heavy liquid metal technologies.

Objectives:

- Select and validate candidate structural materials.
- Demonstrate corrosion control (with surface treatment, oxygen control, etc.).

Experimental demonstrations

Whilst the SFR remains the reference technology, the LFR and the GFR are promising alternatives. The LFR has a rather limited operational experience but it has several similarities with the SFR (e.g. fuel cycle). It was thus agreed within GIF that it should benefit from the relevant outcomes of the R&D on the SFR. An experimental reactor with a capacity in the range of 50 to 100 MWth will be needed to gain experience feedback by 2020.

Objectives:

- Update and improve the definition of the experimental prototype LFR.
- Confirm its feasibility and document its merits for testing LFR technologies in support of a decision around 2012 to proceed towards detailed design studies and implementation.

Molten Salt Reactor (MSR)

The MSR has a long term vision for highly-sustainable reactors that requires significant development in a number of technical areas. The overall plan for the MSR is to be underway with the development of its design features, processing systems and operating parameters within the next five years.

Focus

In the United States, a PB–AHTR (900 MWth) has been selected as the lead commercial-scale plant AHTR concept.

In Europe, since 2005, R&D on MSR is focused on fast spectrum concepts (MSFR) which have been recognized as long term alternatives to solid-fuelled fast neutron reactors with attractive features (very negative feedback coefficients, smaller fissile inventory, easy in-service inspection, simplified fuel cycle…). MSFR designs are available for breeding and for minor actinide burning.

Objective:

- Advance cooperative R&D work to further resolve feasibility issues and assess the performance of the different types of MSRs that have been considered.

Materials and on-line chemistry

A wide range of problems lies ahead in the design of high temperature materials for molten salt reactors. The Ni–W–Cr system is promising. Its metallurgy and in-service properties need to be investigated in further details regarding irradiation resistance and industrialization.

Objectives:

- Progress towards resolving feasibility issues and update priority R&D needs about structural materials for MSRs and on-line or batch-wise spent salt treatment processes.
- Plan for associated experiments.

GIF Definitions

AF	advanced fuel (SFR signed Project)
CDBOP	component design and balance-of-plant (SFR signed Project)
CD&S	conceptual design and safety (GFR signed Project)
CMVB	computational methods validation and benchmarking (VHTR Project)
EG	experts group
EMWG	economic modeling working group
FA	Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy System
FCM	fuel and core materials (GFR Project)
FFC	fuel and fuel cycle (VHTR signed Project)
FQ	fuel qualification test (SCWR Project)
GACID	global actinide cycle international demonstration (SFR signed Project)
GIF	Generation IV International Forum
GFR	gas-cooled fast reactor
HP	hydrogen production (VHTR signed Project)
ISAM	integrated safety assessment methodology
LFR	lead-cooled fast reactor
M&C	materials and chemistry (SCWR Project)
MAT	materials (VHTR Project)
MSR	molten salt reactor
MWG	methodology working group
PA	project arrangement
PG	policy group
PMB	project management board
PRPPWG	proliferation resistance and physical protection working group
RSWG	risk and safety working group
SSC	system steering committee
SCWR	supercritical-water-cooled reactor
SFR	sodium-cooled fast reactor
SIA	system integration and assessment (SFR Project)
SIAP	senior industry advisory panel
SO	safety and operation (SFR signed Project)
SRP	system research plan
TH&S	thermal-hydraulics and safety (SCWR signed Project)
TS	technical secretariat
VHTR	very-high-temperature reactor

Organizations

ANRE	Agency for Natural Resources and Energy (Japan)
CAEA	China Atomic Energy Authority (People's Republic of China)
CEA	Commissariat à l'énergie atomique et aux énergies alternatives (France) *(Previously Commissariat à l'énergie atomique)*
CNRS	Centre National de la Recherche Scientifique (France)
DME	Department of Minerals and Energy (Republic of South Africa)
DOE	Department Of Energy (United States)
FZK	ForschungsZentrum Karlsruhe (Germany)
IAEA	International Atomic Energy Agency
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Center (Euratom)
KAERI	Korea Atomic Energy Research Institute
MEST	Ministry of Education, Science and Technology (Republic of Korea)
MOST	Ministry of Science and Technology (People's Republic of China)
NEA	Nuclear Energy Agency (OECD)
NRCan	Department of natural resources (Canada)
NRF	Nation Research Foundation (Republic of Korea)
NRI	Nuclear Research Institute (Czech Republic)
ORNL	Oak Ridge National Laboratory (United States)
PBMR Pty	Pebble Bed Modular Reactor (Pty) Limited (Republic of South Africa)
PSI	Paul Scherrer Institute (Switzerland)
VTT	Valtion Teknillinen Tutkimuskeskus (Technical Research Center of Finland)

Others

AHTR	advanced high-temperature reactor
ALISIA	Assessment of LIquid Salts for Innovative Applications
ANTARES	AREVA New Technology based on Advanced gas-cooled Reactors for Energy Supply
AVR	Arbeitsgemeinschaft Versuchsreaktor
BWR	boiling water reactor
CFD	computational fluid dynamics
CRP	coordinated research program
DHR	decay heat removal
ELSY	European Lead-cooled SYstem
EROS	Experimental zeRO power Salt reactor
ESFR	example sodium fast reactor
GTHTR300C	gas turbine high temperature reactor 300 for cogeneration
GT-MHR	gas turbine-modular helium reactor

HPLWR	high performance light water reactor
HTR-PM	high temperature gas-cooled reactor power generating module
HTR-10	high temperature gas-cooled test reactor with a 10 MWth capacity
HTTR	high temperature test reactor
IHX	Intermediate Heat eXchanger
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
ISTC	International Science & Technology Center
LOCA	loss of coolant accident
LWR	light water reactor
MA	minor actinides
MSFR	molten salt fast reactor
NGNP	new generation nuclear plant
NHDD	nuclear hydrogen development and demonstration
ODS	oxide dispersion-strengthened
PBMR	pebble bed modular reactor
PHWR	pressurized heavy water reactor
PP	physical protection
PR	proliferation resistance
PWR	pressurized water reactor
PYCASSO	PYrocarbon irradiation for Creep And Shrinkage/Swelling on Objects
R&D	research and development
SA	system arrangement
SCC	stress corrosion cracking
SCW	super-critical water
SGTR	steam generator tube rupture
SSTAR	small secure transportable autonomous reactor
THTR	thorium high temperature reactor
TRISO	tristructural isotopic (nuclear fuel)
TRU	transuranic

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This fourth edition of the *GIF Annual Report* highlights the main achievements of the Forum in 2010. The ten active members are continuing their efforts to develop the six most promising concepts for the future generation of nuclear power as selected when the Forum was established. Two additional R&D Project Arrangements became effective in 2010, increasing the total to ten. Two Memorandum of Understanding were also signed to facilitate co-operation on the lead-cooled fast reactor (LFR) and molten salt reactor (MSR) systems.

AEN NEA Printed by the OECD Nuclear Energy Agency for the Generation IV International Forum