# GEN IV International Forum℠

## 2009 Annual Report

# Message from the Chairman

It is a great honor for me to present the Generation IV International Forum (GIF) 2009 annual report to the nuclear community. I think that the GIF achievements presented in the document are quite commendable. When the Charter was signed in 2001 nine countries constituted the GIF; the number of members is now thirteen, including twelve countries and one international entity. After a decade of cooperation between its partners, GIF now has reached the stage of undertaking collaborative projects addressing specific R&D issues.

Recently, the potential role of nuclear power to enhance security of energy supply and alleviate the risk of global climate change has been recognized more broadly by policy makers and economic actors. Many national and international activities are aiming at launching new nuclear programs or reviving programs previously on standby. In this context, GIF offers a unique framework to strengthen international cooperation for the development of advanced nuclear systems.

Within GIF we are supporting the development of Generation IV systems responding to the requirements of the 21st century and meeting the GIF goals regarding proliferation resistance, sustainability, safety and reliability, and economics. Some concrete steps towards the development of Generation IV reactors were achieved in 2009 and early in 2010 including the first criticality of the experimental fast reactor in China and the re-start of the prototype fast reactor "MONJU" in Japan. Also, the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) project in France and the Japanese Sodium Fast Reactor (JSFR) project in Japan have progressed steadily, and demonstration-scale fast reactor facilities are expected to be in operation in the 2020s.

In 2009, the Russian Federation acceded to the GIF intergovernmental Framework Agreement and is ready now to participate actively in collaborative research programs. During the year several milestones were achieved including the signature of four Project Arrangements (PA) on: safety and operation for the Sodium Fast Reactor (SFR); materials for the Very-High-Temperature Reactor (VHTR); thermal hydraulics and safety for the Super-Critical-Water Reactor (SCWR); and conceptual design and safety for the Gas Fast Reactor (GFR). Furthermore, significant progress was made towards signing PAs for system integration and assessment in each system. Projects on system integration and assessment will be essential to monitor R&D results in the framework of the overall GIF objectives and goals.

The first GIF Symposium, held in Paris, France, in September 2009, was attended by nearly 200 participants who shared feedback from experience and key results obtained for the six GIF systems and in the field of horizontal activities on methodologies for assessing their economics, safety, and proliferation resistance and physical protection. The proceedings from the symposium show that the collaboration led to very fruitful outcomes (see www.gen-4.org/GIF/About/documents/GIFProceedingsWEB.pdf).

Besides the GIF activities, many other international events and national achievements are contributing to the development of Generation IV nuclear systems. An international conference on fast reactor (FR09), an option largely represented within GIF, was held in Kyoto, Japan, in December. Organized by the International Atomic Energy Agency (IAEA), this conference aimed at enhancing exchange of information on programs, operating experience and development achievements of fast reactors in order to identify and discuss critical issues related to fast reactors and their fuel cycles, and eventually to promote international cooperation and R&D in the field. Nearly seven hundred participants from twenty countries, including some countries previously not highly interested in the fast reactor option, and four international organizations attended the conference. The success of this conference, which had not been held in the last 18 years, demonstrates the renewed interest for fast reactors worldwide.

As I mentioned earlier, the development of Generation IV reactors is moving ahead steadily and I am sure that GIF will continue to promote cooperative R&D projects in support of this development. I would like to conclude by thanking all GIF members for their efforts and their valuable contributions to the steady progress of many projects. I hope to meet your expectations through my chairmanship of this unique international endeavor.

*Yutaka SAGAYAMA*

GIF Chairman – July 2010

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# CHAPTER 1 INTRODUCTION

During their last meeting held in L'Aquila (Italy) July 8-10, 2009, the Leaders of the Group of Eight (G8) agreed on the statement that "a growing number of countries have expressed interest in nuclear power programs as a means to address climate change and energy security concerns. In the opinion of these countries, nuclear energy can play an essential role, as it meets the dual challenge of reducing greenhouse gas emissions and lowering fossil-fuel consumption. (...) [The G8] promotes international collaboration at all levels, including cost-benefit analysis, research, infrastructure and human resources development, plant construction, operation, decommissioning and waste management, in order to ensure the highest technically available safety and security standards and accelerate further development and deployment of innovative technologies. (...) [They also] call on all countries interested in the civil use of nuclear energy to engage in constructive international cooperation".

Created in 2000, especially to foster international collaboration at the detailed level of actual R&D, the Generation IV International Forum (GIF) can and is playing a predominant role in addressing this most challenging issue.

GIF is a cooperative international endeavor organized to develop the research necessary to test the feasibility and performance capabilities of fourth generation (Generation IV) nuclear systems with the goal of making such systems deployable in large numbers by 2030, or earlier. In 2009, the GIF cooperative organization, which already included nine active members (Canada, Euratom, France, Japan, People's Republic of China, Republic of Korea, Republic of South Africa, Switzerland and United States), expanded through the accession of the Russian Federation to the GIF Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy System (FA).

The goals of fourth generation nuclear plants are to improve:

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

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

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

d) proliferation resistance and physical protection.

GIF members have already selected six generic systems for further R&D: the Gas-cooled Fast Reactor (GFR); the Lead-cooled Fast Reactor (LFR); the Molten Salt Reactor (MSR); the Sodium-cooled Fast Reactor (SFR); the Super-Critical Water Reactor (SCWR) and the Very High Temperature Reactor (VHTR). Effective collaboration on the various systems expanded in 2009 with the signature by the People's Republic of China of the SFR System Arrangement (SA) and the signature by System Steering Committees of SCWR, SFR and VHTR of three additional Project Arrangements (PA).

Regarding GIF governance, Yutaka Sagayama (JAEA, JAP) became the new Chair of the GIF Policy Group, the governing body of the Generation IV International Forum in December 2009. Yutaka Sagayama succeeded Jacques Bouchard (CEA, FRA) who chaired the Policy Group from 2006 to 2009. The two vice-chairmen of the Policy Group are Christophe Béhar (CEA, FRA) and Peter Lyons (DOE, USA). The new Policy Director is Pascal Anzieu (CEA, FRA) replacing Massimo Salvatores, and Harold McFarlane (INL, USA) is the new Technical Director following Ralph Bennett who has been in this position since the creation of GIF.

As a result of progress made and the desire to communicate the general thrusts to a wider audience, the first *GIF International Symposium* was organized in 2009, in Paris, France. The symposium, held in parallel with the GLOBAL 2009 meeting, provided the opportunity to present to the scientific and industrial communities an overview of both the collaborative framework and the main achievements of GIF. A common session, open to GLOBAL participants, provided the nuclear community an overview of the work performed within GIF, while status and progress reports were given to the 200 registered GIF Symposium participants through 25 presentations. The proceedings of the meeting are available on the GIF public website (www.gen-4.org).

This 2009 Annual Report is the third annual report issued by GIF. It includes three chapters in addition to this introduction plus 4 appendices, as follows:

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 2009. 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 next 5 years.

Appendix 3 reproduces the Table of Contents of the Proceedings from the GIF Symposium held in Paris (France) in 2009.

Appendix 4 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 a wealth of 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[1] 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

| 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 (CEA) | 02/2005 | 11/2006 | | 02/2006 | 11/2006 |
| Japan (JAP) | Agency for Natural Resources and Energy (ANRE)<br>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)<br>Ministry of Science and Technology (MOST) | 12/2007 | | | 03/2009 | 10/2008 |
| Republic of Korea (KOR) | Ministry of Education, Science & Technology (MEST)<br>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 | | | | |
| 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 |

1. 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.

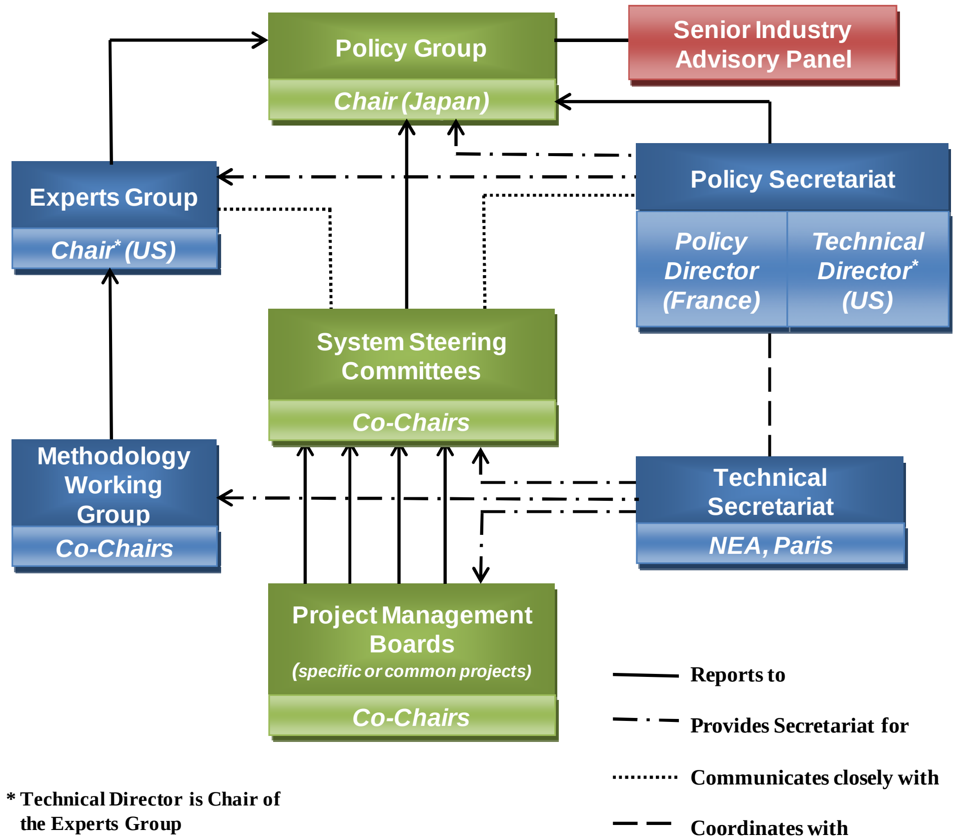
Argentina, Brazil and the United Kingdom[2] have signed the GIF Charter but did not accede to or ratify the FA; accordingly, within the GIF, they are designated as "non-active Members".

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 as of 1 January 2010

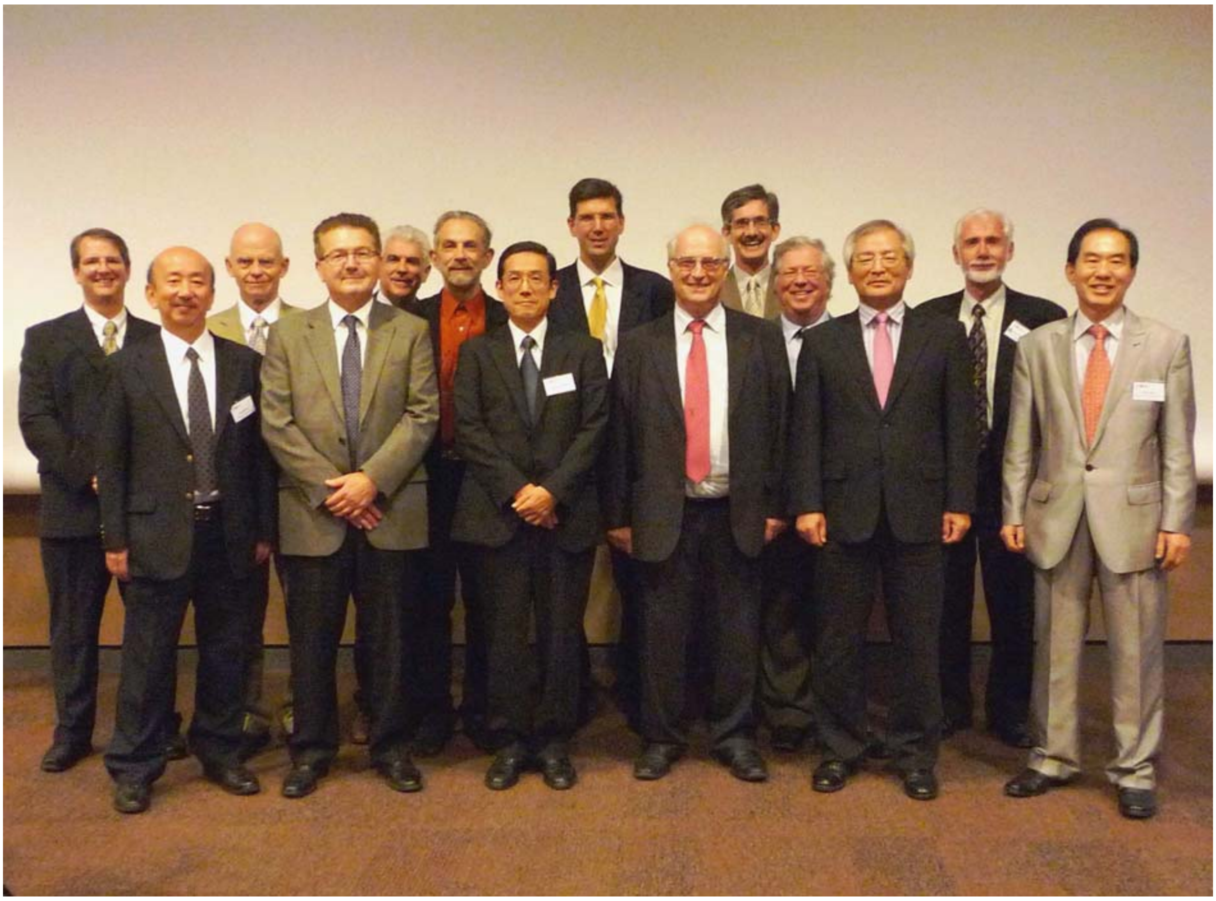



2. The United Kingdom participates in GIF activities through Euratom.

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.

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.

Figure 2-2: Policy Group in Paris (September 2009)



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

The GIF Charter and Framework Agreement allow for the participation of organizations from public and private sectors of non-GIF Members in PAs and in the associated PMBs, but not in SSCs. Public and private sector organizations, including those from non-GIF Members, may join any PA, but 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. The Policy Secretariat assists the Policy Group and Experts Group in the fulfillment of their responsibilities. Within the Policy Secretariat, the Policy Director assists with the conduct of the Policy Group whereas the Technical Director serves as Chair of the Experts Group and assists the Policy Group on technical matters. The Technical Secretariat (TS), provided by the Nuclear Energy Agency (NEA) of the 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 1 March 2010, System Arrangements have been signed by several Members for four systems (GFR, SCWR, SFR and VHTR). For the LFR and the MSR, collaborative R&D is currently pursued by interested Members under the auspices of provisional SSCs.

Four Project Arrangements (PAs) have been signed within the SFR system, and are effective: the Advanced Fuel (AF) PA; the Global Actinide Cycle International Demonstration (GACID) PA; the Component Design and Balance-Of-Plant (CDBOP) PA; and the Safety and Operation (SO) PA. Within the VHTR system, three PAs have been signed: the Fuel and Fuel Cycle (FFC) PA; the Hydrogen Production (HP) PA; and the Material (MAT) PA which was signed on September 2009 but for which validation by all members of the SSC is still pending before it can become effective. For the GFR system, the Conceptual Design and Safety (CD&S) PA was signed in September 2009 and is now effective. For the SCWR system, the Thermal-Hydraulics and Safety (TH&S) PA was signed on October 2009 and is effective. Several projects are in the process of signature, and others 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 1 March 2010.

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.

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.

Table 2-2: Status of signed arrangements and provisional cooperation within GIF as of December 2009

| | 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 | | P | P | P | P | | P | P | | P | P |
| CMVB Project | | | P | | P | P | P | P | | | P |
| **SFR SA** | | | X | X | X | X | X | | O | | 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 | | P | P | P | P | O | 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 | | | | | | P |
| **MSR System** | | | P | P | | | | | O | | P |

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 |

## 3.1 Systems

The main results obtained in 2009 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 2009, 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 2009. More detail on scientific and technical aspects of the systems may be found in conference papers and journal articles listed in the bibliography provided at the end of the chapter.

### *3.1.1 Very-High-Temperature Reactor (VHTR)*

#### Main characteristics of the system

The VHTR is defined by its technical features using fully ceramic coated-particle fuel, graphite for neutron moderation and helium as coolant and by its innovative applications beyond dedicated electricity production (e.g. cogeneration of electricity and heat and/or hydrogen). The temperature requirements and the power level of the VHTR are derived from the industrial processes to be coupled to the nuclear heat source. Modular arrangement of VHTR enhances the overall reliability – as well as the robustness, economics and safety features – of the reactor system.

Due to the high coolant gas temperature of around 800°C (or higher), VHTRs have the unique potential to adopt current conventional power plant and process technologies, thereby opening the way for substituting conventional boilers and burning of fossil fuel in a spectrum of large scale industrial processes (e.g. refineries, petrochemistry, coal liquefaction, steel making, etc.). This can contribute to substantial conservation of fossil fuel and reduction in $CO_2$ emissions.

Efficient and $CO_2$-free production of hydrogen is an additional but challenging target of VHTR. Currently, most hydrogen is produced by steam reforming, which converts and consumes natural gas. Steam reformers or direct water splitting processes like thermo-chemical, electro-chemical or hybrid processes can be coupled with VHTR to further reduce or eliminate $CO_2$ emissions during production of hydrogen for use in manufacturing synfuel, fertilizers, chemicals or steel.

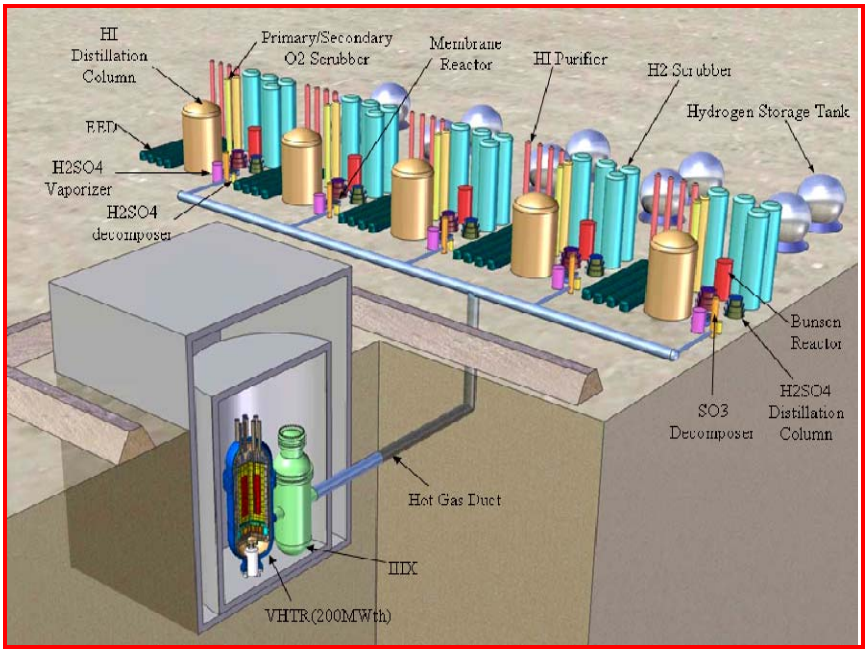
The basic technology for the VHTR has been established in former high-temperature gas reactors such as the US Peach Bottom and Fort Saint-Vrain plants as well as the German AVR and THTR prototypes. The technology is being advanced through near and medium-term projects, such as HTR-PM, PBMR, GTHTR300C, ANTARES, NHDD, GT-MHR and NGNP, led by several plant vendors and national laboratories respectively in the People's Republic of China, the Republic of South Africa, Japan, France, the Republic of Korea and the United States. The construction of a two-module HTR with pebble bed core (HTR-PM) has been started in China. 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. High quality steam of 566°C will be fed into a common steam header.

Experimental reactors such as 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 to improve 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.

Two baseline concepts are available for the VHTR core: the pebble bed-type and the prismatic block-type. The fuel cycle will initially be once-through with low-enriched uranium fuel and very high fuel burn-up. Solutions will be developed to adequately manage the back-end of the fuel cycle and the potential for a closed fuel cycle will be 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.

The electric power conversion unit for VHTR may initially be an indirect steam Rankine cycle applying the latest technology of conventional power plants. Direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles are perceived as longer term options. Other process heat applications will need intermediate heat exchangers to separate the nuclear island from the process side (see Figure 3-1) for non-electric industrial applications. The available high-temperature alloys used for heat exchangers and metallic components determine the current temperature range of VHTR (~800-950°C). In the future, innovative materials like new super alloys, ceramics and compounds may allow coolant gas temperatures up to about 1 000°C.

Figure 3-1: Arrangement of the VHTR and hydrogen production system




## 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 in 2010.

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 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.

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, Switzerland and the United States by 16 September 2009.[3] It should be noted that 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 Computational Methods, Validation and Benchmarking PA will be finalized soon, and should be ready for signature in early 2010.

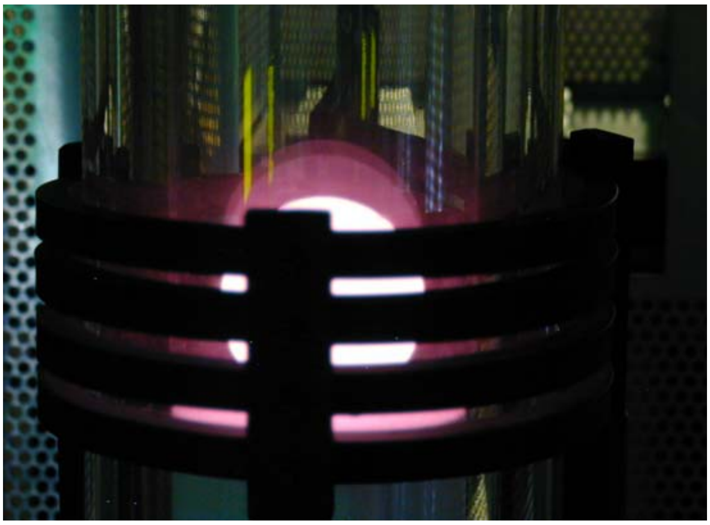
Two other projects – on components and high-performance turbomachinery and on design, safety and integration – are still being discussed by the VHTR SSC but the associated research plans and project arrangements have not been developed yet for those two areas.

### R&D Objectives

The VHTR development approach builds on technologies already used for former and current carbon-dioxide cooled and graphite-moderated reactors that have been successfully operated in the United Kingdom and other countries over several decades. Advanced Gas-cooled Reactors (AGR) operate at maximum gas outlet temperatures up to 650°C. Further increases in operational temperatures (750-780°C) have been demonstrated by commercial-sized High-Temperature Reactors (THTR, Fort Saint-Vrain) by using helium as a coolant. Figure 3-2 illustrates the material conditions, showing a bright glowing graphite pebble heated to 800°C, in an induction furnace. It is obvious that further enhancement of operational temperatures poses considerable requirements especially on the metallic materials and components.

VHTR development is driven not only by the achievement of "very" high temperatures providing higher thermal efficiency for new applications, but also by demonstration of "very" reliable (inherent) safety features, "very" high fuel burn-up and "very" long operational lifetime (more than 60 years), with potential for conflicts among those challenging R&D goals.

Figure 3-2: Graphite pebble heated to 800°C (Photo by FZJ, Germany)



3. As of December 2009, all signatures received, but formal agreement by all SA Signatories still pending to declare MAT PMB effective.

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. The SRP is structured into ***six projects***:

1. ***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.

2. ***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 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.

3. ***Fuel and fuel cycle*** investigations are focusing on the performance of the TRISO coated particles, which are the basic fuel concept for the VHTR. R&D will increase the understanding of standard design $UO_2$ kernels with SiC/PyC coating and examine the use of UCO kernels and ZrC coatings for enhanced burn-up capability, reduced fission product permeation and increased resistance to core heat-up accidents (above 1 600°C). This work will involve fuel characterization, post irradiation examination, safety testing, fission product release evaluation, as well as assessment of chemical and thermo-mechanical materials properties in representative service and accident conditions. R&D will also examine spent-fuel treatment and disposal, including used-graphite management, as well as the deep-burn of plutonium and minor actinides (MA) in support of a closed cycle.

4. ***Materials*** 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, will require 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.

5. ***Components*** need 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 addresses 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.

6. For ***Hydrogen production***, two main processes 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 will address 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 will be examined in terms of technical and economic feasibility in dedicated or cogeneration hydrogen production modes, aiming at lowering operating temperature requirements in order to make them compatible with other Generation IV systems.

## Milestones

The major milestones defined in VHTR System Research Plan are:

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

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

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

- Hydrogen Production Project
  - ✓ Sulfur/Iodine process
    - 2009 Laboratory-scale test and optimization
  - ✓ High temperature electrolysis
    - 2008 Laboratory-scale integrated experiment
    - 2014 Pilot-scale experiment
  - ✓ Alternative processes
    - 2009 Screening and technical evaluation
    - 2010 Evaluation of economics
  - ✓ Coupling technology
    - 2010 Process evaluation and component technology
  - ✓ Milestones for the hybrid copper-chloride cycle and the hybrid sulfur cycle are being developed for inclusion in the revised project plan.

## Main activities and outcomes

### *Fuel & Fuel Cycle*

During 2009, the first Work Plan was established covering the period 2009-2010, which identifies deliverables largely associated with the Irradiation and Post-Irradiation Examination (PIE), and the Fuel Attributes and Material Properties Work Packages. In regards to irradiation activities, the PYCASSO-I (PYrocarbon irradiation for Creep And Shrinkage/Swelling on Objects, EU 6$^{th}$ Framework Program) irradiation started on 18 April 2008, and completed in 2009. PIE is anticipated in 2010. The PYCASSO-II experiment, which will have a higher fluence, up to $3\times10^{25}$ $n.m^{-2}$, began irradiation in 2009. Nine cycles of irradiation are anticipated. The final design of the AGR-2 experiment being planned for the Advanced Test Reactor was completed along with fuel fabrication activities in the US, France and South Africa, with an anticipated irradiation starting in the Spring of 2010. Because of technical problems with safety instrumentation and extended High Flux Reactor (HFR) downtime, the HFR-EU1 irradiation will continue on into 2010. HFR-EU1 consists of 3 GLE4 pebbles and 2 pebbles produced by INET. Work on the back end of the fuel cycle and the transmutation potential of VHTRs continues in the EU and the US. Members of this PMB have completed a large amount of work related to benchmarking of fuel performance models under normal and accident conditions and benchmarking of quality control techniques, which is nearing completion as part of the larger IAEA CRP6 activity.

### *Materials*

The most important event in 2009 was the subscription of the Materials PA by several signatories. Significant improvements were made with respect to the Generation IV Materials Handbook. Successful first upload trials revealed no further problems. It is planned to upload in the near future the archive deliverables from 2007 through 2009. A screenshot from the user interface is depicted in Figure 3-3.

The drastic change of the operational parameters of current VHTR projects (e.g. US-NGNP) and political decisions in some countries (e.g. France) to reduce efforts for advanced VHTR materials projects had some effects on the Materials PA already in 2009. Of the three working groups (graphite, metals and ceramics), the coordination of work within the graphite group is most advanced. Working group meetings took place in 2009 in Centurion (19-20 February) and in Idaho Falls (1-2 October).

Progress has been made in a number of task areas. Discussions concerning format and content of inputs into the Materials Handbook will enable a smooth transfer of data. Work in the metals group is progressing though not as rapidly as for graphite. The regular participation of a representative from the American Society of Mechanical Engineers (ASME) at the PMB meetings led to starting an exchange between code-related research needs and R&D groups. It is planned to improve this relationship further in the future. This will certainly help tune the research activities coordinated in the PMB according to current needs. In addition, there remains space for less applied and more basic research, for example in the field of materials modeling. The level of activity within the ceramics working group was still low in 2009. This was mainly due to the fact that the interest in ceramic solutions for the control rod diminished as a result of the reduced gas temperatures. However, ceramics will remain important in the materials project, as insulation materials and for Gas-cooled Fast Reactor applications.

Figure 3-3: User interface of the Generation IV Materials Handbook
(Data pool for materials data generated in the VHTR Materials PMB)


### *Hydrogen Production*

During 2009 the VHTR Hydrogen Production (HP) Project continued to compile the results obtained to date and provided to the project by the member countries. Working groups of technical experts were proposed to focus cooperative efforts on specific topics.

Evaluation of the Sulfur-Iodine (S-I) thermochemical cycle for $H_2$ production progressed through efforts of several members providing flowsheet analyses of the S-I cycle. These analyses are planned to be synthesized into a combined overview of the state of the art by the end of 2010. Benchmark exercises on a reference flowsheet are also planned to be performed. Materials screening and development activities were conducted involving membranes and adsorbents for separations, and catalysts for $SO_3$ and HI decomposition. Interested members have progressed to performance of component and closed-circuit bench-scale experiments at full temperature, pressure and flux rates to define and evaluate key parameters such as thermodynamic properties and rate constants.

High Temperature Electrolysis (HTE) splits water in a device very similar to a solid oxide fuel cell (SOFC), and the results of several national programs for electricity production from fuel cells are being monitored to ensure that the progress in SOFC technology provides key developmental data for the HTE program. Modeling activities for the HTE process have included optimizing system design for various plant configurations, examination of cogeneration options, and analyses of performance of cell configurations. Tests of button cells and small stacks of "standard" cells were conducted to investigate performance and longevity issues. Current efforts are focused on identifying the causes of cell degradation and performing tests of small stacks of cells. Three members are actively pursuing advancements in electrode materials, cell interconnect technologies, leak management solutions, and optimized operating conditions.

Evaluation of the many alternative hydrogen production cycles available has focused the project's interest on two additional cycles: the Hybrid Copper-Chloride (Cu-Cl) cycle and the Hybrid Sulfur (HyS) cycle. Other cycles are being pursued as well, but to a lesser degree. Additionally, tasks involving economic evaluation of the various hydrogen production processes coupled to nuclear reactors are planned to be performed in conjunction with the Economic Modeling Working Group.

The final area of collaboration being pursued under the HP project regards analysis of the issues encountered when coupling hydrogen production processes to a nuclear reactor. Factors being considered are design-associated risk analysis, safety (including tritium abatement), and system integration. Performance calculations for interactions between the reactor and hydrogen plants are being evaluated in steady state to be followed by dynamic simulations. Work is beginning on coupling component technologies, such as process heat exchangers, high temperature isolation valves, hot fluid ducting, and a thermal load absorber.

### *Computational Methods Validation and Benchmark*

Computational Methods Validation and Benchmark (CMVB) focuses on ensuring 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. In general, the computational methods must be shown to have a computational envelope[4] that encompasses the reactor plant operational and accident envelopes. Within the required calculational envelope, the calculation methods must be validated using accepted practices (Roache 2009; ASME V&V20 Standard) and must be in conformance with the relevant quality standards (e.g. ISO 9000 or ASME NQA-1). In addition, the methods' calculational uncertainties must be quantified. Computational tools are used in areas such as thermal-hydraulics, structural mechanics, core physics, chemical transport, and may be used together via some form of coupling to calculate scenarios that require multiphysics solutions.[5] Of these, some numerical models are presently under development (e.g. pebble-bed and prismatic reactor physics). However, the development of many numerical models is considered ready for validation. Validation of the completed numerical models and software tools is underway within the CMVB organizations.

Validation of the numerical models and software that are the focus of the CMVB organizations will 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. Within the context of accomplishing Items (c) and (d), validation data sets that are found to meet the desired data standard may be defined as "standard problems" and will be identified as such by the CMVB organizations.[6] Some of the validation studies performed under Item (d) will be "standard problem" analyses performed by a number of organizations. Standard problem validation studies may be based on either experimental data sets or code-to-code comparisons, depending on the validation objectives and the availability of validation data. The experimental data sets range from those describing basic phenomena to integrated experiments including presently operational experiments such as the 30 MWth High Temperature Test Reactor (HTTR), HTR-10 or

4. The computational envelope for a numerical model is the domain where the numerical model may be used to calculate a phenomena or fluid/heat transfer behavior with confidence (with a quantifiable calculational uncertainty). Thus, a computational envelope is the domain where the numerical models are validated.
5. Examples of coupling include reactor physics numerical models coupled to thermal-fluids models.
6. Standard problems are distinguished from validation analyses in that standard problems are: (i) performed by a group of people or organizations using a set of rules, practices, and procedures that all the standard problem participants must follow, (ii) the analyses performed by the standard problem participants are submitted to a standard problem monitor for evaluation and compilation, and (iii) a report is generally issued that compares and discusses the standard problem findings. In contrast, a validation analysis is generally performed by a single analyst or group to achieve a predefined goal for a specific organization.

vintage experiments such as the Arbeitsgemeinschaft Versuchsreaktor (AVR) or plant data from the Fort St.-Vrain plant.

The work of interest to the PMB in these areas is distributed within six work packages (WP) as noted in Table 3-1.

Table 3-1: List of CMVB Work Packages (WP) and WP Lead Organizations

| WP No. | WP Title | Lead |
|---|---|---|
| 1 | Phenomena identification and ranking table (PIRT) methodology | KAERI |
| 2 | Computational fluid dynamics (CFD) | NGNP (DOE) |
| 3 | Reactor core physics and nuclear data | PBMR |
| 4 | Chemistry and transport | PBMR |
| 5 | Reactor and plant dynamics | NGNP (DOE) |
| 6 | High-temperature test reactor | JAEA |

### *3.1.2 Sodium-cooled Fast Reactor (SFR)*

#### Main characteristics of the system

Sodium-cooled Fast Reactor (SFR) nuclear energy systems are among the six candidate technologies selected in the Generation IV Technology Roadmap for their potential to meet the Generation IV technology goals. The primary missions identified for the SFR are (1) contribution to sustainability, in particular through its capabilities for actinide management, and (2) electricity production.

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

The main characteristics of the Generation IV SFR that make it especially suitable for the missions identified are:

(1) High potential to operate with a high-conversion fast-spectrum core with the resulting benefits of increasing the utilization of fuel resources.

(2) Capability of efficient and nearly complete consumption of transuranics as fuel, thus reducing the actinide loadings in the high level waste with benefits in disposal requirements and potentially in non-proliferation.

(3) High level of safety obtained with the use of active and passive means that allow accommodation of transients and bounding events.

(4) Enhanced economics achieved with the use of high-burn-up fuels, fuel cycle (e.g. disposal) benefits, reduction in power plant capital costs with the use of advanced materials and innovative design options, and lower operating costs achieved with improved operations and maintenance.

Owing to the significant past experience accumulated with sodium-cooled reactors in several countries, the deployment of Generation IV SFR prototype systems is targeted for 2020. Difficulty in achieving enhanced economics with high level of safety is deemed be one of the obstacles to early deployment of SFR.

Eight goals for the Generation IV nuclear energy systems are defined in the four broad areas of sustainability, economics, safety and reliability, and proliferation resistance and physical protection. The broad design requirements for the SFR systems, shown in Table 3-2, are established in order to satisfy the development targets corresponding to the Generation IV goals. Three major options are considered: a large size (600 to 1 500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors (S. Kotake et al., 2008) as shown in Figure 3-4; an intermediate-to-large size (300 to 1 500 MWe) pool-type reactor (Mignot et al., 2008; Joo et al., 2008) 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 (Chang et al., 2005) as shown in Figure 3-6. The outlet temperature is 500-550°C for all three options.

Table 3-2: Major Broad Design Requirements for SFR System

| SFR Design Requirements | | Generation IV Goals | |
|---|---|---|---|
| Actinide management | Breeding ratio: 0.5-1.3* | Sustainability | 1: Resource utilization (1.0-1.3) |
| TRU transmutation | TRU transmutation under fast reactor multi-recycle and long-term storage of LWR spent fuel | | 2: Waste minimization and management |
| Radioactive release | Equivalent to or less than present LWRs | | |
| PR&PP | Excludes pure plutonium state throughout system flow | Proliferation Resistance and Physical Protection | 1: Minimize diversion or undeclared production; reactors have passive features that resist sabotage |
| Safety | Operability, maintainability and reparability | Safety and Reliability | 1: operations will excel in safety and reliability<br>2: very low likelihood and degree of reactor core damage<br>3: eliminate the need for offsite emergency response |
| Electricity generation cost | Cost-competitiveness with other means of electricity production and a variety of market conditions, including highly competitive deregulated or reformed markets ** | Economics | 1: life-cycle cost advantage over other energy sources (Low overnight construction cost, Low production cost) |
| Operation cycle | 18 months, and more | | 2: level of financial risk comparable to other energy project |
| Construction duration | As a goal, large-scale: 42 months, medium-scale modular type: 36 months | | |

* Conversion ratio of 0.5-1.0 might be taken to pursue Sustainability-2: waste minimization and management.

** Bus-bar cost will be evaluated using the methods specified by the Generation IV Economic Methodology Working Group; some expected targets for a First-of-a-kind (FOAK) plant are ~ 4¢/kWh and Construction cost: ~ 2 000 $/kWe. For a future Nth-of-a-kind plant, the cost target is to be competitive with advanced LWR system costs evaluated with a similar technique.

Figure 3-4: Loop-configuration SFR

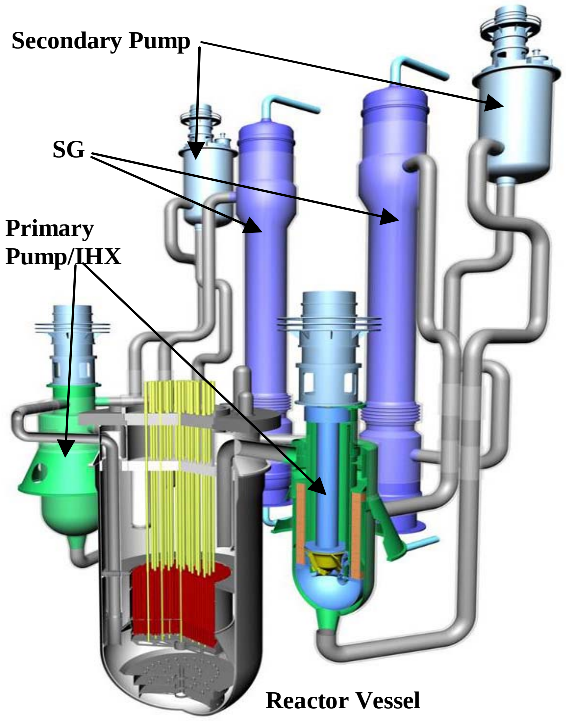



Figure 3-5: Pool-configuration SFR

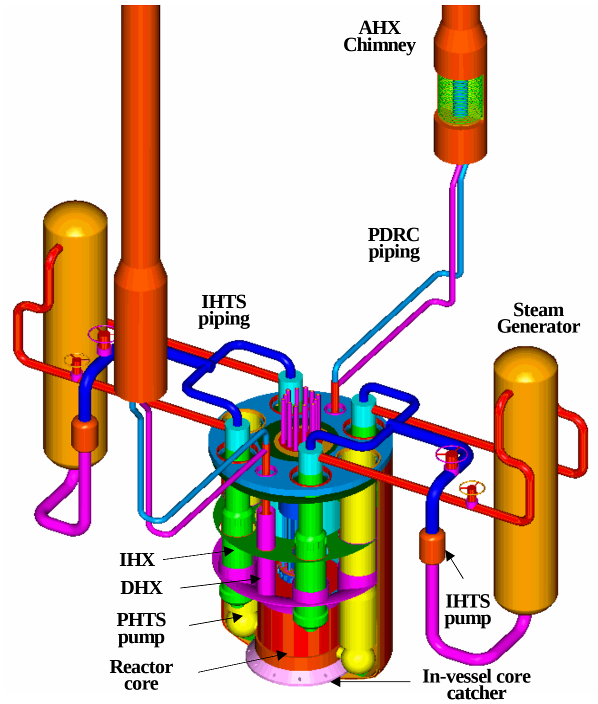
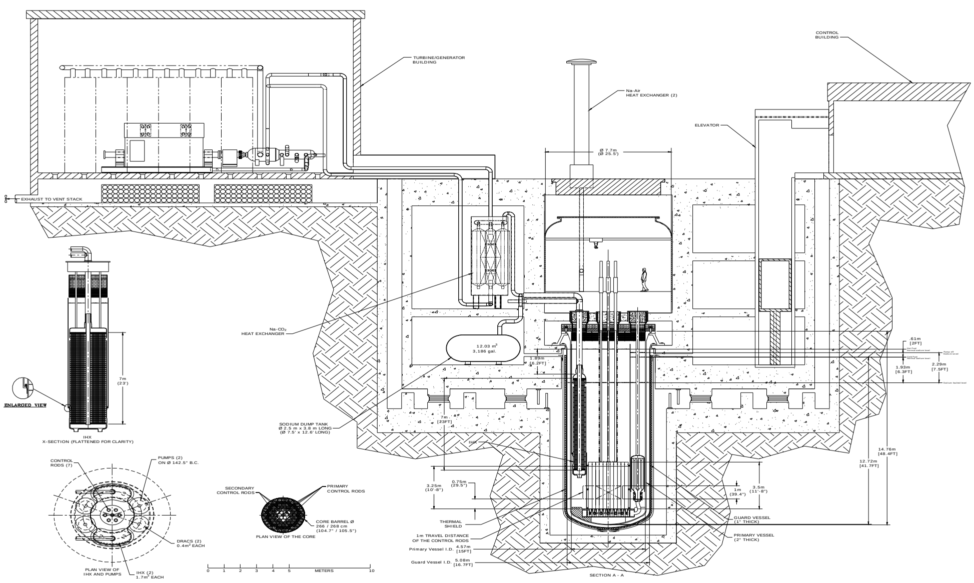



Figure 3-6: Small modular SFR configuration



The design and performance parameters of the three options are illustrated in Table 3-3.

Table 3-3: Design Parameters of Generation IV SFR Concepts

| SFR Design Parameters | Loop | Pool | Small Modular |
|---|---|---|---|
| Power Rating, MWe | 1500 | 600 | 50 |
| Thermal Power, MWth | 3570 | 1525 | 125 |
| Plant Efficiency, % | 42 | 42 | ~38 |
| Core outlet coolant temperature, $^{o}$C | 550 | 545 | ~510 |
| Core inlet coolant temperature, $^{o}$C | 395 | 370 | ~355 |
| Main steam temperature, $^{o}$C | 503 | 495 | 480 |
| Main steam pressure, MPa | 16.7 | 16.5 | 20 |
| Cycle length, years | 1.5-2.2 | 1.5 | 30 |
| Fuel reload batch, batches | 4 | 4 | 1 |
| Core Diameter, m | 5.1 | 3.5 | 1.75 |
| Core Height, m | 1.0 | 0.8 | 1.0 |
| Fuel Type | MOX (TRU bearing) | Metal (U-TRU-10%Zr Alloy) | Metal (U-TRU-10%Zr Alloy) |
| Cladding Material | ODS | HT9M | HT9 |
| Pu enrichment (Pu/HM), % | 13.8 | 14.3 | 15.0 |
| Burn-up, GWd/t | 150 | 79 | ~87 |
| Breeding ratio | 1.0-1.2 | 1.0 | 1.0 |

The SFR closed fuel cycle facilitates management of high-level waste and in particular of plutonium and other actinides. Important safety features of the system include a long thermal response time, a large margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the power conversion system. Water/steam and 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 on electricity markets. In addition, the SFR fast neutron spectrum extends the lifetime of natural resources through using available fissile and fertile materials (including depleted uranium) considerably more efficiently than thermal-spectrum reactors with once-through fuel cycle.

Besides the SFR research and development conducted so far, significant near-term activities include Phenix end-of-life tests, restart of Monju, and start-up of the China Experimental Fast Reactor. Commercial-scale plants are under construction in the Russian Federation and India.

### 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 2009, China joined 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 & Technology of the Republic of Korea.
- The China National Nuclear Corporation of the People's Republic of China.

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.

A new collaboration between GIF SFR and the European Sodium Fast Reactor (ESFR) has been discussed in 2009. This Collaborative Project (CP) addresses key viability and performance issues to support the development of an innovative system for competitive electricity generation in Europe. Contacts at the project level should help to define the contributions from ESFR to the SFR project.

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

Advanced Fuel (AF) Project:

Fuel-related research aims at developing high burn-up MA bearing fuels as well as claddings and wrappers capable of withstanding high neutron doses and temperatures. It includes: research on remote fuel fabrication techniques for fuels that contain MA and possibly traces of fission products as well as performance under irradiation of fuels, claddings and wrappers. Candidates under consideration are: oxide, metal, nitride and carbide (since 2008) for fuels; and Ferritic/Martensitic & Oxide-Dispersion Strengthened (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. The Brayton cycle, e.g., if shown to be viable, would reduce the cost of 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 (MAs:

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 five SFR system R&D projects are given below.

- SIA Project

  ✓ Definition of SFR system options
  2009- Initial specification of SFR system options

  ✓ Assessment of SFR system options
  2010-11 Compilation of self-assessment results for SFR system options
  2011- Solicit economics, PR&PP, and safety self-assessment results using the GIF methodologies as contributions from the concept developers

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

- SO Project

  ✓ R&D for Safety
  2008-9 Preliminary assessment of candidate safety provisions and systems
  2008-12 Performance assessment of safety provisions and systems
  2011-15 Qualification of safety provisions and systems

  ✓ R&D for reactor operation and technology testing
  2008-11 Tasks related to SIA Project
  ✓ Phenix end-of-life program
  ✓ Thermal-hydraulics/general system
  ✓ Feedback from the decommissioning of liquid metal fast reactors

  2008-12 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

  2006-7 Preliminary evaluation of advanced fuels
  2007-10 Evaluation of MA-bearing fuels
  2011-15 High-burn-up fuel behavior evaluation
  2016- Demonstration and application of the selected advanced fuel

- CDBOP Project

  2007- Viability study of proposed concepts
  2007-10 Performance tests for detailed design specification
  2011-15 Demonstration of system performance

- GACID Project

  2007-12 Preparation for the limited MA-bearing fuel irradiation test
  2007-12 Preparation for the licensing of the pin-scale curium-bearing fuel irradiation test
  2007-12 Program planning of the bundle-scale MA-bearing fuel irradiation demonstration

### Main activities and outcomes

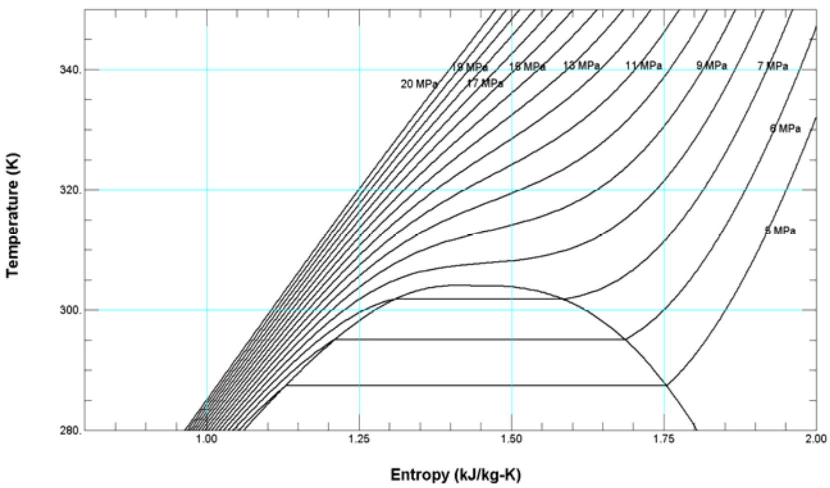
Activities on integration and assessment were conducted through joint meetings of the SFR SSC and the provisional PMB. The SIA approach for integrating and reviewing the Technical Project contributions is complete. A procedure for adding new design concepts to the System Research Plan roster was codified; the proposal by a concept developer must include a self-assessment of performance compared to GIF goals for review by the PMB. A comprehensive list of SFR R&D needs was completed; this list was distributed to the SSC and Technical Projects to help guide future Project scope decisions. Another aspect of the SIA integration role is to act as the technical arm of the SSC to "review" the Technical Projects for consistency and integration with the R&D needs. The need and mechanism for access to Technical Project deliverables must be clarified, but a proposal based on Project participation was developed. Proposed trade study contributions from several members were reviewed. It was agreed that two types of trade studies can be envisioned: 1) scoping studies of the trends and general design precepts, and 2) more detailed quantitative trade studies of particular features in specific Generation-IV SFR designs. The criteria for SIA acceptance will be the usefulness of trade study results as evaluated by the official PMB. The initial trade study contributions will be included in the draft Program Plan and accepted or denied at the first official PMB meeting. The assessment role of the SIA Project will rely on self-assessment contributions that can be solicited from the concept developers. The next step to formalize the SIA Project is to complete the Project Plan; negotiations on the Project Arrangement will start in early 2010.

In the field of safety and operation, the SO Project has been effective since June 2009. Passive safety options were investigated and their feasibility was clarified with respect to inherent features of reactivity feedback, self actuated shutdown system, fusible shutdown system, and various passive decay heat removal systems. R&D on severe accident issues achieved the analysis code development, the evaluation of design measures for prevention and consequence mitigation, and the experimental investigation on molten fuel behavior. Analyses for safety architecture identified the applicability of the proposed safety analysis methodology. Existing reactor testing and experiences produced useful data; i.e. thermal-hydraulic asymmetric test data, core flow test data, In-Service Inspection test and experience data.

In the field of advanced fuels, the AF Project has been effective since March 2007. 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. 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. 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, at the request of SIA project.

Regarding component design and balance-of-plant, the CDBOP PA has been effective since October 2007. In 2009, work progressed on in-service inspection technologies, summarizing and reporting of repair experience, high temperature leak-before-break assessment, and supercritical-$CO_2$ (S-$CO_2$) Brayton cycle advanced energy conversion. In the study of in-service inspection technologies, various sensors were tested to assess their performance for ultrasonic inspection and control of in-vessel structures from outside of the reactor vessel, for under-sodium viewing with an external ultrasonic transducer using a plate-type waveguide sensor, and under-sodium inspection by real time sensing for deformation/displacement detection and high resolution for crack detection. Experience from replacement or modification involving the intermediate sodium circuit in Monju and rules for operating a sodium facility following repair and modification were reported. Material tests including creep-fatigue crack initiation and crack growth for Mod. 9Cr-1Mo (G91) steel for weld metal in addition to the base metal were continued for development of a design code assessment methodology for high temperature leak-before-break assessment. 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. A series of small-scale main supercritical $CO_2$ compressor tests that has been completed in the United States provides data demonstrating stability and controllability of the compressor over the full range of conditions of interest at and near the $CO_2$ critical point, including the supercritical region above the two-phase dome, the liquid-like and vapor-like regions to the side of the dome, and inside the dome (See Figure 3-7 and Figure 3-8).

Figure 3-7: Compression and Control near the Critical Point of $CO_2$




In addition to confirmation of the fundamental compressor performance at and near the critical point, the recent tests address the essential supporting turbomachinery technologies for small-scale demonstration of a complete supercritical $CO_2$ cycle incorporating two compressors in a split-flow configuration and a turbine, by measuring bearing loads, seal leakage rates, and rotor windage losses. Compressor modeling utilized in modeling S-$CO_2$ power cycle plant dynamic behavior and control strategies was compared with data from the small-scale compressor tests and found to be in good agreement. New small-scale S-$CO_2$ compressor testing has been initiated in Japan. Material corrosion/oxidation and carburization testing in high-pressure and high-temperature flowing $CO_2$ was continued, Na-$CO_2$ reaction tests were carried out in a surface interaction mode, and initial performance testing of a new compact diffusion-bonded heat exchanger with interconnected airfoil-shaped channels was carried out. Discussion continued among CDBOP members on collaboration on S-$CO_2$ Brayton cycle research and development.

Figure 3-8: Brayton Cycle compression loop
(In the process of re-assembly at Sandia National Laboratory)

In the field of global actinide cycle, the GACID PA has been effective since September 2007. During the year 2009, activities performed in common by the members included evaluation of MA-bearing fuel material properties, analysis and evaluation of irradiated fuel data, and preliminary program planning for bundle-scale MA-bearing fuel assembly irradiation demonstration in Monju.

### *3.1.3 Super-Critical Water Reactor (SCWR)*

#### Main characteristics of the system

The Super-Critical Water Reactor (SCWR) is a high temperature, high pressure water-cooled reactor that operates above the thermodynamic critical point (374°C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be lumped into two main categories: pressure vessel (PV) concepts proposed first by Japan and more recently by an EU partnership – the European High Performance Light Water Reactor (HPLWR), and; pressure tube (PT) concepts proposed by Canada, generically called CANDU-SCWR. Other than the specifics of the core design, these concepts are considering many common options (e.g. outlet temperatures, fuel based on $UO_2$, thermal neutron spectra, steam cycle options, materials, etc.). Therefore, the R&D needs for the two reactor types are similar. This 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. Significant 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 options for designs 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 (provisional); 3) Thermal-hydraulics and Safety; and 4) Fuel Qualification Testing (provisional). Table 2-2 lists the members and shows the status of these PMBs.

Although not mature yet for construction, the two promising examples of SCWR options, PT & PV, which have been worked out in more detail as described below, highlight the technical challenges and support the SCWR technology projects with constraints and target data. Activities within the joint SCWR project on System Integration and Assessment (SI&A) were placed on hold pending further clarification on tasks within this project. Until such clarification is received, the SCWR System Steering Committee (SSC) will manage the SI&A activities. The member states (i.e. Canada, Euratom, and Japan) agreed to continue work on joint projects focusing on technology development, before a joint prototype development is initiated.

Regarding the Materials and Chemistry project, progress was made in 2009 in the areas of corrosion and stress corrosion cracking testing, surface modification and coatings, and water chemistry (including radiolysis and corrosion product transport). Signing of the Project Arrangement is expected to take place early in 2010.

The Thermal-hydraulics and Safety PMB members established the Project Plan, which is a key component of the Project Arrangement signed by signatories from Canada, Euratom, and Japan effectively on 5 October 2009. The Republic of Korea has been a strong participant in the PMB for the past few years, but has not committed to signing the Project Arrangement due to the suspension of the national R&D program on SCWR. The Project Plan describes the coordinated research activities in the technology development areas identified in the System Research Plan (SRP). Most identified activities are performed on an individual basis, but some may require integrated efforts of all or some participants. Each participant has completed a considerable amount of work for these activities in 2009. Their contributions are summarized below.

In 2009 the SCWR member states Canada, Euratom and Japan started to work out another joint project plan for testing a small-scale fuel assembly cooled with supercritical water in a critical arrangement. Future cooperation on this Fuel Qualification Testing project is envisaged.

In October 2009, France announced that, due to the evolution of its activities towards a focus on Sodium and Gas systems, a decision was taken to stop participation in the SSC but to continue work on radiolysis studies.

### R&D Objectives

The following critical-path R&D projects have been identified in the SRP:

- System integration and assessment – Definition of a reference design, based on the PT and PV designs, that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance. This work will involve identification of an achievable outlet temperature based on materials and fuel performance, as well as linkages to proven super-critical steam cycles in fossil-fired power plants.
- 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 in-core and out-of-core components of both PT and PV 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 behavior.
- 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 an SCWR has never been operated, 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 in 2009

### *System Integration and Assessment (Canada, EU, Japan, observers: R. Korea, China)*

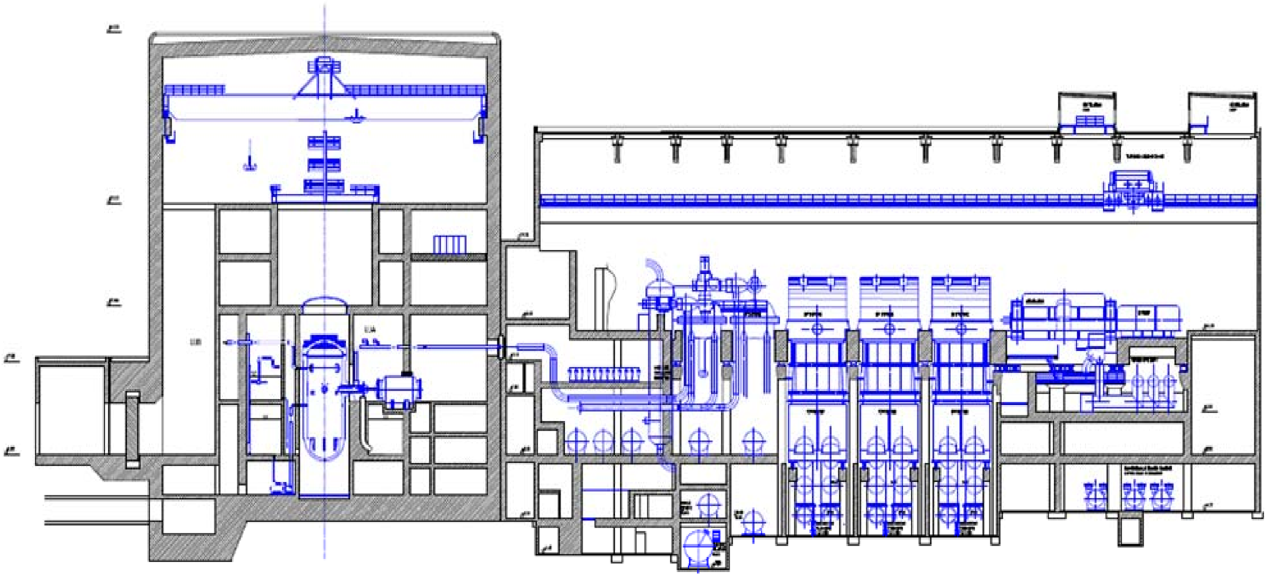
Significant progress has been achieved on the design of the HPLWR. Initiated in 2006, this year marked the third year of the project. The thermal core design with 2 300 MW thermal power has been improved according to the latest suggestions to flatten the power distribution and to stabilize the moderator flow. Neutronic analyses of the HPLWR core have been carried out comprising power distribution of a fresh core with uniform enrichment and an equilibrium core and burn-up predictions including reactivity feedback coefficients. In addition, thermo-hydraulic analyses have been carried out (i.e. sensitivity analyses of disturbances influencing power distribution together with hot channel analyses and predictions of the hottest fuel rod). The coolant flow and temperature in the gaps between assemblies have been calculated. Structural bowing analyses of the assembly boxes have been carried out to identify pad locations at the outer box walls. Finally, residual heat removal with low coolant mass flow (e.g. during core disassembly) has been investigated.

Even though the core design with its multiple coolant heat up with intermediate mixing is certainly a challenging approach, the latest neutronic, thermal-hydraulic and mechanical analyses confirm its feasibility.

The design of the containment is based on those of typical 3rd generation boiling water reactors. It features a pressure suppression pool, 4 core flooding pools, 4 active low pressure coolant injection systems, and 8 depressurization systems into the core flooding pools. A passive coolant injection system still requires further optimization. The balance of plant includes a supercritical-pressure steam cycle with a start-up system to allow full pressure operation in the entire load range. High pressure, medium pressure and 3 low pressure turbines, each with dual flood, condensers, 7 preheater stages, a feedwater tank and feedwater pumps, have been dimensioned to estimate size and costs. The layout of the turbine building has been completed and the size and location of all major components have been identified. Together with the additional layout of the nuclear island shown in Figure 3-9, the power plant design is now complete and remains to be assessed with respect to the criteria of the Generation IV International Forum.

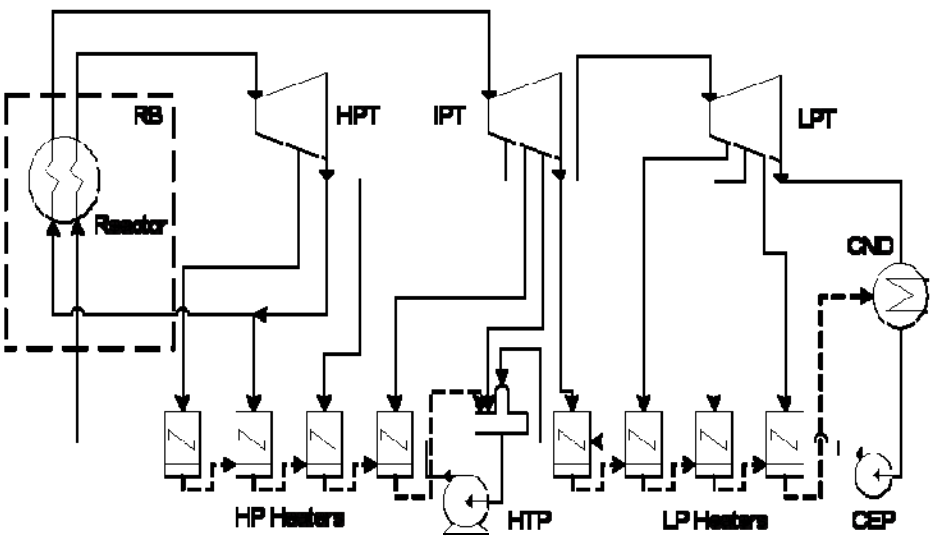
In Japan, the design of the Super Fast Reactor, a PV type SCWR with a fast neutron spectrum, has been improved to a higher core power density of 300 MW/m$^3$. The thin fuel rods of only 5.5 mm outer diameter, arranged in a hexagonal array, form seed assemblies with 271 rods each. They are mixed with blanket assemblies using ZrH layers to improve neutron leakage and, therefore, lower the coolant void worth. The core with around 1 600 MW thermal power and more than 500°C outlet temperature is suitable to burn minor actinides while producing new plutonium with a conversion ratio of more than 96%.

Figure 3-9: Layout of the reactor and turbine building of the HPLWR power plant



In Canada, a reheat-channel option is being considered for the CANDU SCWR to improve the cycle efficiency. This technology has already been implemented into fossil-power plants and can be adopted in the CANDU SCWR. Figure 3-10 presents the typical layout and thermal cycle. The majority of the steam from the high-pressure turbine is directed to specific reheat channels in the reactor core for superheating before passing to the intermediate-pressure turbine. This option is extremely economic and would increase the cycle efficiency to about 50% (which is close to 40% greater than current reactor designs).

Figure 3-10: Typical supercritical CANDU reactor layout and thermal cycle



*Materials and Chemistry (Canada, EU, Japan and France, observers: R. Korea, China)*

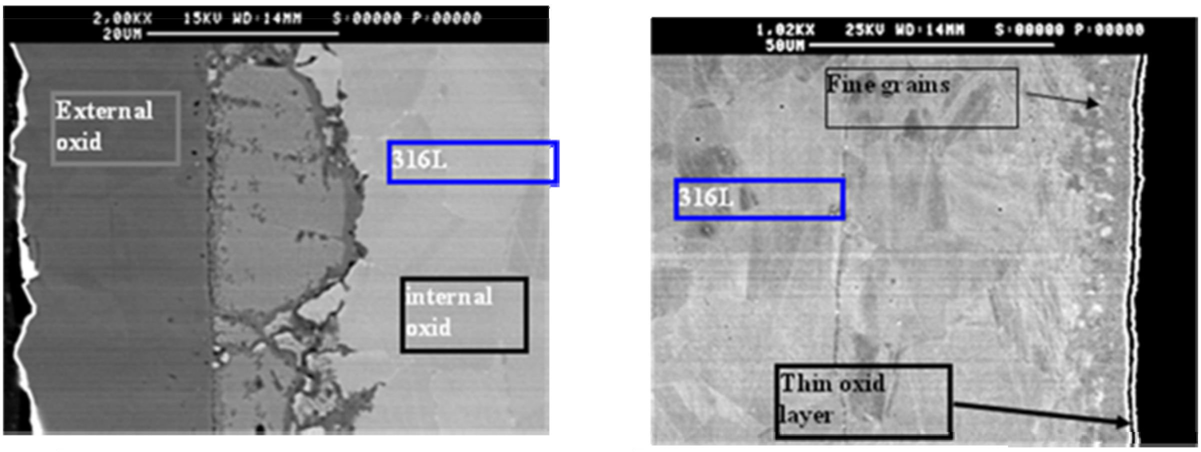
During 2009 corrosion and/or stress corrosion cracking (SCC) tests were conducted by all participants and new data were added to the database that is being compiled by Canada. This database currently includes information on the general corrosion of about 90 alloys and is being expanded to include data on SCC. Tests were conducted in a variety of facilities including static autoclaves, flow loops and pressurized capsules; a number of new test facilities were commissioned in 2009.

Data on general corrosion for a range of candidate materials (including ferritic/martensitic steels, austenitic stainless steels, ODS steels and titanium) continue to be reported. Data were reported for Hastelloy C276, Alloy 625, 304NG, 316L, commercial and modified SUS310S, SS 1.4970 (15Cr15NiTi), AL-6XN, XN26TW, P91, P92, MA 956 (an ODS steel), 15H2MFA, EUROFER97, titanium gr.2 alloy, and other alloys. Weight change measurements and a variety of metallographic and surface analysis techniques have been used to characterize these materials following exposure to SCW. The results of long-term tests were reported for XN26TW (1 year, 620°C) and 316L (3 500 h, 600°C). A significant finding was the large effect of surface microstructure on corrosion as shown in Figure 3-11, comparing the corrosion attack on a polished stainless steel coupon with that on a machined coupon. If the relevant mechanisms can be well-understood, this may provide a means of imparting a higher corrosion resistance to materials currently not considered ideal candidates for use in an SCWR.

Results from a detailed study on the corrosion of irradiated commercial SUS310S austenitic stainless steel and three other experimental materials proposed by Hitachi (H2) and Toshiba (T3F and T6) were reported. Several promising candidates for fuel cladding were discussed, including modified 310 SS (25Cr20Ni), PM3000 (20Cr F/M ODS) and SS 1.4970 (15Cr15NiTi). While there is no overall consensus on the best material for fuel cladding, there was general agreement that the Hitachi H2 modified 310SS containing Zr appears to be the best candidate reference material for the fuel qualification testing. However, test data are needed at temperatures up to 700°C; the required tests could probably be performed at the VTT facility (which can reach up to 695°C).

Agreement was reached on the materials to be included in round-robin tests with various participants agreeing to supply the chosen materials. The joint purchase of the ODS alloys for the round-robin testing was discussed; the capability of producing ODS materials is also being developed. A draft procedure for coupon cutting, polishing and weighing was developed, as well as a draft set of test conditions and analysis methods to be used in the round-robin tests.

Figure 3-11: Comparison of polished and machined 316L coupons after corrosion test (600°C for 3 500 h)




Polished

Machined (not polished)

Results from combined creep and oxidation tests were reported for 316NG, 347H and SS 1.4970 (15Cr15NiTi). The data suggest that the SCW environment increases the creep rate compared to that found in a He environment.

Work continued in the area of corrosion mechanisms and oxide growth modeling, in particular on the effects of diffusion of vacancies/interstitials in the oxide and the effect of dopants. To overcome the low efficiency of conventional molecular dynamics simulations at the relatively low temperatures of interest (650°C), temperature-accelerated molecular dynamics was used for this work. Preliminary simulations with $Cr_2O_3$ grain boundaries were performed. Work on computational modeling of radiation damage in Fe-based alloys, thermo-physical solute and defect property studies of Fe-based alloys, and microstructure-based irradiation creep predictions was also carried out.

A major effort is now underway to develop surface treatments, including the application of coatings, to modify the corrosion resistance of materials that exhibit good mechanical properties but have poor corrosion characteristics in SCW. This approach may be a viable option if no alloy can be identified that possesses all of the properties required for use in a SCWR. However, a significant effort will be required to ensure that coatings remain intact throughout the intended lifetime of the component. In 2009 reviews of the stability of candidate ceramics in SCW were performed and a number of ceramic materials were prepared for preliminary evaluation under SCW conditions. Work was carried out to characterize the thermal expansion coefficients of candidate substrate materials and coatings and to optimize the bond alloys. Fe-25%Cr alloy was cast and atomized for the purpose of coating by cold-spray, and preliminary trial of cold-spray coating on P91 was performed. Preliminary experiments coating Al onto P91 using cold-spray and then conversion treatment to alumina by plasma oxidation were also carried out.

The goal of the R&D tasks on water chemistry is to define the reference water chemistry for longer-term testing and chemistry control strategies for the SCWR. Some suggestions for SCWR water chemistries were proposed, including the idea of a "dual water chemistry" in which the feedwater chemistry differs from the water chemistry downstream of the core. Preliminary results from experimental and modeling studies on corrosion product transport were reported in 2009. Data from fossil-fired SCW power plants and thermodynamic analyses suggest a risk of corrosion product deposition in-core near the critical point. Corrosion results using a range of water chemistries used in fossil-fired SCW power plants were reported; ammonia led to significant releases of Fe and Ni likely due to formation of metal-ammonia complexes. It was suggested that LiOH, used in PWRs and PHWRs but not in fossil-fired SCW plants, might be a viable water treatment for the SCWR. The metal release properties of 304 stainless steel at elevated temperatures from 300 to 550°C under different oxidizing water chemistries (dissolved $O_2$ concentrations from <10 ppb to 200 ppb) using radiotracers were reported. The main obstacle to progress in the area of corrosion product transport is the lack of reliable data on the solubility of relevant metal oxides. Work is underway to address this shortcoming using a combination of modeling and experiment.

Control of water chemistry and corrosion product transport may require one or more purification steps. In 2009, various high temperature-pressure (HTP) separation technologies for purifying SCW in SCWR were reviewed to identify possible purification strategies.

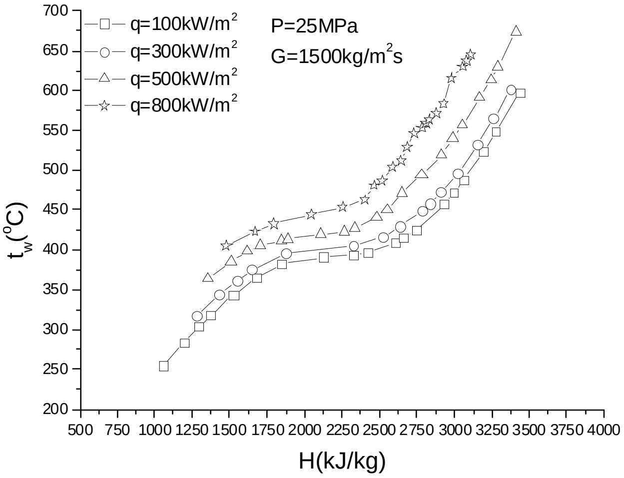
Fundamental work, both experimental and computer modeling, continued on the effects of radiation on supercritical water. The aim is to develop sufficient understanding of water radiolysis under the expected operating conditions of SCWR in order to: a) predict the concentrations of oxidizing species (e.g. $O_2$, $H_2O_2$) expected in-core and downstream of the core, and b) to allow the development of mitigating strategies. Experimental techniques include pulse radiolysis methods while the simulations involved the use of molecular dynamics and Monte Carlo to simulate radiolytic reactions in SCW.

*Thermal-hydraulics and Safety (Canada, EU, Japan, observers: R. Korea, China)*

In Canada, a heat-transfer experiment is being performed with an annulus in supercritical water flow. The inner heater element has an outer diameter of 8 mm. It has been installed vertically inside an unheated flow tube of 16 mm in inside diameter. Several issues were encountered during the commissioning phase and most of them have been resolved. One remaining issue is the burnout of the lead wires on the sliding thermocouples due to the high temperature inside the heated tube. Surface temperature measurements obtained with fixed thermocouples (Figure 3-12) are being analyzed. The sliding-thermocouple mechanism design is being improved.

A number of test facilities are being designed and constructed in Canada. Most of these facilities are established for heat-transfer tests with tubes, annuli and bundle sub-assemblies in water, carbon dioxide or refrigerant flows. A separate test facility for critical flow experiments at supercritical water conditions is being constructed. In addition, the construction of the test facility for supercritical water flow stability has been completed. Commissioning will commence once the data-acquisition system is connected and becomes fully functional.

Figure 3-12: Wall-temperature measurements obtained from the Supercritical Water Heat-Transfer Test with an Annulus



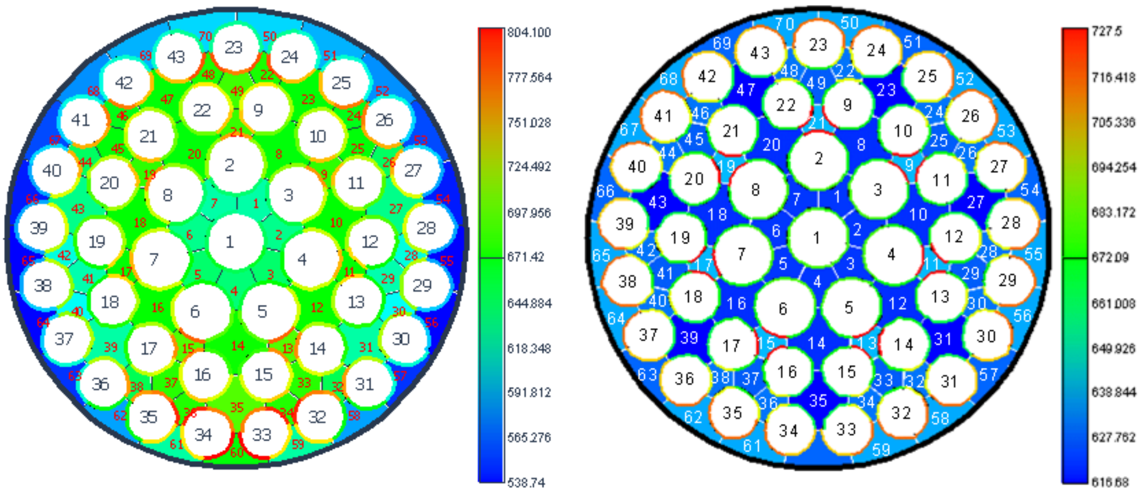

Thermal-hydraulic characteristics at supercritical water flow conditions are required in support of the fuel-bundle design and qualification and safety analysis for SCWR. An experimental database on supercritical heat transfer for water flow (both upward and downward) in tubes has been compiled from published literature. The database covers pressures up to 40.5 MPa (local to critical-pressure ratios up to 1.8), mass fluxes up to 2.4 $Mg{\cdot}m^{-2}{\cdot}s^{-1}$ and fluid temperatures up to 767°C (local to critical-temperature ratios up to 1.6).

One key criterion used in the nuclear safety analyses of existing operating plants is the requirement to prevent dryout. In existing power plants, prevention of dryout assures that fuel will remain wet and well cooled during accident scenarios. Because of the lack of phase change in the core, SCWRs cannot use design criteria based on the critical heat flux concept (unlike existing CANDU or Light Water Reactors). Criteria must be based on fuel and cladding temperature limits; therefore, heat transfer calculations and uncertainties are critical for safety system analyses. This includes any parallels to regulatory requirements such as those employed by the Canadian Nuclear Safety Commission, the US Nuclear Regulatory Commission and the International Atomic Energy Agency. A review of regulatory requirements was initiated focusing on the applicability to SCWR. The review of Canadian licensing criteria has been completed and a review of relevant BWR and PWR requirements is nearing completion.

In recent years, the definition of design-based accidents has shifted to a probabilistic approach where each accident scenario is categorized based on its probability of occurrence and its possible outcome. Furthermore, key safety decisions and maintenance requirements are also shifting to Risk Informed or Cost-Benefit approaches where statistical information, level of knowledge and end consequences are used to assess the appropriateness of each option. A hierarchical method will be used to define design-based accidents based on both bottom-up and top-down approaches. Since the conceptual model of the Generation IV design is evolving with time, the goal of this work is to determine a "first cut" for design-basis accidents so that tool and methodology developments can focus on these high importance failures. Work has been initiated in 2009 in terms of preliminary risk model development, government regulations and technical standards review.

The CANDU SCWR fuel is being optimized using a simplified sub-channel code. Figure 3-13 illustrates distributions of surface and fluid temperatures in the CANDU 37-element bundle together with optimized CANFLEX bundles at supercritical conditions. The maximum surface and fluid temperatures in the optimized CANFLEX bundle are systematically lower than those in the 37-element bundle. This demonstrates the possibility of improved operating and safety margins through fuel-design optimization.

Figure 3-13: Optimization of CANDU Fuel Design for SCWR




**Horizontal**

**Max. Tw: 804°C**

**Max. Tb: 693°C**

| Parameters | Values |
|---|---|
| Coolant inlet pressure | 25 MPa |
| Coolant inlet temperature | 350°C |
| Coolant exit temperature | 625°C |
| Average coolant mass flux | 890.3 kg·m$^{-2}$·s$^{-1}$ |
| Average heat flux | 0.978 MW·m$^{-2}$ |
| Limit of cladding surface temperature | 850°C |

**Vertical**

**Max. Tw: 727°C**

**Max. Tb: 637°C**

Canada participated in the IAEA Coordinated Research Program (CRP) on heat transfer and safety analysis codes at supercritical conditions (Republic of Korea also participates in the CRP but both EU and Japan are not formal participants). The second meeting of this CRP took place in 2009 and participants have deposited a large amount of experimental data in the NEA-IAEA databank. A couple of benchmarking exercises have been initiated for comparison of model and computer code applicability to supercritical flow. Preparation of the draft technical document has been initiated.

In Europe, the HPLWR safety system has been proposed consisting of a low-pressure coolant injection system, ADS valves, building condenser, poisoning system and a core catcher. Additionally, a high-pressure coolant injection system has also been considered. The performance of both coolant injection systems has been simulated, showing that both can cool the reactor efficiently. Furthermore, parametric studies of depressurization events were carried out to determine, for example, the appropriate size of valves and actuation pressures of the ADS valves. Safety analyses were carried out to give a feedback to the design. Loss-of-feedwater transients were simulated to investigate temperature evolution within the core. The calculated temperatures are within the acceptance criteria. A series of ten reactivity-induced accidents were simulated. The acceptance criteria are fulfilled, in spite of the strong feedback, where in some cases the hot channel temperatures are not far from the acceptance criterion limits.

In the HPLWR design, a helical wire-wrap spacer has been adopted in the rod bundle to improve mixing between subchannels. In order to investigate heat transfer and derive a correlation to be applied to rod bundle flows, different geometries have been simulated using computational fluid dynamics (CFD) software tools. The idea is to model heated rods and heated annuli, with and without wires, and a 4-rod bundle to investigate the influence of the presence of the wire and the geometry on heat transfer. CFD results have been processed resulting in correction factors to upgrade available heat transfer correlations from literature to account for the effect of the wire-wrap spacer and the effect of geometry.

As an example, CFD results for a 4-rod bundle with and without the presence of wire-wrap spacers are shown in Figure 3-14, and Figure 3-15. Relatively high heat flux and low mass flux have been assumed (hot channel conditions). The result demonstrates the strong non-uniformity of surface temperature for the case of the bare rod-bundle. This non-uniform temperature distribution is mitigated by the presence of the helical wire in the rod bundle. The surface temperature is globally lowered by the presence of wire-wrap spacers, but some hot spots still exist at the surface of the fuel rods.

Figure 3-14: Surface-temperature distribution (in Kelvin) for 4-rod bundle without wire-wrap spacers

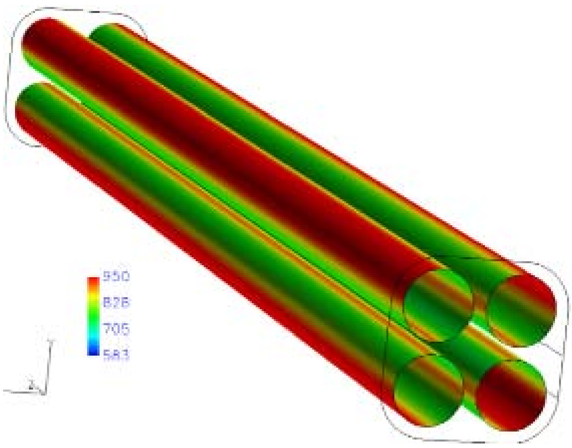
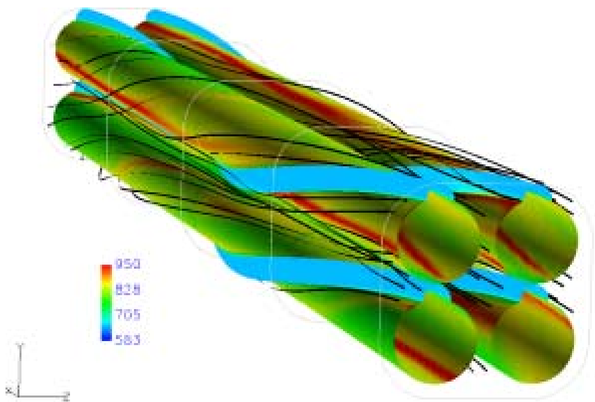


Figure 3-15: Surface-temperature distribution (in Kelvin) for 4-rod bundle with wire-wrap spacers



Thermal-hydraulics tests at supercritical pressure conditions with water and Freon flows have been done in Japan to obtain heat transfer data using a tube and a bundle.

Experiments were performed with a supercritical pressure HCFC22 forced circulation loop, newly set up at Kyushu University, Japan. HCFC22 has been used as a substitute for water because its critical pressure and temperature values of 4.99 MPa and 96.2°C respectively, are far lower than those of water (22 MPa and 374°C), providing experimental flexibility. Steady-state tests were carried out with a single tube of 4.4 mm ID and with a bundle subassembly (Bundle-I) composed of seven heater rods simulating the actual fuel bundle geometry.

Figure 3-16 shows the typical result of wall temperatures and heat transfer coefficients in the tube and the 7-rod bundle subassembly. In the bundle subassembly, occurrence of heat transfer deterioration is generally suppressed even for upward flow, and the heat transfer characteristic is similar to that in the normal heat transfer region of the tube flow.

The Republic of Korea is not a formal member of the PMB, but has participated as an observer in PMB activities and contributed experimental heat-transfer data obtained at KAERI's dedicated facility SPHINX (a unique test facility with carbon dioxide flow). Supercritical heat transfer experiments have been performed with downward flow of carbon dioxide through six different test sections: tubes of 4.57, 6.32, and 9.0 mm inside diameters, a concentric annular channel of 8 x 10 mm (1 mm gap), and an eccentric annular channel of 9.5 x 12.5 mm (wider gap 2 mm and narrow gap 1 mm). Equivalent experiments using similar test sections (the small tube diameter was 4.4 mm instead of 4.57 mm) with upward flow were completed in 2008.

Figure 3-16: Comparisons of experimental wall temperatures and heat-transfer coefficients between tube and bundle

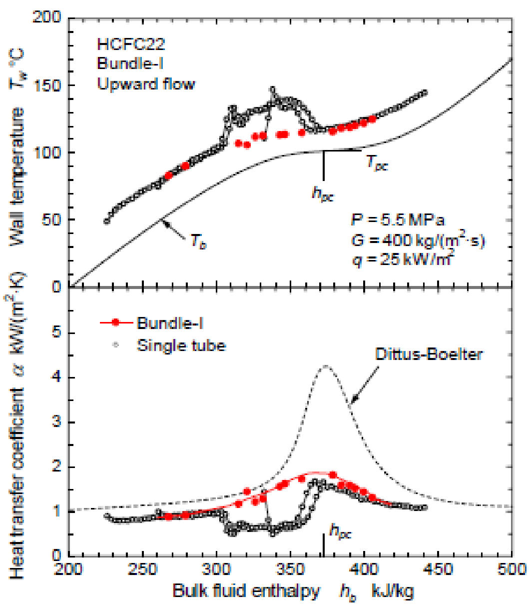
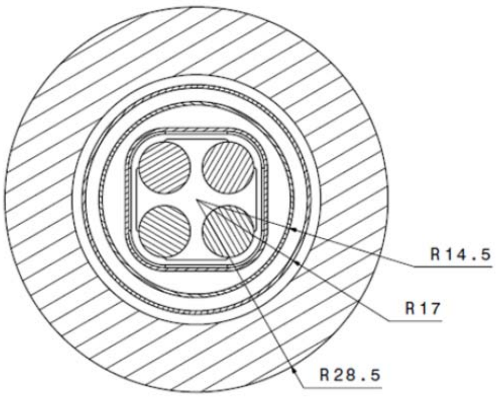



Figure 3-17: Assembly and pressure tube for SCWR fuel qualification test




*Fuel Qualification Test (Canada, EU, Japan)*

A generic fuel qualification test, requiring the licensing of a nuclear facility operated with supercritical water for the first time, is considered as a milestone before a prototype reactor can be built. The pool type reactor LVR-15, situated in Řež in the Czech Republic, was found to provide a suitable environment for a pressure tube with 57 mm outer diameter, filled with 4 fuel rods of 60-cm length in its lower part. It shall simulate peak power conditions of the High Performance Light Water Reactor (HPLWR) with heat and mass flux close to hot channel conditions. Figure 3-17 shows a cross section of the test section with 4 fuel rods having diameter and pitch equivalent to those foreseen for the HPLWR concept. The tube is connected to a supercritical water loop at 25-MPa pressure and 300°C temperature, with a recuperator boosting the coolant temperature at the inlet of the test section to typical reactor conditions. The loop shall be ready for operation in 2015. Experience with the supercritical water loop for radiolysis and water chemistry tests, already built for this reactor, shall be used as a basis.

### 3.1.4 *Gas-cooled Fast Reactor (GFR)*

#### Main characteristics of the system

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

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

#### Status of cooperation

The System Arrangement was signed at the end of 2006 by Euratom, France, Japan and Switzerland. Two projects were then discussed, 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 2010.

#### 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. Therefore, the GFR R&D focuses on aspects specific to this system, i.e. core and in-core design and components, and safety.

An experimental demonstration and technology reactor, named ALLEGRO (formerly ETDR), is proposed to be built in the coming decades. With a thermal power around 80 MWth, ALLEGRO is foreseen to demonstrate the viability of the GFR system, an essential step since no reactor of this type has been built before. ALLEGRO incorporates, at a reduced scale, all the architecture and the main materials and components foreseen for the GFR, except for the power conversion system. Its safety principles are those proposed for GFR. It will also contribute to the development and qualification of an innovative refractory fuel element that will be able to withstand operation at high power density and high temperature.

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

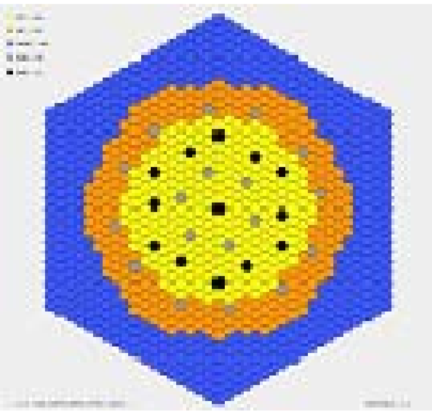
### *Core design*

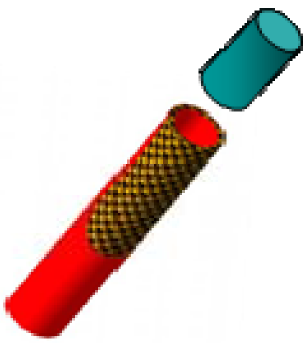
Comparisons were made of the performances of three different core designs, all of which were based on a mixed carbide fuel material, but with different clad materials and fuel element designs:

- A plate fuel element clad with a Composite Matrix Ceramic of SiC (already studied in 2008 and 2009);
- A pin fuel element clad with a Composite Matrix Ceramic of SiC;
- A plate fuel element clad with a Vanadium alloy.

The three cores perform well, but the third needs to operate at a lower temperature and there are still some large uncertainties on the Vanadium alloy properties. Finally, for ceramic clad elements the greater ability to fabricate a pin geometry element compared to a honeycomb plate element led to the ceramic pin being chosen as the new reference fuel element design for GFR cores (see Figure 3-18 below).

Figure 3-18: Fuel loading map for a ceramic core made of pin fuel element clad with SiC ceramic





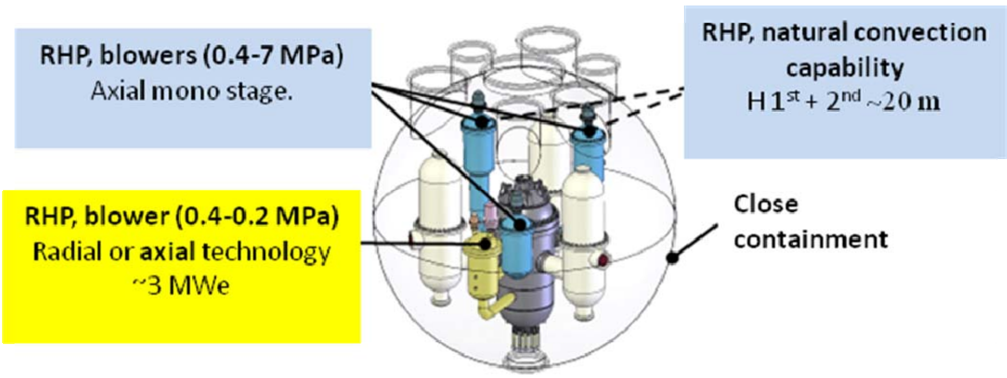
*Decay Heat Removal systems*

A strategy to actuate the various systems for Decay Heat Removal (DHR) was set up in order to minimize the risk and a Probabilistic Safety Analysis was established:

- for the most frequent situation (with integrity of the primary circuit) and Anticipated Transient Without Scram (ATWS), the use of normal loops (main blowers operated with pony motors supplied by diesel), using for the heat sink either the steam generator (with by-pass of the turbine) or dedicated air coolers (back-up system, operated in case of loss of the electrical grid);
- for other situations, the use of dedicated DHR systems designed to operate at nominal primary pressure as well as to the large range of possible backup pressures after depressurization. This system is composed of (Figure 3-19):
  - **R**eactor **H**igh **P**ressure cooling system: 3 x 100% with blowers as normal systems (0.4 - 7 MPa) & 2 x 100% with natural convection as backup system ;
  - **R**eactor **L**ow **P**ressure cooling system: 1 x 100% with blower designed for very low pressure (0.4 - 0.2 MPa).

As a possible option, helium or heavy gas injection from dedicated reservoirs is studied to improve the DHR systems under LOCA conditions in case of blower failure (intermediate back-up pressure situation). The use of a Brayton machine in order to add passivity to the DHR system in case of close containment failure (low back-up pressure situation) is also investigated.

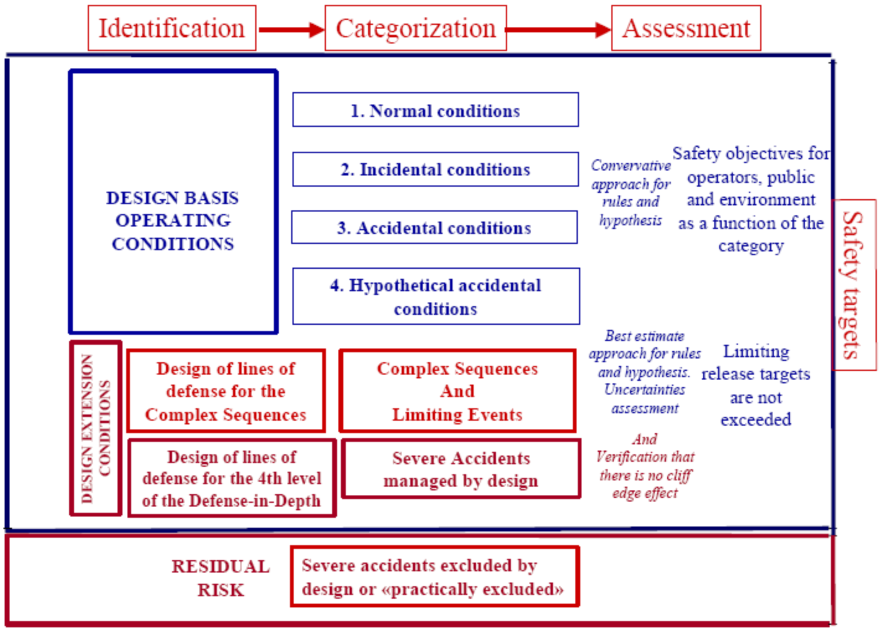
Figure 3-19: View of primary system, dedicated DHR loops inside the close containment




*Safety Approach*

A first probabilistic assessment was performed to verify that there are no vulnerable areas in the design with the potential for high-level risk sequences. In this way, probabilistic safety assessment can identify any requirement for additional design features for preventing or mitigating accidents (see Figure 3-20 below).

Figure 3-20: Sketch of the methodology for the safety analysis




### *Study of the Reference Design Basis Accidents*

Preliminary acceptance criteria were retained for the assessment of the Design Basis Accidents (DBA):

- Category 3 situations with clad temperature < 1 450°C and upper plenum temperature < 1 250°C;
- Category 4 situations, the more limiting criterion being considered among fuel temperature < 2 000°C, clad temperature < 1 600°C, upper plenum temperature < 1 250°C, and no degradation of the flow paths such that core cooling can be maintained;
- Categories 3 and 4: a controlled state must be reached at the end of the sequence.

The reference situations analyzed result from the combination of the initial state of the reactor (full power), of an initiating event (IE) and of the single aggravating failure inducing the most adverse effect on the consequence of the transient.

The single aggravating failures considered were:

- Failure of a diesel train or of a blower when actuated;
- Failure to open a DHR loop, or failure to close one main loop (required to operate the DHR loops).

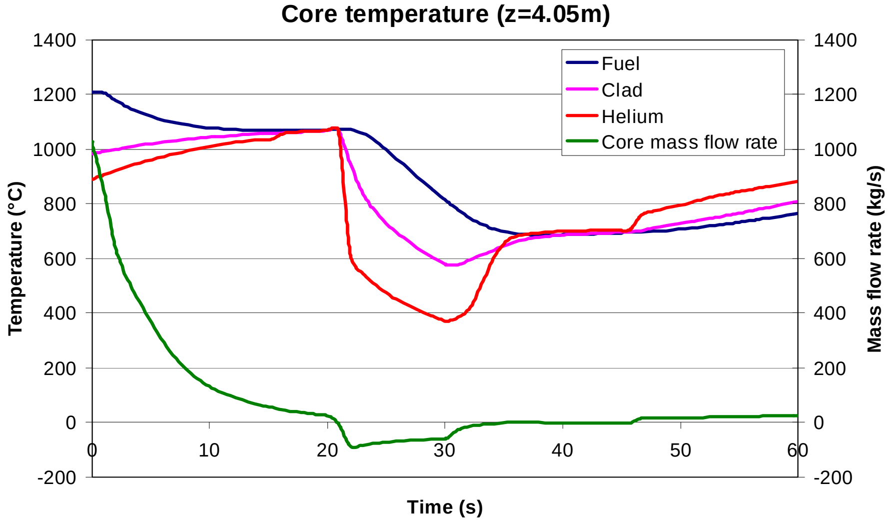
The following Category 3 reference situations have been studied:

- Loss Of Off-site Power (LOOP) longer than 2 hours;
- Small break in the primary circuit, or in the IHX, or in the secondary circuit.

The following Category 4 reference situations have been studied:

- Large break in the IHX or in the primary circuit (Figure 3-21);
- All the plant transients (about 30 cases) have been simulated using the CATHARE-2 computer code.

Figure 3-21: LB-LOCA, calculated core mass flow rate and temperatures

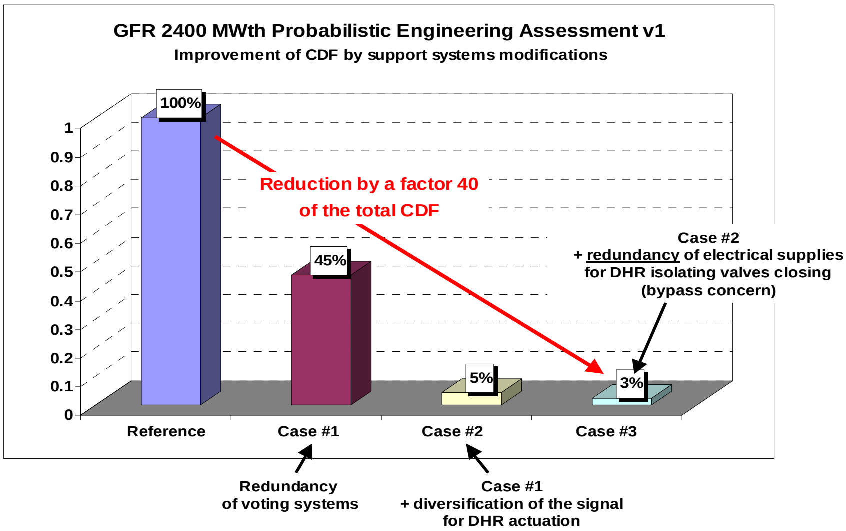



Additional Design Basis Situations were also added:

- LOOP combined with multiple failures;
- Small primary break combined with multiple failures;
- Control of the small primary breaks initiating events with the DHR system operating with a natural circulation flow.

Finally, for most likely events, significant margin is available between the overheating calculated in the transient and the acceptance criteria. Moreover, after an iterating process between probabilistic analysis and design, the dimensioning, the redundancy and the diversification of the DHR system enable a postulated partial core bypass due to an erroneous configuration of coolant pathway to be tolerated, with a large temperature margin to the acceptance criteria for most situations. In the case of a very fast depressurization, the acceptance criteria are also met, but with a reduced margin. Globally, the Core Damage Frequency can be reduced by a factor of around 40 by optimizing the design and the control system, as shown in Figure 3-22 below.

Figure 3-22: Core Damage Frequency estimated by Probabilistic Safety Analysis and optimization of its value by design



*Severe accidents*

Several families of severe accident scenarios leading to severe plant conditions have been preliminarily identified. An approach was proposed to distinguish those families depending on the integrity of the safety barriers, the magnitude and the dynamics of the phenomena induced by the accidents and the possible threshold effects in the scenario. A preliminary set of situations was identified by means of this approach; they are classified depending on several criteria like dynamics of the phenomena, integrity of the barriers, core geometry, reactivity control, knowledge of phenomena, and overall ability to control and mitigate the accident or whether it is possible to demonstrate practical elimination.

Moreover, the ability of the GFR to withstand severe plant conditions relies mainly on the behavior of the core materials at a high temperature, including a chemically aggressive atmosphere due to nitrogen ingress, possible water ingress and more improbably due to air ingress. As a result, experimental tests are under way in order to assess the capability of the highly refractory GFR core materials to withstand the accidents associated with severe plant conditions.

Experimental tests have been initiated to measure the behavior of UPuC and SiC-SiCf at 2 000°C subject to different atmospheres. Oxidation is expected on composites since dissociation may occur for mixed carbide. As far as the air ingress situation is concerned, experimental studies, carried out between 1 000°C and 1 700°C, showed two oxidation features: a passive oxidation with the formation of a protective $SiO_2$ layer at low temperature and high oxygen partial pressure, and an active oxidation with the formation of an unstable SiO layer at high temperature and low oxygen partial pressure. The passive oxidation regime would not lead to a loss of clad mechanical properties. On the contrary, the active regime must be demonstrated to be reached only for a limited duration and within a limited region of the core.

### *3.1.5 Lead-cooled Fast Reactor (LFR)*

#### Main characteristics of the system

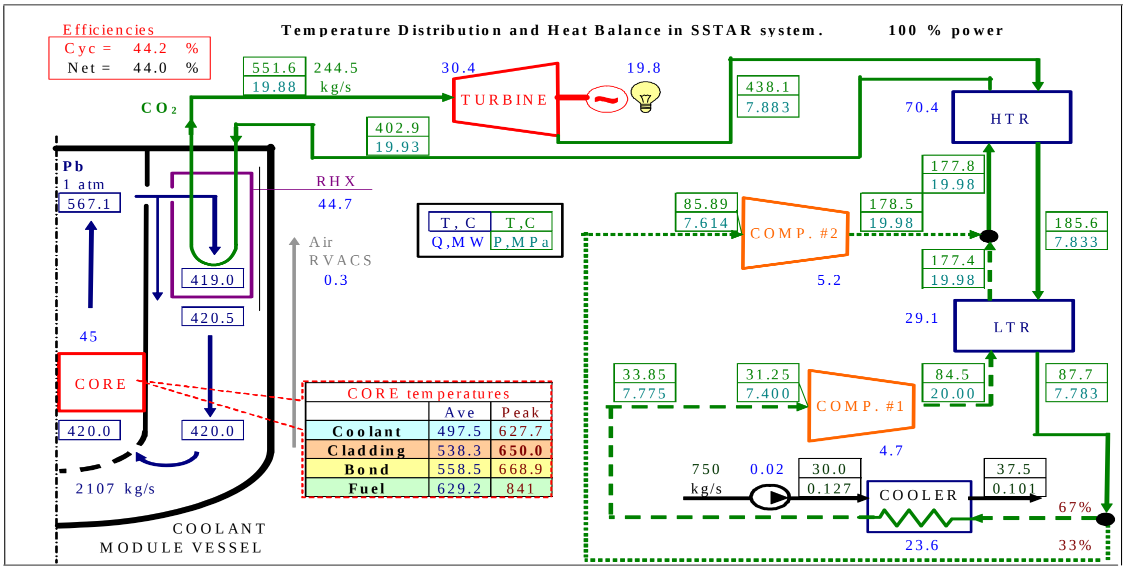
The LFR features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of minor actinides, both self-generated and from reprocessing of spent fuel from LWR's, and as a burner/breeder with thorium matrices. An important feature of 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 grid-connected power stations.

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

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

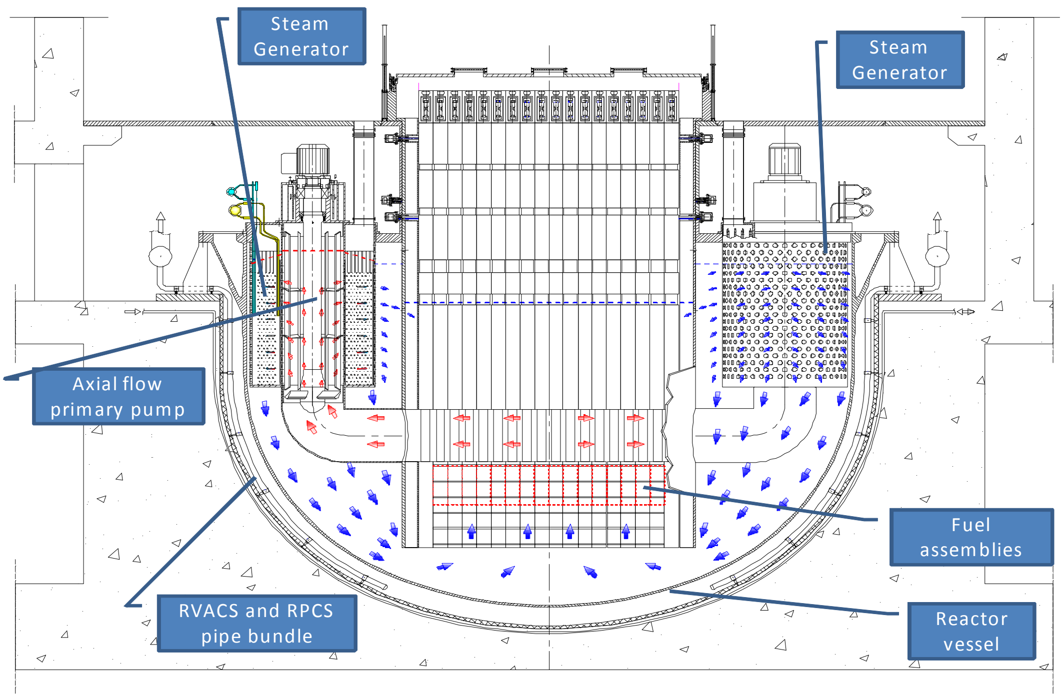
The reference design for the SSTAR in the United States is a 20 MWe natural circulation reactor concept with a small transportable reactor vessel (Figure 3-23). Specific features of the lead coolant, the nitride fuel containing transuranic elements, the fast spectrum core, and the small size combine 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.

Figure 3-23: SSTAR pre-conceptual design and operating parameters




The ELSY reference design (Figure 3-24) is a 600 MWe reactor cooled by pure lead (L. Cinotti, *et al.*, 2008). This concept has been under development since September 2006 within the 6th Euratom Framework Programme. The ELSY project is being performed by a consortium consisting of seventeen organizations from Europe. ELSY aims to demonstrate the possibility of designing a competitive and safe fast reactor using simple engineered technical features while fully complying with the mission identified in the GIF Roadmap of minor actinides burning capability.

Figure 3-24: ELSY configuration



### 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) which was reviewed by the Experts Group (EG) in mid-2007 and mid-2008. In addition, informal meetings were held with representatives of the nuclear industry, research organizations and universities involved in LFR development. A revision of the SRP is in preparation for the next review by the EG.

In 2009 discussions were held on the mode of cooperation on LFR R&D in GIF. The Policy Group took the decision to set up a Memorandum Of Understanding (MOU) for both the MSR and LFR systems. This MOU would provide a more flexible structure for R&D cooperation on those systems in the GIF framework for the mid-term.

Typical design parameters of the SSTAR and ELSY concepts are summarized in Table 3-4.

Table 3-4: Key design parameters of GIF LFR concepts

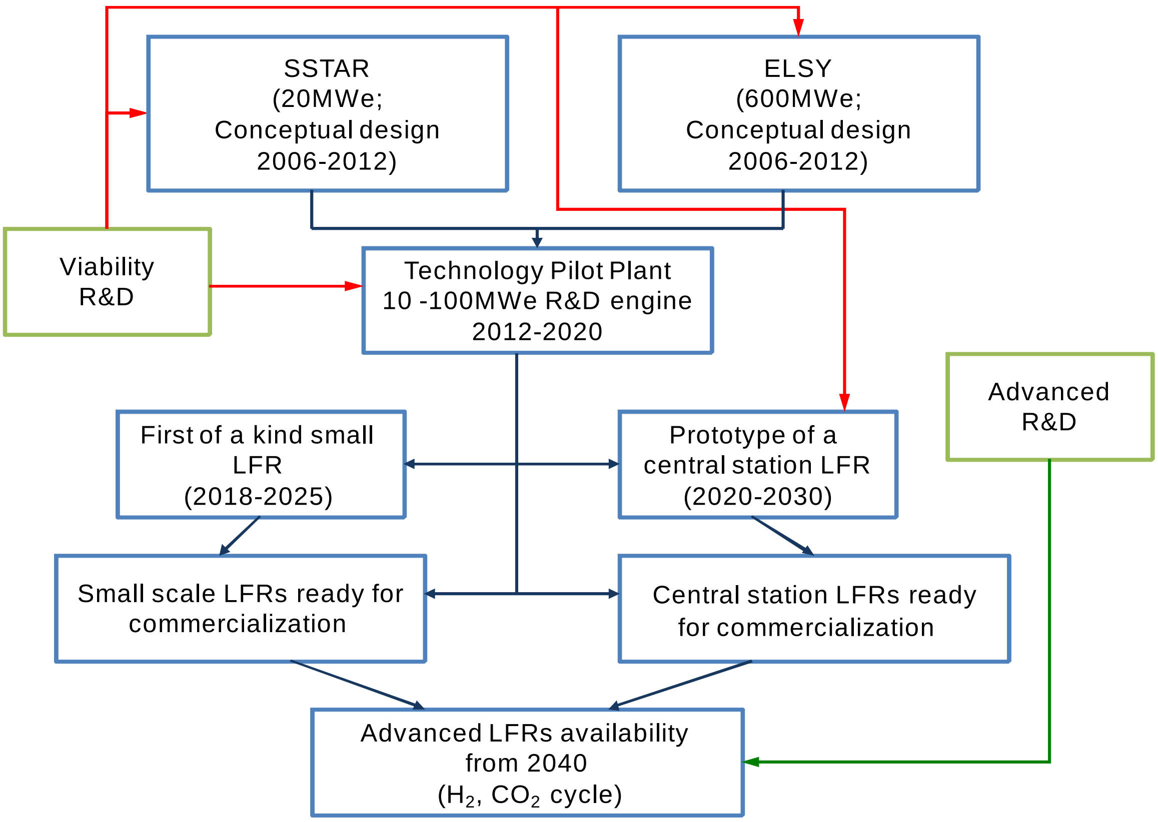
| Parameters | SSTAR | ELSY |
|---|---|---|
| Power (MWe) | 19.8 | 600 |
| Conversion Ratio | ~1 | ~1 |
| Thermal efficiency (%) | 44 | 42 |
| Primary coolant | Lead | Lead |
| Primary coolant circulation (at power) | Natural | Forced |
| Primary coolant circulation for direct heat removal | Natural | 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 |

### 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 3-25 illustrates the basic approach recommended in the draft SRP. It portrays the dual track viability research program with convergence to a single, combined technology pilot plant leading to the eventual deployment of both types of systems.

Figure 3-25: 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 of 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). The four areas are: system integration and assessment; lead technology and materials; system and component design; and fuel development. General issues of concern for LFR development include corrosion of structural materials, lead technology, in-service inspection, instrumentation, assessment of the steam generator tube rupture (SGTR) accident, fuel development, control rods operating in lead and refueling in lead.

### Main activities and outcomes

In 2009, the activities have been largely devoted to the LFR development in Europe focusing on the completion of the conceptual reference configuration of ELSY including core, primary system, steam generator units, primary pumps, decay-heat removal systems, refueling system, containment system and overall plant layout.

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

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

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

The core consists of an array of open square fuel assemblies surrounded by reflector assemblies, a configuration that presents reduced risk of coolant flow blockage (A. Alemberti, *et al.*, 2009). An alternative solution based on closed hexagonal fuel assemblies is retained as a back-up option. The upper part of the fuel assembly is peculiar to the novel ELSY design, because it extends well above the fixed reactor roof, and the fuel elements, whose weight is supported by lead, are fixed at their upper end in the cold gas space, well above the lead surface. This avoids the classical problem of a core support grid immersed in the coolant, which would complicate in-service inspection in lead.

Considering the high temperature and lead environment, any approach requiring the use of in-vessel refueling equipment would represent a tremendous R&D effort and substantial associated technical risk, especially because of the need to develop reliable bearings operating in lead, an unknown technology at present. For these reasons, the adopted design approach represents a real breakthrough. Installation of steam generators inside the vessel is one of the main challenges of a LFR design (L. Cinotti *et al.*, 2009). In operation there is need for a sensitive and reliable leak detection system and a highly reliable depressurization and isolation system.

Extensive analyses have been initiated to address the most critical safety and mechanical issues. Both protected and unprotected transients are under evaluation.

The provisions adopted to prevent core damage ensure that no initiating events will progress to the point of damaging the core so that the core damage frequency targets established by the Risk and Safety Working Group (RSWG) can be met. The safety-related functions relative to reactor operation are mainly based on the action of passive mechanisms, with no (or limited) external support means required to be available. In particular, it was investigated whether the core could suffer massive clad failures as a consequence of an unprotected loss of flow accident.

Mechanical analyses performed in 2009 include consideration of the seismic loads associated with lead sloshing effect as well as other loads associated to the SGTR accident.

Other beneficial effects of the specific ELSY design provisions have been confirmed, namely the support of the reactor building by 2D seismic isolators, the short-height vessel, and the innovative provisions conceived to make the primary system more tolerant to the SGTR accident.

Following the conclusion of the ELSY project in early 2010, the LFR design activity will continue forward under the next 7$^{th}$ Framework Programme of Euratom with the Lead-cooled European Advanced DEmonstration Reactor (LEADER) project. LEADER 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.

In the US it is recognized that, if SSTAR were to be developed for near-term deployment, the operating system temperatures would likely have to be reduced (e.g. to a maximum coolant temperature of 480°C, as in the ELSY design) to enable the use of proven materials and qualification of an existing fuel type such as the metallic fuel. Scoping calculations have been carried out in order to assess the near-term feasibility of a pilot plant/demonstration test reactor (demo) operating at low temperatures enabling the use of T91 ferritic/martensitic steel and Type 316 stainless steel, both steels already proven by test during the past decade to be corrosion-resistant to lead alloys up to ~ 550°C with active dissolved oxygen control. Neutronic and thermal hydraulic analyses indicate that a 100 MWth lead-cooled metallic-fueled demo with forced flow and 480°C core outlet temperature may be a viable concept supporting the development of both the ELSY and SSTAR LFRs.

Important complementary activities are in progress in US and Europe with the aim of providing a suitable solution for the fuel cladding material, which is the most critical technological issue related to the use of lead as coolant.

At Massachusetts Institute of Technology, in the US, Functionally Graded Composite (FGC) cladding materials are under development. At KIT in Germany, work is continuing toward the optimization of the GESA material coating process to control Al content in the coating in order to assure long-term corrosion protection to cladding. T91 specimens representative of fuel cladding, Al coated with sputtering technology at the University of Trento, Italy, are being tested in flowing lead in the CHEOPE loop at ENEA; results, even if preliminary, are very encouraging.

A thorough understanding of the thermal hydraulic behavior of complex components in a pool-type reactor will be gained by three different experiments, which have the aim to characterize, respectively, a single fuel rod (at KTH, Sweden), a fuel bundle mock-up, (at KIT, Germany) and a cooling loop of a core (at ENEA, Italy).

In Japan, the Tokyo Institute of Technology is active on several LFR systems research activities and technological development for use of heavy liquid metal and associated structural materials. Included among these is the work on the CANDLE travelling-wave reactor.

### *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 (L. Mathieu *et al.*, 2009). 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.).

Apart from MSR systems, studies are performed on employing liquid salt in other advanced reactor concepts, for example as primary coolant in Fluoride-cooled High-temperature Reactors (FHR), or as an alternative to sodium in the secondary (intermediate) heat transfer loop in Sodium-cooled Fast Reactors (SFR) and as an alternative to intermediate helium in Very-High-Temperature Reactors (VHTR).

More generally speaking, the development of higher temperature salts as coolants could bring new nuclear and non-nuclear applications. These salts could facilitate heat transfer for nuclear hydrogen production concepts, concentrated solar electricity generation, oil refineries and shale oil processing facilities, among other applications (C.W. Forsberg *et al*., 2007).

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 reactor designs, and coated particle fuels similar to high temperature gas-cooled reactors (C.W. Forsberg *et al*., 2008). 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). The better fluoride salt heat transport characteristics, as compared to helium, enable power densities 4 to 8 times greater as well as power levels over 4 000 MWth with passive safety systems. Fuel cycle characteristics are essentially identical to those of the VHTR, while intermediate heat transport, power conversion and balance of plant are essentially identical to those of the "reference" MSR.

#### Status of cooperation

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

In 2009 discussions were held on the mode of cooperation on MSR R&D in GIF. The Policy Group took the decision to set up a Memorandum Of Understanding (MOU) for both the MSR and LFR systems. This MOU would provide a more flexible structure for R&D cooperation on those systems in the GIF framework for the mid-term.

The status of cooperation, the progress of R&D and the perspectives for the development of MSR and FHR were presented at the GIF Symposium in September 2009 (C. Renault *et al.*, 2009). Another presentation highlighted material issues and the effect of chemistry control for the MSFR (S. Delpech *et al.*, 2009b).

In 2009 two meetings of the MSR PSSC were held. An important achievement was a convergence on the implementation of six Projects (see section on R&D issues). In addition, a White Paper was presented to the Proliferation Resistance & Physical Protection Working Group (PRPPWG) and submitted in a revised form in November.

Beyond the GIF framework, the MSR PSSC has significantly contributed to enhance and harmonize international collaborations.

Partners of the MSR PSSC are involved in the Euratom-funded ISTC-3749 project, started in February 2009 with official support from France (CEA, CNRS, EDF), Germany (FZK), the Czech Republic (NRI), the United States (ORNL), EC (JRC-ITU) and IAEA. This project, following ISTC-1606, takes advantage of the large expertise and the facilities existing in Russia.

Finally, a European network on MSR R&D has been active from 2001 (MOST, 5th FWP) until 2008 (ALISIA, 6th FWP). The ALISIA (Assessment of LIquid Salts for Innovative Applications) Specific Support Action (SSA) was the major part of the Euratom contribution to MSR in Generation IV. The project was completed in 2008.

In 2009 a new MSR 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 is, following a positive expert evaluation, under contract negotiations in the frame of the Euratom collaboration with Rosatom – which is negotiating the complementary MARS (Minor Actinides Recycling in Molten Salt) project between Russian research organizations. 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, in order 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 on the 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 for intermediate heat transport in other systems (SFR, LFR, VHTR) are investigated. Liquid salts offer two potential advantages: smaller equipment size, because of the higher volumetric heat capacity of the salts; and the 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 EG 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

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 2009, including:

1. Development of MSFR pre-conceptual designs and performance analysis of MSFR potential for starting with plutonium and minor actinides from PWR wastes (France, E. Merle-Lucotte *et al.*, 2009a and 2009b).
2. Laboratory scale processing of Ni-W-Cr alloys was recently demonstrated. The alloys were found to have acceptable workability and very good high temperature hardness (France, Auger *et al.*, 2009). The full potentials of these kinds of materials as well as Hastelloy N have yet to be tested and characterized over the full range of temperatures and in the presence of fluoride salts.
3. Corrosion tests of Ni-based alloys, (France, S. Fabre *et al.*, 2009).
4. Better understanding of $PuF_3$ solubility in various carrier salts by means of thermochemical modeling (Euratom, O. Beneš *et al.*, 2009).

5. The material property database for molten and liquid salts was extended through experiments and theoretical calculations at Euratom (O. Beneš 2009) and in France (M. Salanne *et al.*, 2009). New experimental facilities were and continue to be developed (Euratom JRC-ITU).

6. Significant improvements were brought to the fuel salt clean-up scheme (S. Delpech, 2009a).

7. A code package for a fast MSR was developed (M.W. Hoogmoed, 2009) by coupling the 3-D time-dependent diffusion code DALTON with the thermo-hydraulics code HEAT (The Netherlands, TU-Delft).

8. The optimal core configuration and salt composition of a moderated MSR that maximize the power density while keeping the self-breeding capabilities were determined (The Netherlands, TU-Delft). New breeding gain definitions were developed (K. Nagy *et al.*, 2010) that account for the unique behavior of the reactor.

9. Construction of a fluoride salt test loop was initiated in the USA.

10. An FHR component test plan was completed in the USA (D.E. Holcomb *et al*, 2009). The test plan provides a roadmap to the major technical demonstrations required to enable a test scale FHR to be built.

11. Construction of a surrogate material compact integral effect test apparatus in support of a test scale FHR was initiated (USA). The new apparatus is intended to demonstrate the coupled thermal hydraulics response of FHRs to transients including loss of heat sink and loss of forced circulation.

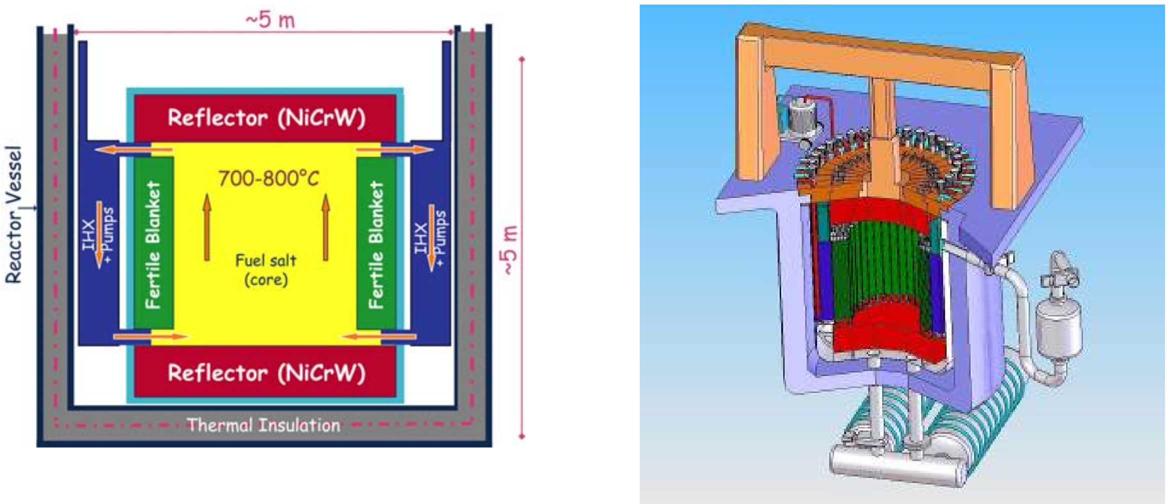
12. Criticality tests for the assessment of FHR fuel and core behavior have been negotiated (USA, Czech Republic).

Some of these topics are further discussed in the following section.

### *MSFR design and performance analysis*

The potential of MSFRs was highlighted (L. Mathieu *et al.*, 2009). Design studies led to realistic drawings of the MSFR system, showing the main components of the reactor and their arrangement in the vessel (Figure 3-26). For the 3 000 MWth (~1 350 MWe) 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).

Figure 3-26: MSFR pre-conceptual design



In terms of fuel cycle, two basic options were investigated, $^{233}$U-started MSFR and TRU-started MSFR, both operated on the thorium fuel cycle. The latter option aims at circumventing the unavailability of $^{233}$U necessary to start the first generation of MSFR (E. Merle-Lucotte *et al.*, 2009a and 2009b). Calculations demonstrated the performance of this option. Studies show that an inventory of $^{233}$U lower than 4 metric tons per GWe can be easily reached for the $^{233}$U-started MSFR version, while the TRU-started MSFR version allows burning rates ranging from 87% to 93% after 50 years of operation. These studies also brought to light the limitations of the concept due to the irradiation damage to the structural materials (S. Delpech *et al.*, 2009b).

*Measurement of the properties of reference salt systems*

Work performed within the ISTC-3749 project in the Russian Federation will allow to determine missing or uncertain data for molten salt mixtures containing transuranic (TRU) elements which were identified in previous years, such as melting points, TRU solubility, thermal conductivity and expansivity.

New experimental facility ies (Figure 3-27 and Figure 3-28) were and continue to be developed at the Institute for Transuranium Elements (JRC-ITU) to measure thermal conductivity, heat capacity, viscosity and density. The FFFER (Forced Fluoride Flow for Experimental Research) loop is under construction (CNRS, France). FFFER is dedicated to bubbling studies and will be operated with LiF-NaF-KF salt (Figure 3-29).

Figure 3-27: A high temperature calorimeter installed at JRC-ITU

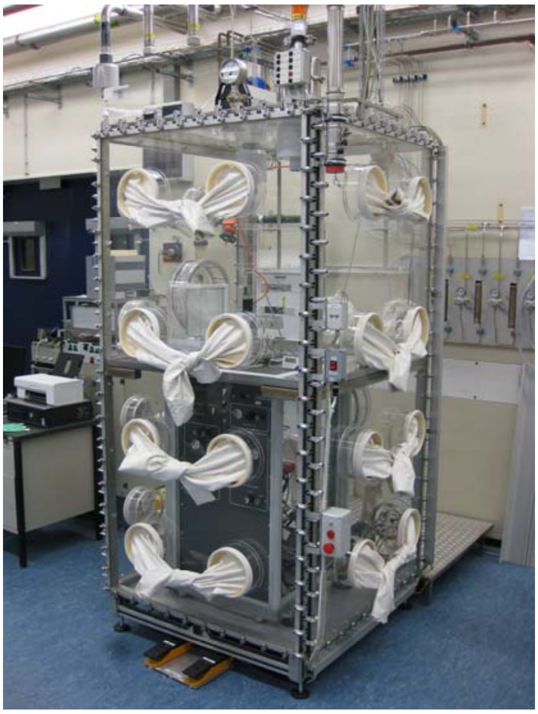
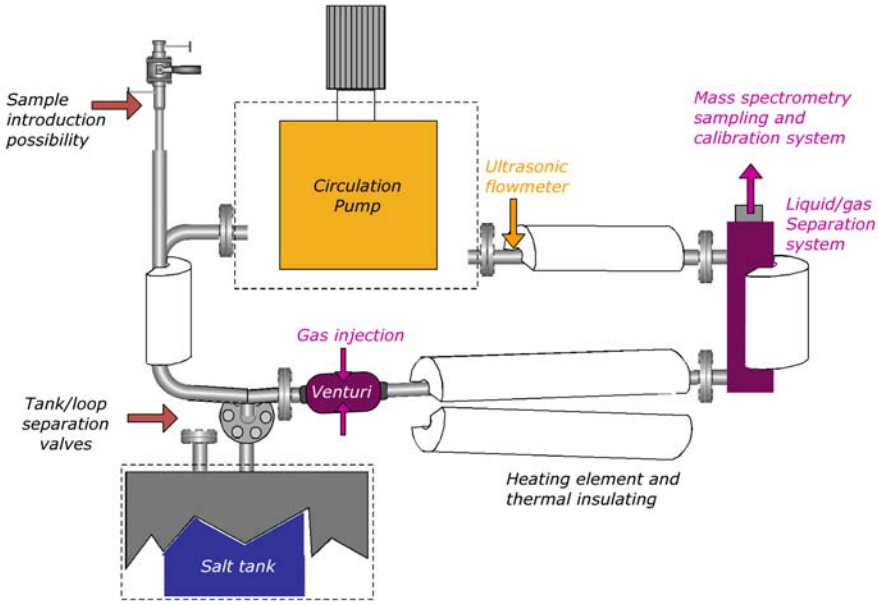


Figure 3-28: A viscometer and densitometer installed at JRC-ITU

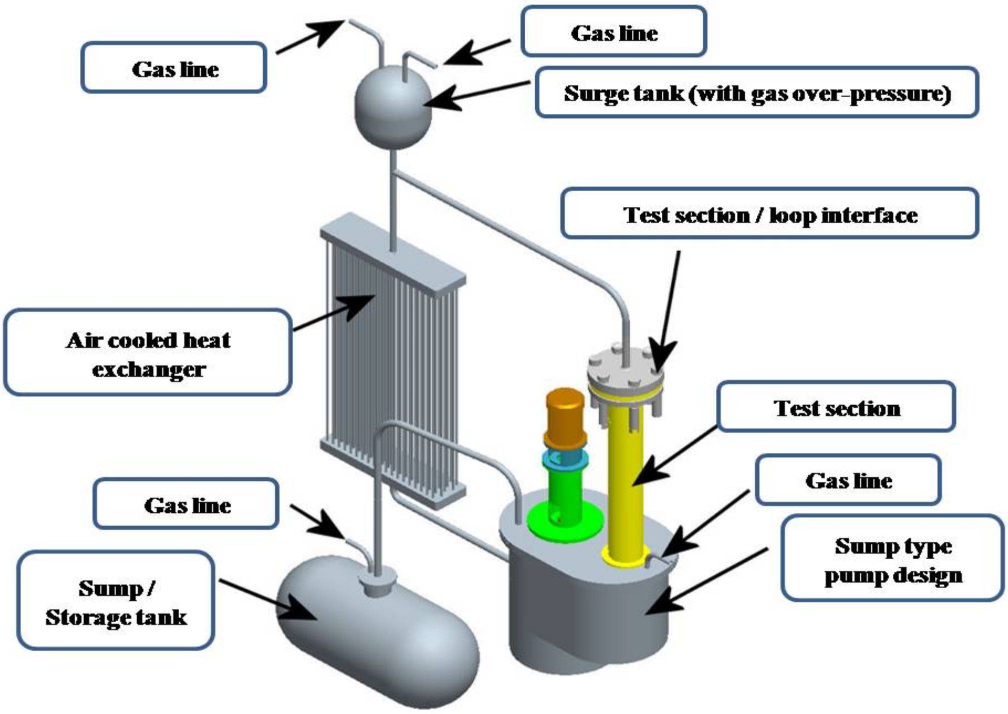
Figure 3-29: Simplified view of FFFER facility (CNRS, France).




*FHR design and assessment*

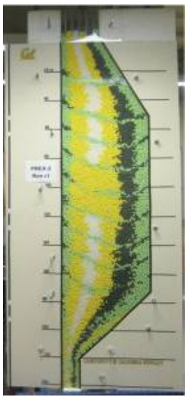
An induction heated fluoride salt test loop is currently under construction at Oak Ridge National Laboratory (ORNL) in the United States (Figure 3-30). The loop is intended to demonstrate heat transfer properties of fuel-simulant graphite pebbles with energy deposition properties similar to those anticipated in pebble bed FHRs (PB-AHTR). The loop also features a silicon carbide test segment along with a mechanical joint between the ceramic and metallic loop sections.

Figure 3-30: Conceptual view of ORNL induction heated fluoride salt test loop



Pebble recirculation experiments at UC Berkeley (Figure 3-31) verified the capability to generate radially zoned, annular pebble core configurations, allowing the implementation of radial thorium blankets to increase FHR conversion ratios (R. Hong *et al*., 2009). Thermal hydraulic modeling studied core coolant flow distribution and transient response to loss of forced circulation events. Neutronic studies identified optimal depletion levels for thorium blanket pebbles and for LEU and LWR-TRU seed pebbles.

Figure 3-31: UC Berkeley Pebble Recirculation Experiment (PREX-2)
for a 15° sector of a 900-MWth FHR core, verifying viability of radially zoned annular FHR pebble cores with thorium blankets.



The SPHINX (SPent Hot fuel Incinerator by Neutron fluX) project was originally defined as a suitable experimental basis at representative scale for the demonstration of MSR-burner feasibility (M. Hron *et al.*, 2008). It relies on the utilization of the zero power experimental reactor LR-0 being operated in the Nuclear Research Institute Řež (NRI), Czech Republic. This full-scale physical model of the PWR cores was modified in order to allow the measurement of all the neutronic characteristics of the MSR burner and/or breeder blanket, initially at room temperature and in future stage at close to operational conditions (Figure 3-32).

Figure 3-32: LR-0 zero power critical test facility (SPHINX project)

Because two baseline concepts (MSFR, FHR) are now considered in Generation IV, a corresponding broadening of the SPHINX project was discussed and formally adopted at the end of 2008. The LR-0 will thus be used for the validation of FHR neutronics models (reactivity coefficient variation with temperature) in the frame of collaboration between the Czech Republic (NRI) and United States. 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.

## 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 recommendation from the Economics Crosscut Group of the Generation IV Roadmap Project that a standardized cost estimating protocol be developed to provide decision makers with a credible basis to assess, compare, and eventually select future nuclear energy systems taking into account a robust evaluation of their economic viability.

The methodology developed by the EMWG is based upon the economic goals of Generation IV nuclear energy systems as adopted by GIF:

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

The methodology produced by the EMWG consists of:

- *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems, Rev. 4* (GIF/EMWG/2007/004);
- G4ECONS Software Package;
- *Users Manual for G4ECONS Version 2.0* (GIF/EMWG/2007/005).

The validity of the Cost Estimating guidelines and the G4ECONS software, for both Generation III and Generation IV systems was demonstrated through sample calculations. A CD containing the complete methodology is available from the NEA.

In 2009 the EMWG further developed a standard Training Presentation. The Training Presentation is modularized in order to be useful from the management level to the detailed end-user level. EMWG members are ready to provide the presentation to GIF groups upon request.

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 giving an overview of the Methodology and its applications.

Furthermore, the EWMG began in 2009 the 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 actinide management applications.

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 PRPPWG was created to develop, implement and foster the use of an evaluation methodology to assess Generation IV nuclear energy systems with respect to GIF proliferation resistance and physical protection goals (see www.gen-4.org/Technology/roadmap.htm). Information on the activities of the Group since its creation and on the outcomes from its work up to 2008 may be found in GIF Annual Reports 2007 and 2008. The Proliferation Resistance and Physical Protection (PR&PP) methodology developed by the Group is described in a document entitled *Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems Revision 5*, released for general distribution in 2006 (GIF/PRPPWG/2006/005, www.gen-4.org/Technology/horizontal/PRPPEM.pdf) together with a volume of addenda.

In 2009, the PRPPWG, following its revised Terms of Reference and the guidance of the Policy Group (PG), focused its activities on:

- collaborating with the GIF System Steering Committees (SSCs);
- publicizing the methodology and examples of its application within and outside GIF; and
- improving the methodology, taking advantage of feedback from users' experience.

The interaction with SSCs aims at raising the awareness within GIF system research teams of proliferation resistance and physical protection aspects of their respective systems. This should facilitate integrating these considerations at an early stage of system design, in an approach similar to that adopted for nuclear safety. The overarching goal is to understand PR&PP aspects of each system concept in order to enhance the resistance to proliferation and the robustness of physical protection of the nuclear systems being considered within GIF.

Sharing of information and expertise between GIF system research teams and the Group was pursued in 2009 by telephone conferences and during meetings, and the collaboration with SSCs was strengthened through undertaking jointly the drafting of System White Papers (SWPs). The overall objectives of SWPs are to: describe each concept with emphasis on its PR&PP relevant aspects, taking into account potential threats (e.g. diversion of weapons-usable materials); highlight key features of each system contributing to its proliferation resistance and physical protection; and identify R&D needs for designing intrinsic measures aiming at increasing proliferation resistance and physical protection of the GIF systems.

The Group supported the preparation of 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.

Provisional versions of the SWPs were discussed at a workshop for SSC representatives held at Brookhaven National Laboratory (BNL), NY, USA, on 7-8 July 2009. Each of the six GIF systems were represented by one or more members of the SSC and/or experts who had been involved in the drafting of the SWPs. Members of the PRPPWG guided and moderated the discussions. The workshop led to the establishment of a work plan for finalizing the SWPs and pursuing collaborative activities. Updated versions of the SWPs, taking into account the recommendations issued during the workshop, were prepared by the end of 2009 and it is planned to reconvene a workshop in 2010 for finalizing the SWPs and discussing the feasibility of having the SSCs undertake a more detailed assessment for one or more of the GIF systems.

The progress report on preparation of the SWPs, which was presented to the GIF Experts Group (EG) at its May 2009 meeting, triggered high interest from the EG members who recommended that the papers be compiled within a comprehensive document, also covering (insofar as is feasible) cross-cutting issues – such as back-end of the fuel cycle – which are not treated by SSCs. Accordingly, the PRPPWG has included in its work plan for 2010 the preparation and release of a document on PR&PP aspects of GIF systems; the document is expected to be available in the second semester of 2010.

Publicizing the methodology and its application was pursued through participation of members of the PRPPWG in international conferences and seminars. Whenever possible and relevant, members of the Group organized specific sessions during those international events, dedicated to the outcomes and results of the Group activities. In particular, several sessions that focused on the PR&PP methodology and examples of its application were held during the Global 2009 Conference (www.sfen.fr/index.php/plain_site/global_2009/) held in Paris, France, in September. The sessions were well-attended and discussions following the presentations provided insights on potential user needs and areas or topics deserving further attention from the PRPPWG.

The report on the Example Sodium Fast Reactor (ESFR) case study, issued in draft at the end of 2008, was reviewed and finalized in 2009 taking into consideration comments, remarks and suggestions from the EG. The final draft of the document, titled *ESFR Case Study Report* (GIF/PRPPWG/2009/002), was issued in October 2009 and is being reviewed by the GIF community prior to its eventual release for general distribution expected by mid-2010.

The main objective of the case study was to exercise the PR&PP methodology on a specific Generation IV system and to illustrate how its results may assist decision makers in comparing design options from PR&PP viewpoints. The analysis covers four threat strategies: concealed diversion of material; concealed misuse of facilities; breakout followed by overt diversion or misuse; and, theft of weapons-usable material or sabotage of facility system elements.

The case study illustrates a practical approach for applying the PR&PP methodology in a traceable way leading to accountable and dependable results for evaluating PR and PP pathways. Lessons learned from the study include the following:

- Each PR&PP evaluation should start with a qualitative analysis allowing scoping of the assumed threats and identification of targets, system elements, etc.
- There is a need to include detailed guidance for qualitative analyses in the methodology.
- Access to proper technical expertise on the system design as well as on safeguards and physical protection measures is essential for a PR&PP evaluation.
- The use of expert elicitation techniques can ensure accountability and traceability of the results and consistency in the analysis.
- Qualitative analysis offers valuable results, even at the preliminary design level.
- For a broader use of the methodology, greater standardization would be needed.

The case study generated a number of additional insights which will be reflected in future work for enhancing the methodology. It demonstrated that the PR&PP methodology has the potential to be a powerful tool that can be applied at the conceptual design stage of advanced nuclear energy systems, and generate guidance for detailed system design. Future work will include efforts to further exercise this approach and demonstrate its usefulness in guiding the design of Generation IV nuclear energy systems.

Successive studies carried out using the PR&PP methodology have shown that the approach provides structured qualitative analysis with traceable, accountable and dependable results, as well as useful information to system designers even when detailed design information is largely missing. It can support the identification of differences in PR and PP aspects of design variations. The comparative assessment of alternative design assumptions can generate insights on functional requirements applicable at the stage of detailed design work.

Drawing from the experience gained through the ESFR case study, collaboration with SSC representatives and feedback from other users, the Group undertook in 2009 a review of the measures and metrics adopted in the PR&PP methodology. The methodology defines a set of high-level parameters, called measures, which are representative of the PR&PP robustness of a system against potential threats. Quantitative metrics associated to each measure are proposed to help quantify the analyses (see GIF/PRPPWG/2006/005).

The review of measures and metrics undertaken in 2009 confirmed that the approach adopted in the PR&PP methodology is adequate and that results from the analyses provide relevant indicators to support decision making. However, it highlighted that metrics may need to be adapted to each specific case study taking into account both the context and objectives of the analyses as well as the goals and priorities of the decision makers for whom the study is carried out. The work will be continued in 2010.

Preliminary outcomes from the review of measures and metrics identified the desirability of making the outcomes from PR&PP analyses more understandable to policy makers. In particular, there was a suggestion that reducing the number of indicators could help policy makers in using the results from PR&PP assessments. However, the consensus of the Group remains strongly in favor of providing the decision makers with sufficient information to reflect the various aspects of intrinsic PR&PP.

Work continued in 2009 towards developing a harmonized understanding of the PR&PP and INPRO PR (International Project on Innovative Nuclear Reactors and Fuel Cycles) assessment approaches, through an ad hoc subcommittee comprised of members from both development teams. The goal of this work is to understand the relative strengths and possible synergies of the two approaches and, if possible, simplify and align the overlapping evaluation processes in each approach. A fundamental finding of this subcommittee to date is that the two approaches are complementary: the INPRO PR approach is a high-level assessment or check-list which can incorporate the PR&PP methodology where a detailed and systematic evaluation is required. This work will be concluded in 2010.

### 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.

The year 2009 provided the opportunity to clarify the pertinence of the approach proposed by the RSWG with, first, the definition of Objectives and principles and second, with the development of a methodology allowing the assessment of the systems versus these Objectives and principles.

During 2009 the work of the RSWG focused on the development of a methodology, the Integrated Safety Assessment Methodology (ISAM), for use throughout the Generation IV technology development cycle. 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 as well as 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 neither be used 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 to perform meaningful evaluations of Generation IV concepts for the attainment of the Generation IV safety objectives.

The integrated methodology consists of five distinct analytical tools and stages which are structured around the last one, the Probabilistic Safety Assessment. The tools/stages are the following:

- Qualitative Safety Requirements/Characteristic Review (QSR);
- Phenomena Identification and Ranking Table (PIRT);
- Objective Provision Tree (OPT);
- Deterministic and Phenomenological Analyses (DPA);
- Probabilistic Safety Analysis (PSA).

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.

Figure 3-33 shows the overall task flow of ISAM and indicates which tools are intended for use in each phase of Generation IV system technology development.

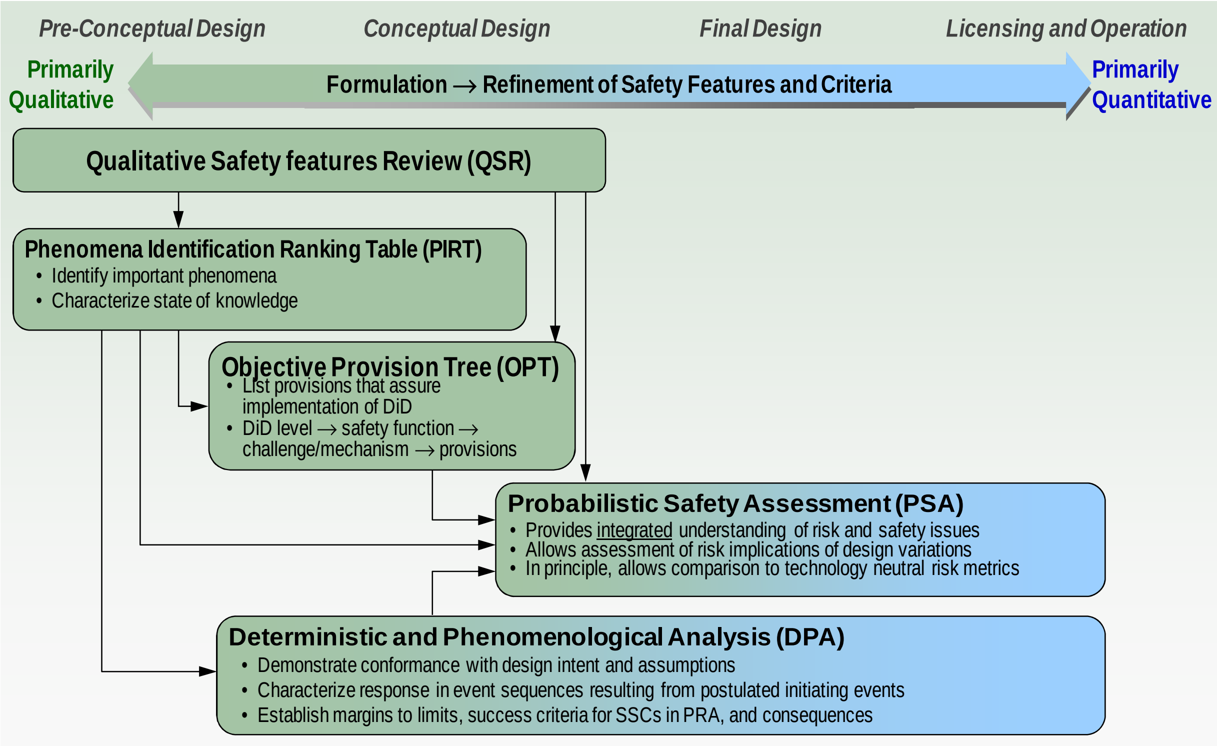
During the next several years, the work of the RSWG will focus on formulating and documenting the assessment methodology in detail, working through a host of technical issues associated with the methodology, developing and demonstrating sample applications to selected hypothetical and practical problems 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.

Concerning the interactions with other GIF internal or external entities, it is interesting to evoke contacts with the Senior Industry Advisory Panel (SIAP); the PR&PP working group and the Systems Steering Committees (SSC), the Multinational Design Evaluation Programme (MDEP) and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO)

SIAP

Following SIAP, GIF and System Steering Committees should develop an "overarching plan" that fosters the future commercialization of each of the Generation IV systems, i.e. "*a complete picture from a sufficiently high level; it is about the scope and not necessarily only for future*". In this context the role of the methodology groups is important to help in ensuring the coherency of the process. Concerning more specifically the RSWG, it is worth noting that SIAP also asks for a "*clear safety case logic*" which has to be "*convincing to Nuclear Regulators and the Public*" and in which all claims have to be researched and backed-up with sound evidence, which may "*need helping regulatory staff move from existing practices to those appropriate for new circumstances*". This is consistent with the RSWG approach which is based on the idea that, while keeping the foundations in terms of safety objectives and principles, it should be possible to provide adequate and innovative answers based on the characteristics of the Generation IV systems (e.g. inherent characteristics, passive systems, etc.). These converging views between the SIAP and the RSWG will be the subject of discussions to deepen the analysis of the positions of the two groups.

Figure 3-33: Proposed Gen-IV Nuclear Systems Integrated System Assessment Methodology (ISAM)



SSCs

Following a strategy analogous to the one adopted by the PR&PP, the RSWG distributed a draft *White Paper* to launch the interaction with SSCs. The key objective of this document is to provide a sort of "table of content" 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 Expert Group. Following this recommendation, the ISAM methodology was presented during the PR&PP Workshop in July 2009. 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

In 2009 a critical comparison was made between the INPRO requirements and the *Qualitative Safety Requirements/Characteristic Review (QSR)* which is one of the tools of the ISAM. The results showed that improvements were required both for the RSWG QSR and for the INPRO methodology. Since then, the set of RSWG/QSR recommendations has been corrected to integrate the inputs from the analysis. Specific action is engaged to interact with the IAEA/INPRO team in order to discuss the suggestions for improvements of the INPRO methodology.

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# CHAPTER 4 COLLABORATIONS WITH OTHER INTERNATIONAL PROGRAMS

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

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

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

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

The INPRO (www.iaea.org/INPRO/) initiative started in 2000 under the auspices of the IAEA which ensures its management. Its main objective is to support the safe, sustainable, economic and proliferation-resistant use of nuclear technology to meet the global energy needs of the 21$^{st}$ century. As of December 2009, it has 31 members (including 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.

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. It is also 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 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 2009, 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. In this way, in addition to the collaboration between the PR&PPWG and INPRO (see Chapter 3.2.2), the RSWG made a comparison between INPRO requirements and the Qualitative Safety Requirements from INSAM, which lead to an improvement implemented in RSWG/QSR, and suggestions for improvement of INPRO methodology to be discussed with them.

## 4.2 Global Nuclear Energy Partnership (GNEP)

The Global Nuclear Energy Partnership (GNEP) is an initiative 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 partnership consists of 25 partner countries, 31 observer countries and three permanent observer intergovernmental organizations – the International Atomic Energy Agency (IAEA), the Generation IV International Forum (GIF) and Euratom.

GNEP 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). Both working groups work closely with the IAEA and GIF, and as appropriate, with industry and other international organizations.

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 GNEP encompasses a broader policy vision than GIF, which focuses on technology progress through collaboration within specific R&D projects, both endeavors have similar goals for future nuclear systems, most notably improvement of waste management and enhancement of proliferation resistance. In its capacity as a GNEP permanent observer, GIF participates in GNEP meetings at all three levels. 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. The GIF also participated in the IDWG's May 2009 workshop on Small and Medium Size Reactors, and described its R&D activities in that field.

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

In order to achieve its long-term goals, MDEP will focus first on cooperation and convergence of regulatory practices that will eventually develop into convergence of regulatory requirements. 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, MDEP STC Chair attended one meeting of the RSWG in 2009, 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 goal, and the comparison of operating experience in reviews for new reactors. It was concluded that both groups will continue exchanging information. Also to be noted regarding this collaboration, the NEA, which serves as Technical Secretariat for MDEP as well as for GIF, facilitates exchange of information and realization of synergies between MDEP and GIF.

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. Therefore, a continued exchange of information will be established between the two projects, each of them benefiting from relevant progress and findings of the other.

### A.1.1 Technology Goals of GIF

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

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

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

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

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

### A.1.2 GIF Systems

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

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

Table A.1.1 Overview of Generation IV Systems

| **System** | Neutron spectrum | Coolant | Outlet Temperature °C | Fuel cycle | Size (MWe) |
|---|---|---|---|---|---|
| **VHTR**<br>(very-high-temperature reactor) | thermal | helium | 900-1 000 | open | 250-300 |
| **SFR**<br>(sodium-cooled fast reactor) | fast | sodium | 500-550 | closed | 50-150<br>300-1 500<br>600-1 500 |
| **SCWR**<br>(supercritical water-cooled reactor) | thermal/fast | water | 510-625 | open/ closed | 300-700<br>1 000-1 500 |
| **GFR**<br>(gas-cooled fast reactor) | fast | helium | 850 | closed | 1 200 |
| **LFR**<br>(lead-cooled fast reactor) | fast | lead | 480-570 | closed | 20-180<br>300-1 200<br>600-1 000 |
| **MSR**<br>(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.

# APPENDIX 2 GIF PRIORITY OBJECTIVES FOR THE NEXT FIVE YEARS

*(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 and uranium-oxicarbide 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).

## Super-Critical Water 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

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## 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 | Super-Critical Water 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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