ANNUAL REPORT
2013

GEN IV International Forum℠

Foreword from the GIF Chair

It is my pleasure to present the *2013 Generation IV International Forum Annual Report*, which provides an overview of the most up-to-date technical achievements in the development of Generation IV nuclear energy systems. This year was marked by the completion of many important GIF objectives, several of which are highlighted below.

The Generation IV International Forum (GIF) is now into its second decade. The Forum was formed by a multi-national agreement among countries who recognised that the future of nuclear energy depended on moving to the next generation of reactors and who were willing to collaborate on the research and development (R&D) to make this happen. Over time, the number of active members in GIF has evolved. Today's active GIF members are Canada, Euratom, France, Japan, the People's Republic of China, the Republic of Korea, the Russian Federation, South Africa, Switzerland and the United States. Good progress has been made on Generation IV systems and some of the revolutionary designs currently being developed could be demonstrated within the next decade. Commercial deployment could begin in the 2030s.

GIF maintains a long-standing collaborative relationship with the International Atomic Energy Agency (IAEA) and especially with the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). Co-operation on evaluation methodologies for economics, safety, physical protection and proliferation resistance has been ongoing for several years. In February 2013, GIF and INPRO held an interface meeting to discuss areas of mutual interest, at which time a workshop on sodium-cooled fast reactor (SFR) safety was also held.

In March 2013, the International Conference on Fast Reactors and Related Fuel Cycles (FR13) was held in Paris, France. This meeting was organised by the IAEA and hosted by the government of France. Over 650 attendees from 26 countries participated in the meeting with 375 papers presented. GIF members made major contributions to this very successful meeting.

Over the last few years, a GIF task force has been working to develop safety design criteria (SDC) for the sodium-cooled fast reactor, which is likely to be one of the first Generation IV systems to move through the viability and performance phases to the reactor demonstration phase. The task force completed its initial report, which was subsequently approved by the Policy Group at its meeting in Beijing in May 2013. This report represents an important step in developing international consensus on safety criteria for designers of Generation IV systems. As recommended by the Policy Group, the SFR SDC report was distributed for external review to national regulators and international organisations or programmes, such as the Multinational Design Evaluation Programme (MDEP), the OECD Nuclear Energy Agency (OECD/NEA), and the IAEA. This effort is important not only to harmonise safety criteria across GIF members, but also to demonstrate that Generation IV safety goals are being met. The SDC task force will thus continue its work in developing more detailed guidance on the SDC while factoring in comments from external reviewers.

During 2012 and 2013, GIF took the opportunity to reassess its mission and conducted strategic planning. A strategic plan was developed and approved by the Policy Group at its meeting in Beijing in May 2013. Key initiatives identified in the plan include:

- updating the technology roadmap;
- strengthening R&D collaboration within GIF;
- strengthening ties with other international organisations.

From GIF's beginning, collaborative research and development has been guided by the *Generation IV Nuclear Energy Systems Technology Roadmap* (2002). An update of the *Technology Roadmap* was completed and approved by the Policy Group at its meeting in Brussels in November 2013. The *Technology Roadmap Update for Generation IV Nuclear Energy Systems* (2014) reaffirmed that GIF R&D should continue to focus on the six Generation IV systems selected in the original roadmap. Those six systems are the sodium-cooled fast reactor (SFR), the gas-cooled fast reactor (GFR), the lead-cooled fast reactor (LFR), the supercritical-water-cooled reactor (SCWR), the very-high-temperature reactor (VHTR) and the molten salt reactor (MSR). The updated roadmap outlines progress to date and future R&D challenges for these six Generation IV concepts. It also discusses the work of the methodology working groups, which have focused on developing methods for assessing key attributes of Generation IV systems such as economics, safety, physical protection and proliferation resistance. The *Technology Roadmap Update* is available on the GIF website.

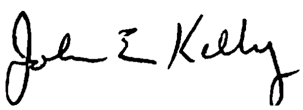
The strategic plan has also identified a number of areas where improvements in collaboration are possible and would lead to more efficient and effective co-operation. The experts group was tasked with developing an implementation plan to address this recommendation, with the improvements to be implemented over the next two years.

Finally, as the Generation IV Forum has matured, the need to strengthen ties with other international organisations has intensified. GIF will continue to reach out to the MDEP, the IAEA, and the NEA, and, in keeping with the strategic plan, GIF will continue to co-operate with the International Framework for Nuclear Energy Cooperation (INFEC) as an observer in the executive and steering committee meetings. Co-operation between GIF and organisations such as IFNEC and the IAEA is essential for future introduction and deployment of Generation IV nuclear energy systems, and the Forum needs to continue strengthening these associations.

GIF is also redoubling its efforts to communicate more effectively with its stakeholders in order to renew interest in Generation IV systems. A new website was launched in 2013, which will provide accurate and timely information to the public and particularly to educators. Generation IV concepts are innovative and exciting and the R&D is technically challenging. As such, it serves as an excellent opportunity to attract and train the next generation of nuclear professionals. GIF intends to continue with its outreach efforts to universities and professional societies so that the entire community can be engaged in Generation IV.

Finally, 2013 was a year of transition for the GIF chair with the chair position rotating back to the United States after three years with Japan. We would like to recognise the contribution of the previous chair, Mr. Yutaka Sagayama, and his staff for their outstanding leadership during a particularly important time in the history of nuclear power, as well as for ensuring a smooth and efficient rotation of the chair responsibilities.

With best wishes,



Dr. John E. Kelly
GIF Policy Group Chairman

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Chapter 1. GIF membership, organisation and R&D collaborations

1.1 GIF membership

The Generation IV International Forum (GIF) has 13 members, as shown in Table 1.1 which are signatories of its founding document, the GIF Charter. Argentina, Brazil, Canada, France, Japan, the Republic of Korea, South Africa, the United Kingdom and the United States signed the GIF Charter in July 2001. 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 1.1: Parties to GIF Framework Agreement, System Arrangements and Memoranda of Understanding as of 31 December 2013

Member	Implementing agents	Framework Agreement (FA)	System arrangements (SA)				Memoranda of Understanding (MOU)	
		Date of signature or receipt of the instrument of accession	GFR	SCWR	SFR	VHTR	LFR	MSR
Argentina (AR)								
Brazil (BR)								
Canada (CA)	Department of Natural Resources (NRCan)	02/2005		11/2006				
Euratom (EU)	European Commission's Joint Research Centre (JRC)	02/2006	11/2006	11/2006	11/2006	11/2006	11/2010	10/2010
France (FR)	Commissariat à l'énergie atomique et aux énergies alternatives (CEA)	02/2005	11/2006		02/2006	11/2006		10/2010
Japan (JP)	Agency for Natural Resources and Energy (ANRE) Japan Atomic Energy Agency (JAEA)	02/2005	11/2006	02/2007	02/2006	11/2006	11/2010	
People's Republic of China (CN)	China Atomic Energy Authority (CAEA) and Ministry of Science and Technology (MOST)	12/2007			03/2009	10/2008		
Republic of Korea (KR)	Ministry of Science, ICT and Future Planning (MSIP) and National Research Foundation (NRF)	08/2005			04/2006	11/2006		
South Africa (ZA)	Department of Energy (DoE)	04/2008						
Russian Federation (RU)	ROSATOM	12/2009		07/2011	07/2010		07/2011	11/2013
Switzerland (CH)	Paul Scherrer Institute (PSI)	05/2005	11/2006			11/2006		
United Kingdom (GB)								
United States (US)	Department of Energy (DOE)	02/2005			02/2006	11/2006		

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

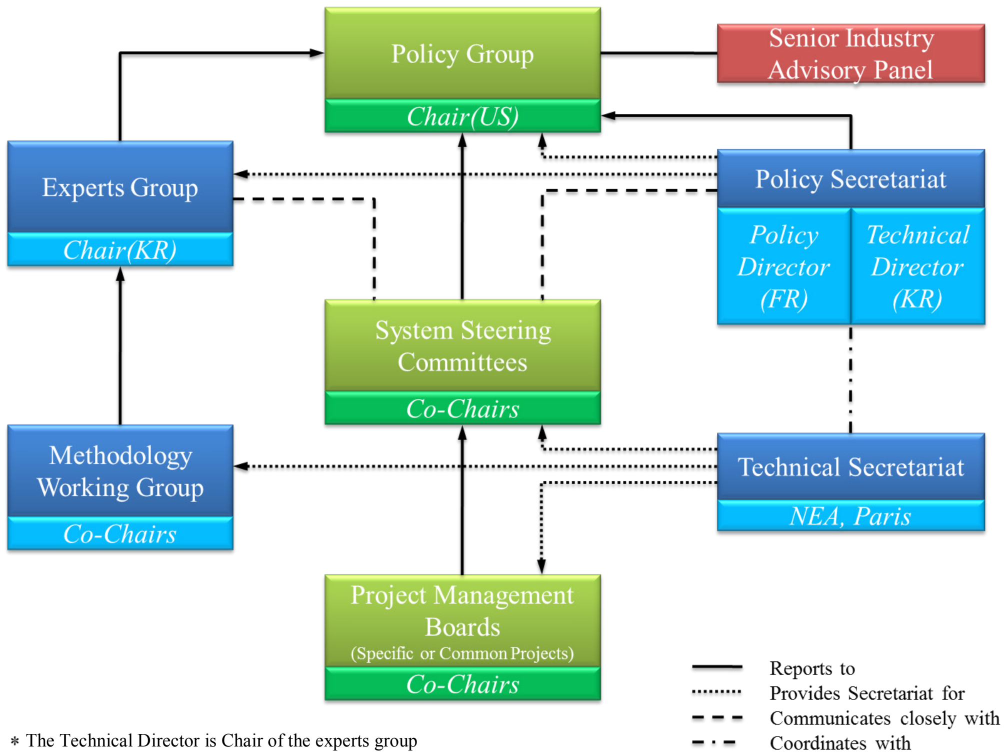
Among the signatories to the charter, 10 members (Canada, France, Japan, the People's Republic of China, the Republic of Korea, the Russian Federation, South Africa, Switzerland, the United States and Euratom) have signed or acceded to the framework agreement (FA) as shown in Table 2.1. Parties to the FA formally agree to participate in the development of one or more Generation IV systems selected by GIF for further research and development (R&D). Each party to the FA 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 the FA; accordingly, within the GIF, they are designated as "non-active members".

Members interested in implementing co-operative R&D on one or more of the selected systems have signed corresponding system arrangements (SA) consistent with the provisions of the FA. This is the case for the sodium-cooled fast reactor (SFR), the very high-temperature reactor (VHTR), the supercritical water-cooled reactor (SCWR) and the gas-cooled fast reactor (GFR). For the molten salt reactor (MSR) and the lead-cooled fast reactor (LFR) systems, memoranda of understanding (MOU) were signed in 2010 by France and EU, and EU and Japan, respectively. The Russian Federation signed the LFR MOU in 2011 and the MSR MOU in 2013. The participation of GIF members in SAs and MOU is also shown in Table 2.1.

1.2 GIF organisation

The GIF Charter provides a general framework for GIF activities and outlines its organisational structure. Figure 1.1 gives a schematic representation of the GIF governance structure and indicates the relationship among different GIF bodies which are described below.

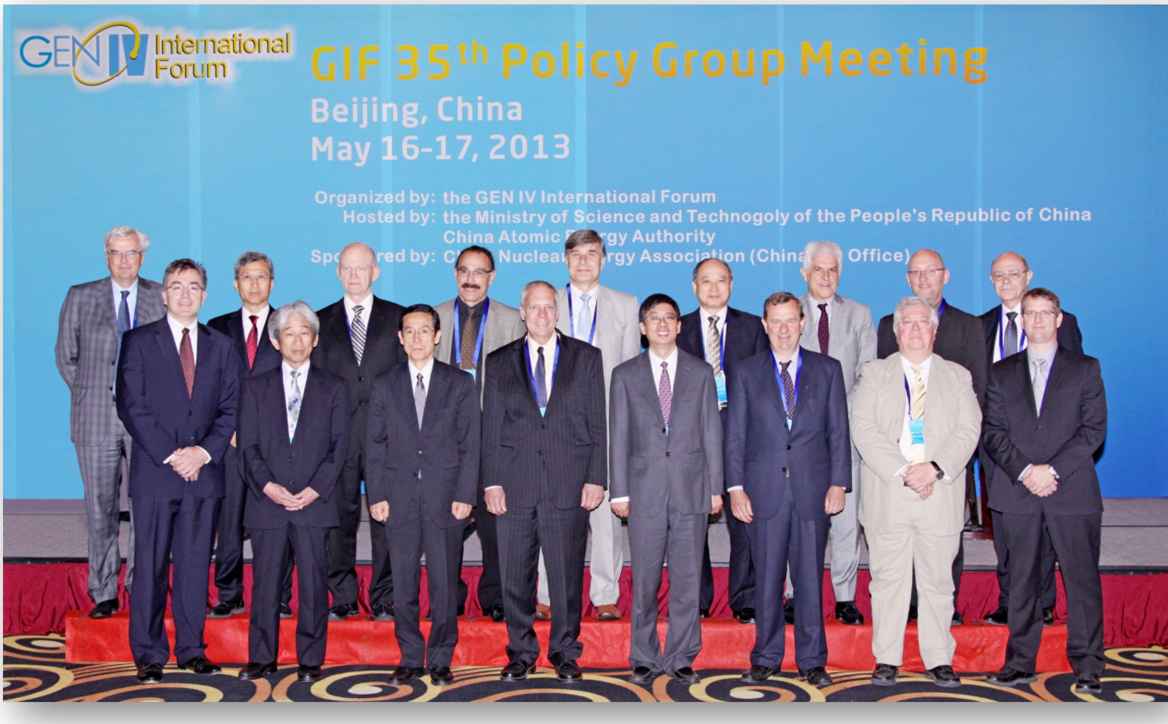
Figure 1.1: GIF Governance Structure in 2013



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 co-operative efforts, the establishment of policies governing GIF activities, and interactions with third parties. Every GIF member nominates up to two representatives in the PG. The PG usually meets two or three times each year (Figure 1.2).

Figure 1.2: Policy group in Beijing (May 2013)



The experts group (EG), which reports to the PG, is in charge of reviewing the progress of co-operative projects and of making recommendations to the PG on required actions. It advises the PG on R&D strategy, priorities and methodology and on the assessment of research plans prepared in the framework of SAs. Every GIF member appoints up to two representatives in the EG. The EG usually meets twice a year with meetings back to back with PG meetings in order to facilitate exchanges and synergy between the two groups.

Signatories of each SA have formed a system steering committee (SSC) in order to plan and oversee the R&D required for the corresponding system. R&D activities for each GIF system are implemented through a set of project arrangements (PAs) 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. Until the PA is signed, a provisional project management board (PPMB) oversees the information exchange between potential signatories. R&D carried out under a MOU (case of the LFR and MSR) is co-ordinated by a provisional system steering committee (PSSC).

The GIF Charter and FA allow for the participation of organisations from public and private sectors of non-GIF members in PAs and in the associated PMBs, but not in SSCs. Participation by organisations from non-GIF members requires unanimous approval of the corresponding SSC. The PG may provide recommendations to the SSC on the participation in GIF R&D projects by organisations 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 modelling working group (EMWG), the proliferation resistance and physical protection working group (PRPPWG), and the risk and safety working group (RSWG) – report to the EG which provides guidance and periodically reviews their work plans and progress. Members of the MWGs are appointed by the PG representatives of each GIF member.

In addition, the PG created dedicated task forces (TFs) to address specific goals or produce specific deliverables within a given timeframe. The progress status of two such TFs are described in this report, one dedicated to the development of safety design criteria for Generation IV systems, with a first focus on SFR, and the other dedicated to advanced simulation.

A senior industry advisory panel (SIAP) comprised of executives from the nuclear industries of GIF members was established in 2003 to advise the PG 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 co-ordinator of GIF activities and communications. It includes two groups: the policy secretariat and the technical secretariat. The policy secretariat assists the PG and EG in the fulfilment of their responsibilities. Within the policy secretariat, the policy director assists with the conduct of the PG whereas the technical director serves as chair of the EG and assists the PG on technical matters. The technical secretariat, provided by the Nuclear Energy Agency (NEA), supports the SSCs, PMBs, MWGs and TFs. 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 technical secretariat work).

1.3 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 SA. As noted previously, each SA is implemented by means of several PAs 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 (PP) consisting of specific tasks to be performed by the signatories.

In terms of PAs, a new Safety and Operation PA under the SFR SA became effective in November 2012, with China, Euratom and the Russian Federation joining the project. Canada withdrew from the VHTR SA and the Materials PA in December 2012, and South Africa withdrew from the same project in November 2012. Russia signed the MSR MOU in November 2013.

Table 1.2 shows the list of signed arrangements and provisional co-operation within GIF as of 31 December 2013.

R&D activities within GIF are carried out at the project level and involve all sectors of the research community, including universities, governmental and non-governmental laboratories as well as industry, from interested GIF and non-GIF members. Indeed, beyond the formal and provisional R&D collaborations shown in Table 1.2, many institutes and laboratories co-operate with GIF projects through exchange of information and results, as indicated in Chapter 2.

Table 1.2: Status of signed arrangements or MOU and provisional co-operation within GIF as of 31 December 2013

	Effective since	CA	EU	FR	JP	CN	KR	ZA	RU	CH	US
VHTR SA			X	X	X	X	X			X	X
HP PA	19-Mar-08	X	X	X	X	O	X			O	X
FFC PA	30-Jan-08		X	X	X	O	X				X
MAT PA	30-Apr-10		X	X	X	O	X			X	X
CMVB PA			P		P	P	P				P
SFR SA			X	X	X	X	X		X		X
AF PA	21-Mar-07		X	X	X	O	X		O		X
GACID PA	27-Sep-07			X	X						X
CDBOP PA	11-Oct-07		O	X	X	O	X		O		X
SO PA	11-Jun-09		X	X	X	X	X		X		X
SIA PA			P	P	P	P	P		P		P
SCWR SA		X	X		X				X		
M&C PA	6-Dec-10	X	X		X				O		
TH&S PA	5-Oct-09	X	X		X				O		
SIA PA		P	P		P				P		
FQT PA		P	P		O				O		
GFR SA			X	X	X					X	
CD&S PA	17-Dec-09		X	X						X	
FCM PA			P	P	P					P	
LFR MOU			X		X				X		O
MSR MOU			X	X	O	O	O		X		O

X = Signatory P = Provisional participant O = Observer

PROJECT ACRONYMS

AF Advanced fuel
CD&S Conceptual design and safety
CDBOP Component design and balance-of-plant
CMVB Computational methods validation and benchmarking
FCM Fuel and core materials
FFC Fuel and fuel cycle
FQT 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

Chapter 2. Highlights from the year and GIF member reports

2.1 General overview

At the end of 2013, the number of nuclear power reactors under construction in the world reached 72, the highest number since 1987. While the nuclear sector still faces many challenges such as financing, ensuring projects are completed on time and on budget, and of course, addressing public concern in the wake of the Fukushima Daiichi accident, nuclear energy remains a technology considered by many countries planning their energy policy. Nearly half of the reactors under construction are now Generation III reactors with higher safety and performance levels compared to previous generations. Generation III reactors will likely be the workhorse of nuclear power generation for several decades to come, but research to develop more innovative nuclear energy technologies and fuel cycles has already begun, and progress is being made as described in this 2013 edition of the GIF Annual Report.

The GIF has continued to work on the goals of achieving the highest levels of safety for the Generation IV systems, with the development of so-called Safety Design Criteria that incorporate lessons learnt from the Fukushima Daiichi accident. In February 2013, a first draft was discussed at a GIF/IAEA safety workshop on the SFR, and an improved version was approved by the Policy Group in May as a "phase 1" report. Guidelines which include quantification of the criteria are now being developed as part of the phase 2 of the work, while in parallel, the GIF has engaged with national regulators and organisations such as MDEP or the NEA's CNRA, inviting them to review the phase 1 report and provide feedback.

In terms of formal agreements to organise R&D efforts, 2013 saw the Russian Federation sign the Memorandum of Understanding overseeing the development of the Molten Salt Reactor, previously signed by Euratom and France in October 2010. China has been invited to attend the System Steering Committee meetings of the SCWR system as an observer. For the SFR system, a new Project Arrangement on System Integration and Assessment has been finalised and the signature process started in December 2013.

Finally, 2013 saw the conclusion of the Strategic Planning activity initiated in 2012, and in particular, with the finalisation of the *Technology Roadmap Update* which assesses progress made since the publication of the first *GIF Technology Roadmap Update* in 2002, and identifies key R&D challenges which need to be overcome to allow the more mature Gen IV systems to move towards their demonstration phase in the next decade.

2.2 Highlights from the experts group

Two EG meetings were held in 2013, in Beijing and in Brussels, two days before the PG meetings. One of the main topics on which the EG focused was the *Technology Roadmap Update* (TRU) report. The report was finalised and approved at the Brussels PG meeting. The role of the EG was to harmonise the extent and depth of the contents provided by the system steering committees and working groups. During the first half of the year, the EG provided high level review comments on the TRU, in particular on the issue of timelines for each system and on key objectives for the next 10 years. Towards the end of the year, the EG accepted the final version of the TRU with minor editorial changes and asked the PG for approval. The TRU was approved in November 2013 and was published in March 2014.

Another notable result of the EG's work relates to its contribution on the development of the SFR Safety Design Criteria. The EG provided necessary review comments when needed and successfully led the task force team to proceed into Phase II, which targets the development of Safety Design Guidelines. The EG is actively working with the Task Force team by suggesting expert recommendations on various topics.

The EG continued to monitor the systems' R&D activities and those of the methodology working groups through regular status reports presented at its meetings. To address the issue of deviations from planned deliverables, several specific actions were taken and the EG was able to understand and explain the deviations.

Various GIF activities were planned and executed under the EG's supervision. For instance, lessons learned from the GIF Symposium were discussed and topics for the SIAP were provided to that group. The special issue of *Progress in Nuclear Energy Journal* was prepared as well as the EG's contribution to the GIF Annual Report 2013. The finalisation of the implementation plans for the Strategic Planning activity and the review of collaborations with universities and other stakeholders were also handled in the framework of the EG and some are still under discussion.

The EG underwent a self-evaluation to identify how it can improve its role and activities within GIF. As a result, many suggestions were made including work assignments, demanding more contributions from each member, the formation of sub-groups etc. The EG will continue to provide expert opinions to the PG to address various GIF activities and will continue to focus on technical reviews of systems, working groups and task force teams.

2.3 GIF member reports

Canada

Nuclear power in Canada

The government of Canada is of the view that nuclear energy is a near emissions-free source of electricity that is safe, reliable and environmentally responsible, as long as it is developed within a robust international framework which adequately addresses security, non-proliferation, safety and waste management concerns. Nuclear energy is an important component of Canada's electricity sources.

In Canada, constitutionally nuclear energy falls within the jurisdiction of the federal government, but the responsibility for deciding the energy supply mix and investments in electricity generation capacity, including the planning, construction, and operation of nuclear power plants, resides with the provinces and their provincial power utilities.

Canada's existing fleet

In 2012, nuclear energy provided close to 15% of Canada's total electricity needs (over 50% in Ontario) and should continue to play an important role in supplying Canada with power in the future.

As of today, Canada has a fleet of 19 reactors in commercial operation with 18 of these reactors being located in the Province of Ontario and one unit in the Province of New Brunswick.

The Province of Ontario

In December 2013, Ontario released its updated Long-Term Energy Plan (LTEP) which stated that, due to lower forecasted electricity demand from changes in the economy and gains in conservation and energy efficiencies, the province will not proceed at this time with new nuclear builds. The government of Ontario is working with Ontario Power Generation (OPG) to maintain the site preparation licence granted by the Canadian Nuclear Safety Commission (CNSC), Canada's independent regulator, to preserve the option to build new nuclear in the future should the supply and demand picture change.

Ontario will go ahead with the refurbishment of the four existing reactors at the Darlington nuclear power plant (NPP) as well as six units at the Bruce NPP. These refurbishments will add about 25-30 years to the operational life of each unit. Refurbishment is to start in 2016 with one reactor at each station, and commitments on subsequent refurbishments will take into account the cost and timing of preceding refurbishments, with appropriate off-ramps in place.

The Province of Quebec

On December 28, 2012, Quebec's only operational nuclear powered generating station, Gentilly-2 (G2), ceased operations, and activities are underway to place G2 in a safe-shut down state.

In March 2013, the Quebec Minister of Sustainable Development, Environment, Wildlife and Parks announced that no certificate of authorisation would be issued for uranium exploration and mining projects in the Province until its environmental assessment agency issues a report on the environmental and social impacts of uranium exploration and mining in general (not expected before 2015).

Uranium production

Canadian uranium production totalled 8 998 tU in 2012, 16% of the total world production. All Canadian production is from mines located in the northern part of the Province of Saskatchewan.

Actions to strengthen Canada's nuclear industry

The Government of Canada continues to focus on strengthening Canada's nuclear industry. Highlights include:

AECL restructuring

In May 2009, the Government of Canada announced the restructuring of Atomic Energy Canada Limited (AECL). The first phase concluded with the divestiture of the CANDU Reactor Division, whose assets were sold to Candu Energy Inc., a wholly-owned subsidiary of the SNC-Lavalin Group Inc., in October 2011.

In early 2012, the Government of Canada launched the second phase of the restructuring of AECL focused on restructuring of the Nuclear Laboratories.

In February 2013, the Minister announced that Canada would undertake a competitive procurement process to seek a contractor to manage the operation of AECL's Nuclear Laboratories based on a Government-owned, Contractor-operated (GoCo) model. Going forward, the nuclear laboratories will focus on:

- managing radioactive waste and decommissioning responsibilities;
- performing science and technology activities to meet core federal responsibilities;
- supporting Canada's nuclear industry through access to science and technology facilities and expertise on a commercial basis.

The Government is also working to understand the potential business case for a forward-looking, industry-driven nuclear innovation agenda.

Nuclear liability

In June 2013, the Canadian Government announced its intention to bring forward legislation that will increase the civil liability of nuclear operators for nuclear damage to $1 billion from the current $75 million. Furthermore, in December 2013, to address the liability and compensation for damages arising from a potential nuclear incident with trans-boundary impacts, the Canadian Government signed the International Atomic Energy Agency's Convention on Supplementary Compensation for Nuclear Damage.

Radioactive waste management

Within Canada, there are currently two long-term radioactive waste management initiatives underway that may result in geological repositories.

- OPG is proposing to prepare, construct and operate a deep geological repository (DGR) facility on the Bruce Nuclear Site within the Municipality of Kincardine, Ontario. The DGR would be designed to manage low- and intermediate- level radioactive waste produced from the continued operation of OPG-owned Bruce, Pickering and Darlington nuclear generation stations within the Province of Ontario. This project is currently undergoing a federal environmental review assessment.
- The Nuclear Waste Management Organization (NWMO), established by the nuclear energy corporations, is seeking an informed and willing community with a suitable site to eventually prepare, construct, and operate a DGR facility for the long-term management of nuclear fuel waste. Over the next several years, the NWMO will continue its work with willing communities as they move through the siting process for this project.

Nuclear co-operation agreements

In September 2013, the Canada–India Nuclear Co-operation Agreement entered into force thus allowing Canadian firms to export and import controlled nuclear materials (including uranium), equipment and technology to and from India, subject to authorisations under the Nuclear Safety and Control Act and the Export and Import Permits Act.

China

Nuclear energy policy

China adheres to the policy of developing nuclear power in a safe and efficient manner. In accordance with "The Nuclear Power Safety Programme (2011-2020)" and "The Medium- and Long-term Nuclear Power Development Programme (2011-2020)" approved and released by the State Council in October 2012, the total installed capacity of nuclear power in operation in China will reach 58GW, and nuclear power capacity under construction 30GW by 2020. In the future, new nuclear power plants shall meet the Gen-III safety standard and be built in the light of the highest safety requirements in the world.

The National Development and Reform Commission (NDRC) announced to implement a benchmark price on newly constructed nuclear power units. In July 2013, NDRC promulgated that the current approach of nuclear power plants setting their own on-grid electricity prices should be changed to a benchmark price policy for all new NPPs, at a rate of 0.43 Yuan per kWh. The nuclear power benchmark price will be comparably stable, and in the future, NDRC will carry out regular assessment and adjustment of the benchmark price according to technology progress of nuclear power, cost changes and electricity market supply and demand status, etc.

The modified National Nuclear Emergency Programme was issued to enhance the nuclear emergency response. On 9 July 2013, the State Council announced the revised National Nuclear Emergency Programme. According to it, there are four levels of emergency: emergency standby, plant emergency, site emergency and off-site emergency (overall emergency), and for each kind of emergency a respective response will be initiated. Meanwhile, the Programme stipulates that the nuclear emergency organisation system will mobilise responding task forces at the national, provincial and operating company levels. Information transparency and timely announcement are also required by the Programme.

In-service nuclear power units are safely operating and construction projects continue on schedule

Two new nuclear units, Ningde Unit 1 and Hongyanhe Unit 1, were put into commercial operation on 12 April 2013 and on 7 June 2013 respectively. By the end of 31 December 2013, there were 17 nuclear power units at five sites in operation, with the total installed capacity of 14.69 GW. The total nuclear power generation amounted to 108 TWh, 12% increase on last year.

All operating nuclear power units maintained good safety records, and their main operational performance indicators reached the international advanced level.

Three new nuclear reactors including Yangjiang Unit 5, Unit 6 and Tianwan Unit 4 started construction in 2013. Construction continues on four AP1000 reactors at two projects in Sanmen and Haiyang. These reactors are the first of a new generation of passively safe nuclear plants. Two EPR reactors under construction are also on schedule at the Taishan site. By the end of December 2013, there were 31 nuclear power units under construction, with a total installed capacity of 33.85 GW.

Research and development on high temperature gas-cooled reactor (HTR) has made great progress

The construction of China's HTR nuclear power plant demonstration project (HTR-PM) was launched in December 2012. The current estimate for its completion is 2017. The engineering design, equipment design, full-scale validation test, fuel element manufacture and irradiation R&D are being carried out as scheduled. In March 2013, its fuel element production line began construction, with an aim to produce fuel element in 2015.

In October 2013, China Atomic Energy Authority (CAEA) agreed the amendment to the project arrangement on FFC for the international research and development of VHTR nuclear energy system and approved the Institute of Nuclear & New Energy Technology (INET) of Tsinghua University to join the fuel and fuel cycle (FFC) project of VHTR nuclear energy system.

Research and development on sodium-cooled fast reactor is underway

According to the China Fast Reactor development plan, the concept design of demonstration project of China Fast Reactor (CFR600) has already been undertaken. China Institute of Atomic Energy (CIAE) has taken part in the R&D programme of system integration and assessment (SIA) for the international research and development of SFR nuclear energy system. Signing of the project arrangement was completed in December 2013.

Research and development on super-critical water-cooled reactors (SCWR) is underway

The sixth international supercritical water reactor symposium (ISSCWR-6), jointly hosted by the China General Nuclear Power Corporation and China Nuclear Energy Association, was successfully held in Shenzhen, China in March 2013. More than 120 representatives from over 10 countries and international organisations exchanged recent research results concerning the core and fuel design, materials, chemistry and corrosion, thermal-hydraulics and safety analysis of supercritical water reactor. Following the symposium, the GIF SCWR SSC meeting was held on March 8, 2013. A Chinese representative presented "The summary of SCWR research and development activities in China".

The 2nd Technical Meeting on Materials and Chemistry for Super-Critical water-cooled Reactors (SCWRs) (TM-44718), hosted by Nuclear Power Institute of China (NPIC), was held in Chengdu, China in July 2013. Participants from six Member States and observers from NPIC exchanged the technical and experimental results in the area of materials and chemistry of SCWR. The participants recommended that the IAEA start a co-ordinated research project (CRP) on this topic to accommodate Member States' efforts in R&D of SCWR materials and chemistry issues.

The conceptual design and relevant R&D at phase-1 stage for the Chinese SCWR, called CSR1000, have been completed. The assessment organised by CAEA for the project of "R&D on SCWR technology at phase-1 stage" with an aim for developing an industrial level SCWR design was conducted by an expert group in November 2013.

In addition, a bilateral co-operation project on the fuel qualification test project between Euratom and Chinese consortia continued throughout 2013.

China GIF Liaison Office

The Chinese government has entrusted the China Nuclear Energy Association (CNEA) to fulfill the function of China GIF Liaison Office. The role of China GIF Liaison Office includes:

- Co-ordinating and organising Chinese participation in GIF activities.
- Establishing regular information exchange mechanism.
- Hosting work meetings regularly to discuss and deploy GIF tasks at home.
- Developing and setting up of the China GIF website in Chinese language with an aim to promote the better and safer development of nuclear power. Through the website, Gen IV nuclear energy knowledge and technology are disseminated timely to governmental bodies, research institutes, universities, nuclear industry players and the public.

Other GIF activities

China hosted the 35 GIF Policy Group meeting and the 29 GIF Expert Group Meeting from 14-17 May, 2013 in Beijing, China. Participants visited the HTR-10 facilities at the Institute of Nuclear and New Energy Technology of Tsinghua University on 15 May, 2013. At the PG meeting, Chinese representative expressed the interest in joining GIF SCWR system arrangement. The internal approval procedures for joining GIF SCWR system arrangement were initiated.

Euratom

Post-Fukushima EU nuclear power plants safety evaluations ("stress tests")

Following the Fukushima Daiichi accident in March 2011, the European Council called for comprehensive and transparent risk and safety assessments of all EU nuclear power plants ("stress tests"). The main aim of the stress tests was to assess the safety and robustness of nuclear power plants in case of extreme natural events (flood, earthquakes and extreme external events). The stress tests were carried out in 2012 and all countries identified: analysis needs; hardware improvements; procedural modifications; and regulatory actions, and proposed implementation schedules in their specific National Action Plans (NAcPs). The European Commission is now following the implementation of the recommendations and country action plans. In April 2013, the first dedicated workshop was held in Brussels in order to peer review the contents and status of implementation of the NAcPs. The "EU NPPs stress tests" review and the NAcP workshop recognised the importance of the Periodic Safety Review process as a powerful tool to be used for continuous improvement of nuclear power plants. Maintaining containment integrity under severe accident conditions also was stressed and is an important issue for accident management. All participating countries also strongly committed to the issue of transparency of their work and demonstrated related improvements.

EU Nuclear Safety Directive

A common nuclear safety legal framework has been set up in European Union through the EU Council's adoption of the Nuclear Safety Directive on 25 June 2009. With it, the EU has become the first major regional nuclear actor to provide legally binding framework for nuclear safety largely based the Fundamental Safety Principles established by the IAEA and the obligations originating from the IAEA's Convention on Nuclear Safety. In 2013, the European Commission proposed legislative measures to further enhance nuclear safety in Europe. The proposed amendments on the existing Nuclear Safety Directive focus on:

- strengthening the role and effective independence of the national regulatory authorities;
- enhancing transparency on nuclear safety matters;
- strengthening existing principles, and introducing ambitious nuclear safety objective for the EU's nuclear installations;
- reinforcing monitoring and exchange of experiences, by establishing a European system of peer reviews.

The new legislative measures will clearly impact the development of EU Generation IV concepts as concerns the design safety provisions.

Symposium on Benefits and Limitations of Nuclear Fission for a Low-carbon Economy

In February 2013, the European Commission held a symposium on Benefits and Limitations of Nuclear Fission for a Low-carbon Economy to provide answers to pressing questions concerning Europe's nuclear research policy for the next seven years within the frame of the Horizon 2020 programme. Among the ten Recommendations that were identified, the need to promote new emerging technologies, to support safety and security, and to participate to international discussions and groups were highlighted. These activities were and will be pursued by Euratom in the frame of GIF.

Euratom Horizon 2020 framework programme

The Euratom Framework Programme (FP) for nuclear research and training activities supports EU research in nuclear fission and fusion. The FP7+2 research programme that was funding the Euratom Gen IV research as part of GIF collaborations finished at the end of 2013 and a new seven-year EU research programme called Horizon 2020 has been agreed and adopted by the EU Parliament and EU Council. Regarding the activities in nuclear fission, there will be stronger emphasis, as far as research on Generation IV is concerned, on safety and security issues.

Sustainable Nuclear Energy Technology Platform (SNE-TP)

The EU has committed for the year 2020 to: reduce by 20% its GHG emissions (compared to 1990), make 20% energy savings and include 20% share of renewable energies in its total energy mix, aiming in the long-term to attain a low-carbon economy in Europe. To reach these goals, the EU Strategic Energy Technology Plan (SET-Plan) identifies a set of competitive low-carbon energy technologies to be developed and deployed in Europe, with nuclear fission representing a key contribution. In this frame, the Sustainable Nuclear Energy Technology Platform (SNE-TP) promotes research, development and demonstration of the nuclear fission technologies.

The SNE-TP gathers about 80 European stakeholders from industry, research and academia, technical safety organisations, non-governmental organisations and national representatives. As concerns innovative technologies the SNE – TP focuses on three GIF systems: the SFR, LFR and GFR with emphasis on safety (as a consequence of the discussions at EU Member States level after Fukushima). In 2013 several European projects linked to the Gas Fast Reactor and to the Lead Fast Reactors were completed and the results were shared with GIF partners. Further Euratom GIF activities will be aligned with the EU SNE-TP priorities.

France

National debate on energy policy

In France, a national debate on energy policy took place during the first half of this year. This debate will be the background of a law which could be enacted by spring 2014. Nuclear energy will remain the main pillar of the French energy mix, more specifically:

- No nuclear power plant other than Fessenheim will be phased out.
- The construction of the EPR at Flamanville is confirmed.
- The choice of the closed fuel cycle is confirmed.

Post Fukuskima safety evaluations

Complementary safety evaluations, so-called stress tests, have been carried out on all of the French nuclear facilities. ASN, the French Nuclear Regulator, published in December 2012 a report to give instructions to the French nuclear operators (mainly EDF, AREVA, and CEA). The report presents the three basic topics (natural hazards, loss of safety functions, and severe accidents) that have structured the stress-tests. It also covers nuclear regulation, emergency management, international co-operation, and subcontracting. It finally gives a roadmap of key activities that have been engaged. This roadmap will be submitted to the European regulator for approval.

Long-term operation

Concerning the current French nuclear fleet, EDF has decided to launch an ambitious refurbishment programme called "Grand Carenage" (i.e. Major Refit), in order to prepare the extension of operation lifetime and to improve safety taking into account the post-Fukushima accident feedback. The total budget of this programme is about Euros 50 billion, 10 of which are devoted to post-Fukushima safety upgrades.

Waste management

A public debate concerning the construction of a deep geological waste repository took place from May to December 2013. It is consistent with the Waste Management Act of 2006. The debate allowed ANDRA, the French National Agency for Radioactive Waste Management, to explain the progress, since 2006, of the so-called CIGEO project, mainly on industrial design, safety, reversibility, site location and monitoring. The target is to start construction in 2018, for operation in 2025

ASTRID programme

Concerning the ASTRID project, two important milestones were achieved in 2013: 1) the decision, after the release of the 2012 reports, to continue the design work and the R&D of ASTRID. The next important milestone is the end of the preliminary design phase in 2015; 2) a formal review of Safety Orientations was carried out by the safety regulatory body in June. The safety approach has been endorsed, and recommendations were made to study more in detail some topics such as sodium aerosol release evaluation, and the post-Fukushima safety approach. The next step is the Safety Option Report to be released at the end of 2015.

Miscellaneous

Even though they are not related to the French nuclear programme, two important decisions deserve to be pointed out:

- The decision of the United Kingdom to build two EPRs and to guarantee the selling price of electricity, which is the most important decision regarding the use of nuclear energy in Western Europe since Fukushima.
- The decision of Turkey to launch exclusive negotiations in order to build four MHI-AREVA ATMEA-1 reactors for the country's second nuclear power plant in Sinop.

Japan

TEPCO's Fukushima Daiichi NPS

TEPCO revised the mid-and-long-term roadmap in June to accelerate the work towards decommissioning. The start of removal work of fuel debris was pushed forward one-and-a-half years earlier than the initial plan – the target is now the first half of fiscal year 2020. The government decided to play a further proactive role in countermeasures against the contaminated water in September. The removal of fuel sub-assemblies in the spent fuel pool of No. 4 reactor started in November.

Japanese energy policy

The government is currently reviewing from scratch the policy to enable zero operation of nuclear power plants in the 2030s decided by the former administration (of the Democratic Party of Japan), and developing the new basic energy plan with responsibility from the viewpoint of securing stable energy supply, reducing energy costs, and so on. Members of a government panel broadly agreed to draft a long-term national energy plan stating that nuclear power is an important source of electricity in Japan.

Operational status of LWRs and activities of Nuclear Regulation Authority (NRA)

The Kansai Electric Power Company's Ohi No. 3 and No. 4 reactors were shut down for regular inspections in September, which means all the current 50 reactors are offline.

New regulatory requirements for commercial nuclear power reactors took effect on July 8. They cover: severe accident measures; measures against large-scale natural disasters such as tsunamis and earthquakes, terrorist attacks including aircraft crashes; prohibition of construction of reactor buildings and other key facilities above active faults; and "back-fitting" system where the new regulations are applied to the existing NPPs. Fourteen units are being evaluated to assess the conformity to the new standards.

JAEA update

In light of the faulty maintenance checks at Monju last year and the radiation leak at the Japan Proton Accelerator Research Complex (J-PARC) in May, the reform of the JAEA is underway. The managing structure of Monju was reformed in order to enhance the governance credentials. The role of Monju was reassessed from the technological perspective by the MEXT and the "Research Plan for Monju" was issued. It focuses on 1) compiling the results of FBR development, 2) reducing the amount and toxic level of radioactive waste, and 3) strengthening the safety of FBRs. These aspects are reflected in the discussion of the new "Basic Energy Plan of Japan."

Republic of Korea

Energy policy

The new administration of President Park Geun-hye officially started its term on February 25, 2013. The government tasks related to nuclear energy are in five categories:

- promoting creative industry through the scientific technology;
- supporting overseas plant construction and nuclear power industry exports;
- consolidating nuclear safety management systems;
- strengthening the safety management of energy supply utilities;
- securing the energy supply and advancing the industrial structure.

Early this year, the level of development of nuclear energy was expected to continue for the next 5 years. However, in October, a civil energy consultation group recommended that the share of nuclear power in total electricity generation in Korea be kept between 22% and 29% by 2035. This was far lower than the proposal of previous administration for 2008-2030. The current share of nuclear power generated by the installed capacity is about 26.4% and the recommendation suggests maintaining the current level of share for the next two decades. The final decision was made to increase it up to 29%, which is the highest value recommended.

Waste management

Regarding the interim storage of spent fuel, the 2nd Atomic Energy Promotion Commission held in November 2012 confirmed that the spent fuel interim storage policy will be determined by public consensus. Accordingly, "Public Engagement Commission on Spent Nuclear Fuel Management" is officially established on 30 October, 2013, under the auspices of the Ministry of Trade Industry and Energy (MOTIE) according to the Radioactive Waste Management Act of 2009.

Nuclear fleet

In May, two NPP site names were changed: "Yong-gwang" and "Ul-jin" to "Hanbit" and "Hanul", respectively. This is the first time regional information is removed from a site's name. The name change was the result of Korea Hydro and Nuclear Power (KHNP) Company's wish to respect the demands of local residents who had suffered from the bad image of nuclear power plants for a long time.

SFR and VHTR research and development

R&D on SFR and VHTR has continued to make significant progress. Based upon the experience gained during the development of the conceptual designs for KALIMER, KAERI has carried out the preliminary design of a prototype SFR in 2013. An important step towards enabling the licence process of SFR is taken recently. Namely, a decision was made to immediately organise a SFR-specialised committee under the auspices of nuclear safety committee, tasked with drafting amendments to the existing nuclear related laws and technology standards in provision of SFR generic safety analysis report submittal planned in 2017 for the specific design of PGSFR. In July, the licence issue of the firefighting agency on sodium experimental facility, STELLA, has been resolved and it is now in the inspection phase. The test results are expected at the end of this year.

VHTR is primarily dedicated to the generation of hydrogen heat applications in the Republic of Korea. In 2013, lab-scale pressurised sulfur-iodine hydrogen production process is to be demonstrated. Also, irradiation test of the TRISO fuel manufactured by KAERI has started in the HANARO research reactor. The VHTR conceptual design study, started in 2012 under a cost-sharing programme with industry, will produce the business plan of the nuclear hydrogen development and demonstration project towards the end of 2014.

Russian Federation

Status of nuclear power and technology development

In the Russian Federation, there are 33 nuclear power units in operation located at 10 NPP sites. The total electric power capacity of all Russian NPPs is equal to 25 242 GWe. Now, over twenty power units are planned to be constructed and are under construction, including nine power units in Russia. Last year, the total electricity production by NPPs in Russia was more than 177 billion kW h, representing 16.8% of Russia's total electricity production. In 2012, the load factor of the Russian NPPs reached 80.9%. After the accident at the Fukushima Daiichi NPP, analysis and enhancement of safety of all the operating, constructed and designed Russian NPPs have been performed in relation to similar events.

Activities on innovation reactor technologies are mainly carried out in Russia within the framework of a specific Federal Target Program (FTP) "Nuclear power technologies of a new generation for period of 2010-2015 and with outlook to 2020". The objective of the FTP is in the development and creation of the new technological platform for the nuclear power based on the transition to the closed nuclear fuel cycle (CNFC) with the 4th generation fast reactors. Within the FTP framework, activities is provided in area of sodium cooled fast reactors (SFRs) and fast reactors with heavy liquid metal coolant (HLMC), and fuel cycles related to them.

Sodium-cooled technologies

Concerning SFR technology, two facilities are operating successfully in Russia:

- industrial power unit BN-600 (more than 33 years);
- test reactor BOR-60 (more than 44 years).

The lifetime of the BN-600 power unit was extended up to the end of March 2020, and the BOR-60 lifetime has been extended to the end of 2014. The construction of a power unit with the BN-800 reactor is underway on the Beloyarsk NPP site, Start of operation is scheduled in 2014.

In accordance with the FTP, the following development is being carried out:

- design of a large-size power unit with the sodium-cooled BN-1200 reactor, that should meet the requirements for the 4th generation nuclear energy systems;
- design of a multifunctional research fast reactor MBIR with sodium coolant that should substitute the BOR-60 reactor.

An international research centre on the basis of the MBIR reactor is planned. According to the FTP, the BN-1200 design is scheduled to be finalised in 2014, MBIR reactor should be put into operation in 2019 in the RIAR (Dimitrovgrad). Construction of the first-of-a kind power unit with the BN-1200 reactor at the Beloyarsk NPP site is being discussed.

Fast reactors with heavy liquid metal coolant

The FTP also envisages the development of designs of the BREST-OD-300 reactor with lead coolant and of the SVBR-100 reactor with lead-bismuth coolant, as well as the construction of their prototype facilities aimed at proving feasibility of these reactor technologies with HLMC. Development of the given designs will be finalised by the end of 2014. Construction of the demonstration facility with the BREST-OD-300 reactor and with on-site CNFC is planned to be implemented at the second stage of the FTP in 2015-2020 on the site of the Siberian Chemical Complex in Tomsk.

Other research activities

In addition, the FTP roadmap envisages:

- Implementation of a large-scale refurbishment of the Rosatom experimental base, including construction of the MBIR reactor and upgrading the fast critical facilities (BFS).
- Creation of industrial basis for fuel production for the advanced reactor facilities and its reprocessing under the CNFC conditions, in particular, production of the MOX-fuel for fast reactors of the new generation, including the BN-800 reactor is under development, activities on creation of the pilot industrial complex for production of the dense nitride fuel as well as demo semi-industrial pyrochemical complex is performed.
- Development of computational codes for justification of the parameters and safety of advanced new generation nuclear energy systems.

Prospective research is also underway in Russia on supercritical-water-cooled reactor, molten salt reactor, and on the concept of gas-cooled fast reactor.

The State Corporation “Rosatom” signed the MOU on MSR on 12 November 2013.

South Africa

In November 2012, the Cabinet of the Government of South Africa endorsed the Phased Decision-Making Approach for implementation of the nuclear programme. It also endorsed the designation of Eskom as the owner-operator as per the Nuclear Energy Policy of 2008. Cabinet also approved the nuclear communication and stakeholder engagement strategy.

Following the successful IAEA Integrated Nuclear Infrastructure Review (INIR) pre-mission in October 2012, South Africa also conducted full INIR Mission during 28 January to 8 February 2013, with the final report from the IAEA received in May 2013.

The INIR Mission also coincided with the visit of the Director General of the IAEA, Mr Yukiya Amano, to South Africa during 7 to 9 February 2013. This visit included tours of the Koeberg Nuclear Power Station and the SAFARI-1 nuclear reactor.

During November 2013, a high level Government delegation, comprising of the Minister of Energy, Minister of Public Enterprises and executives of state owned companies, conducted study tours to a number of key nuclear jurisdictions. The Government has also indicated its intention to make an announcement on the procurement of nuclear power plants by March 2014.

Switzerland

Switzerland has taken the decision to phase out nuclear energy by not building new plants and therefore not replacing the currently operating four nuclear power plants. While the duration of the remaining operation time of the Swiss nuclear reactors should be determined by safety considerations only according to the Swiss licensing regime, it is even though generally assumed for planning purposes in politics that the reactors would be shut down after 50 years of operation. This would bring the date of the closure of the last Swiss nuclear power plant to 2034.

In summer 2013, one utility (BKW-FMB) has announced that it will shut down the Mühleberg BWR-4 by 2019, after 47 years of nuclear operation. This decision was influenced by many factors, the economical ones being central. Another utility (AXPO) is in the process of replacing the reactor vessel heads of the two-unit Beznau PWR and extending the emergency power generation capability with an autarky of seven days. (Beznau-I is operational since 1969 and Beznau-II is operational since 1971.)

In response to the policy change, the key mission of the Nuclear Energy and Safety Department at PSI is to maintain nuclear competence for the foreseeable future. The scientific support to the Swiss Nuclear Regulator will remain a key element of this mission. Another key element represents R&D related to waste management. Strong support to Nuclear Education (with three university professors and many senior scientists as lecturers) will continue. In this new context, a fraction of our R&D resources will be devoted to innovative nuclear technology, and a MSR concept was identified as one promising target.

United States

Nuclear energy continues to be a vital part of the United States "all-of-the-above" energy strategy for a sustainable, clean energy future. Responsibility for advancing nuclear power as a part of this strategy resides with the Office of Nuclear Energy (NE) within the United States Department of Energy (DOE).

Included in this strategy is the small modular reactor (SMR) programme, a six-year, $452 million programme to support the licensing of mature SMR designs, with a goal of realising domestic deployment in the 2022 to 2025 timeframe. In November 2012, DOE announced the selection of Babcock & Wilcox (B&W) for cost-shared investment to support the design development, certification and licensing activities of B&W's mower reactor. In December 2013, DOE announced the selection of Unscaled Power LLC to receive a financial assistance award to support efforts to design, certify and commercialise Musicale's SMR design. Overall, the SMR programme supports the licensing of innovative designs that improve SMR safety, operations and economics through lower core damage frequencies, longer post-accident coping periods, enhanced resistance to natural phenomena, and potentially smaller emergency preparedness zones.

As is the case across all US Government programmes, DOE has to manage with reduced budgets as it carries out its mission. To be more effective, NE is utilising input from industry, academia, and National Laboratories to ensure that the available resources for nuclear energy R&D activities are effectively allocated. The United States continues to strongly believe that leveraging our financial and technical resources through engagement in GIF and other multilateral forums, as well as through bilateral co-operation, is important.

NE's advanced reactor programme, under its Office of Advanced Reactor Technologies, performs research to develop technologies and subsystems that are critical for advanced concepts that could dramatically improve nuclear power performance through the achievement of GIF goals on sustainability, economics, safety, and proliferation resistance. These efforts can be broadly captured in five distinct areas: fast reactors, high temperature reactors, licensing strategies, advanced studies and advanced generic reactor technology, which includes high temperature metals, Instrumentation and Controls, and energy conversion systems. Included within all these topics is an emphasis on advanced non-light water reactors that are small and modular.

Within the past year two notable efforts are underway. First, in regards to TRISO coated particle fuel development for high temperature gas cooled reactors, post irradiation examination of fuel from the first experiment (AGR-1) and elevated temperature testing of irradiated AGR-1 fuel to temperatures that exceed postulated accident conditions of 1 600°C continues. Several tests have been completed at 1 700°C and 1 800°C for over 300 hours with no releases from intact particles thereby demonstrating the robustness of the fuel. On 16 October, 2013 the second TRISO coated particle fuel irradiation experiment AGR-2 was completed with approximately 559 effective full-power days of irradiation, without any fuel failures. Irradiation of the third and fourth (AGR-3/4) TRISO fuel experiment is scheduled for completion in June 2014.

The second important effort pertains to the development of advanced non-light water reactor licensing strategies. NE has been working with the US Nuclear Regulatory Commission (NRC) over the past several years on a licensing strategy for high-temperature gas-cooled reactors. The NRC will soon be issuing staff positions on licensing basis event selection, mechanistic source term, functional containment performance requirements and emergency planning. Expansion of this effort to address advanced reactor concepts more broadly have been initiated to develop Generic Design Criteria that will be used to develop technology specific design criteria for advanced non-light water reactors. The NRC is also reviewing the GIF sodium-cooled fast reactor safety design criteria.

In support of the nuclear energy industries long term viability, NE is working to train the next generation of nuclear engineers and scientists. To that end, NE funds a number of research activities at US universities. Through our Nuclear Energy University Programs (NEUP), NE is currently funding an Integrated Research Project (IRP) to examine a path forward for deployment of fluoride salt-cooled high temperature reactors (FHRs). We have also recently awarded an IRP to evaluate reactor materials at very high irradiation doses by using microstructures and properties developed under accelerated irradiation from multiple ion beams to benchmark the methods needed to predict those generated under high dose neutron irradiation. In October 2013, NE issued a new solicitation for research to be conducted at US universities on a broad range of advanced non-light water reactor research areas, and also included a new IRP solicitation to further examine technology challenges associated with FHRs.

From a commercial sector standpoint, construction continues on four AP1000 reactors at two projects in Georgia and South Carolina, which represent a new generation of passively safe nuclear plants. Current estimates for completion for all four units are in the 2017 to 2019 timeframe. Construction also continues on the Tennessee Valley Authority Unit 2 reactor at its Watts Bar plant. Commercial operation is set for late 2015. The unit will be the first US reactor to enter service since Unit 1 began operating in 1996.

Meanwhile, US utilities continue to respond to the orders and requests for information (RFIs) issued by the NRC in March 2012 in response to the recommendations of the agency’s post-Fukushima Near-Term Task Force. US nuclear reactor operators provided their integrated plans for complying with three NRC orders in February 2013. In addition, the industry is continuing to implement its “FLEX” approach for responding to extreme external events leading to station blackout and involving multiple units.

On 11 January 2013, the US Department of Energy released its strategy for the management and disposal of used nuclear fuel and high-level radioactive waste, which responds to the final report and recommendations made by the Secretary of Energy’s Blue Ribbon Commission on America’s Nuclear Future (BRC). The Strategy provides a framework for moving toward a sustainable programme to deploy an integrated system capable of transporting, storing, and disposing of used nuclear fuel and high-level radioactive waste from civilian nuclear power generation, defence, national security and other activities.

Chapter 3. System reports

This chapter gives a detailed overview of the achievements made in 2013 in the research and development activities carried out under the four system arrangements (VHTR, SFR, SCWR, GFR) and under the two MOUs (LFR and MSR).

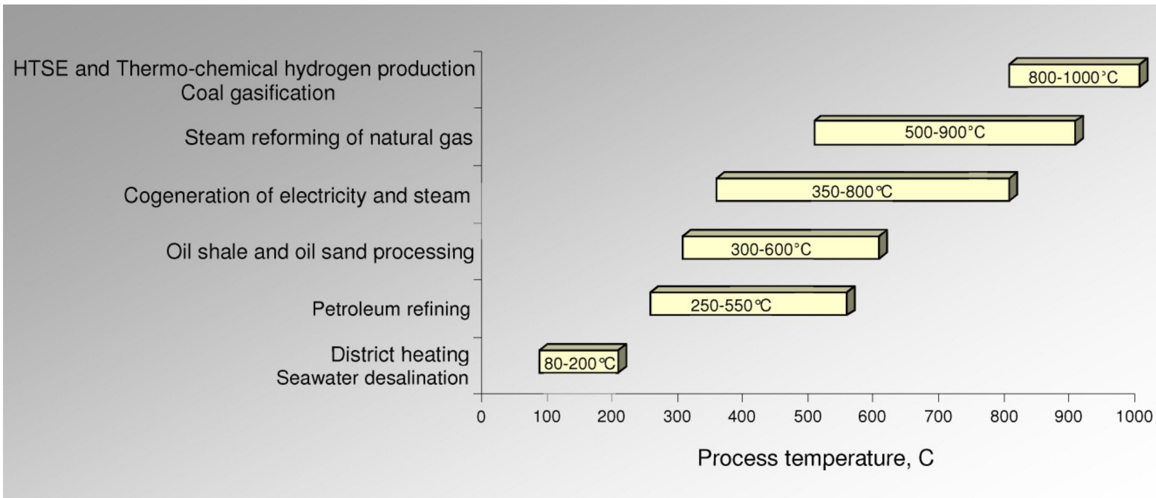
3.1 Very-high-temperature reactor (VHTR)

3.1.1 Main characteristics of the system

The very-high-temperature reactors are the descendants of the high temperature reactors developed in the 1970s-1980s. They are characterised by a fully ceramic coated particle fuel, the use of graphite as neutron moderators and helium as coolant, resulting in inherent safety and process heat application capability.

Use of helium as coolant and ceramics as core structure material allows operation at temperatures as high as 1 000°C at core outlet. This opens the possibility of producing hydrogen using processes with no green-house gas emission, such as thermochemical cycles (Iodine Sulfur) or high temperature steam electrolysis (HTSE). Beyond electricity generation and hydrogen production, high temperature reactors can provide process heat for use in other industries, substituting fossil fuel applications (Figure 3.1).

Figure 3.1: Industrial applications versus temperatures



Courtesy: Phil Hildebrandt, Battelle Energy Alliance, Global Petroleum Conference, June 11, 2008

As previously noted, the basic technology for the VHTR has been established in former high-temperature gas reactors such as the US Peach Bottom and Fort Saint-Vrain plants as well as the German AVR and THTR prototypes, also Japanese HTTR test reactor and Chinese HTR-10 test reactor. These reactors represent the two baseline concepts for the VHTR core: the prismatic block-type and the pebble bed-type. The fuel cycle will initially be once-through with low-enriched uranium fuel and very-high-fuel burnup, and also possibly be plutonium-based fuel or thorium-based fuel. Solutions need to be developed to adequately manage the back-end of the fuel cycle and the potential for a closed fuel cycle also needs to be fully established. Although various fuel designs are considered within the VHTR systems, all concepts exhibit extensive similarities allowing for a coherent R&D approach, as the TRISO coated-particle fuel form is the common denominator for all. This fuel consists of small particles of nuclear material, surrounded by porous carbon buffer, and coated with three layers: pyro-carbon/silicon carbide/pyro-carbon. This coating represents the first barrier against fission products release.

Former reactors were operated already at temperature up to 950°C (high-temperature reactors). VHTR can supply nuclear heat and electricity over a range of core outlet temperatures between 700 and 950°C, or more than 1 000°C in future. The available high-temperature alloys used for heat exchangers and metallic components determine the current temperature range of VHTR (~700-950°C). The final target for GIF VHTR has been set at 1 000°C or above, which implies the development of innovative materials such as new super alloys, ceramics and compounds. This is especially needed for some non-electric applications, where a very high temperature at the core outlet is required to fulfil the VHTR objective of providing industry with very high temperature process heat.

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

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

The technology is being advanced through near and medium-term projects, such as HTR-PM, NGNP, GT-MHR, NHDD, and GTHTR300C, led by several plant vendors and national laboratories respectively in the People's Republic of China, the United States, the Republic of Korea and Japan. The construction of HTR-PM demonstration plant (two pebble bed reactor modules with one super heat steam turbine generating 200MWe) has been started in China (Figure 3.2) on 9 December 2012. Each reactor module will have a power of 250 MWth. The coolant gas temperature will be 750°C, which represents the current state of the art for materials and the requirement of high temperature steam generation. High quality steam of 566°C will be fed into a common steam header. HTR-PM demonstration plant will connected to the grid in 2017, which will represent a major step toward a Generation IV demonstration plant.

Status of co-operation

The VHTR system arrangement 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. South Africa, which has expressed high interest in the VHTR, formally acceded to the GIF framework agreement in 2008, but announced in December 2011 that it no longer intends to accede to the VHTR SA. Canada withdrew from the SA at the end of 2012.

The fuel and fuel cycle project arrangement became effective on 30 January 2008, with implementing agents from Euratom, France, Japan, the Republic of Korea and the United States. The project arrangement has been extended to include input from China.

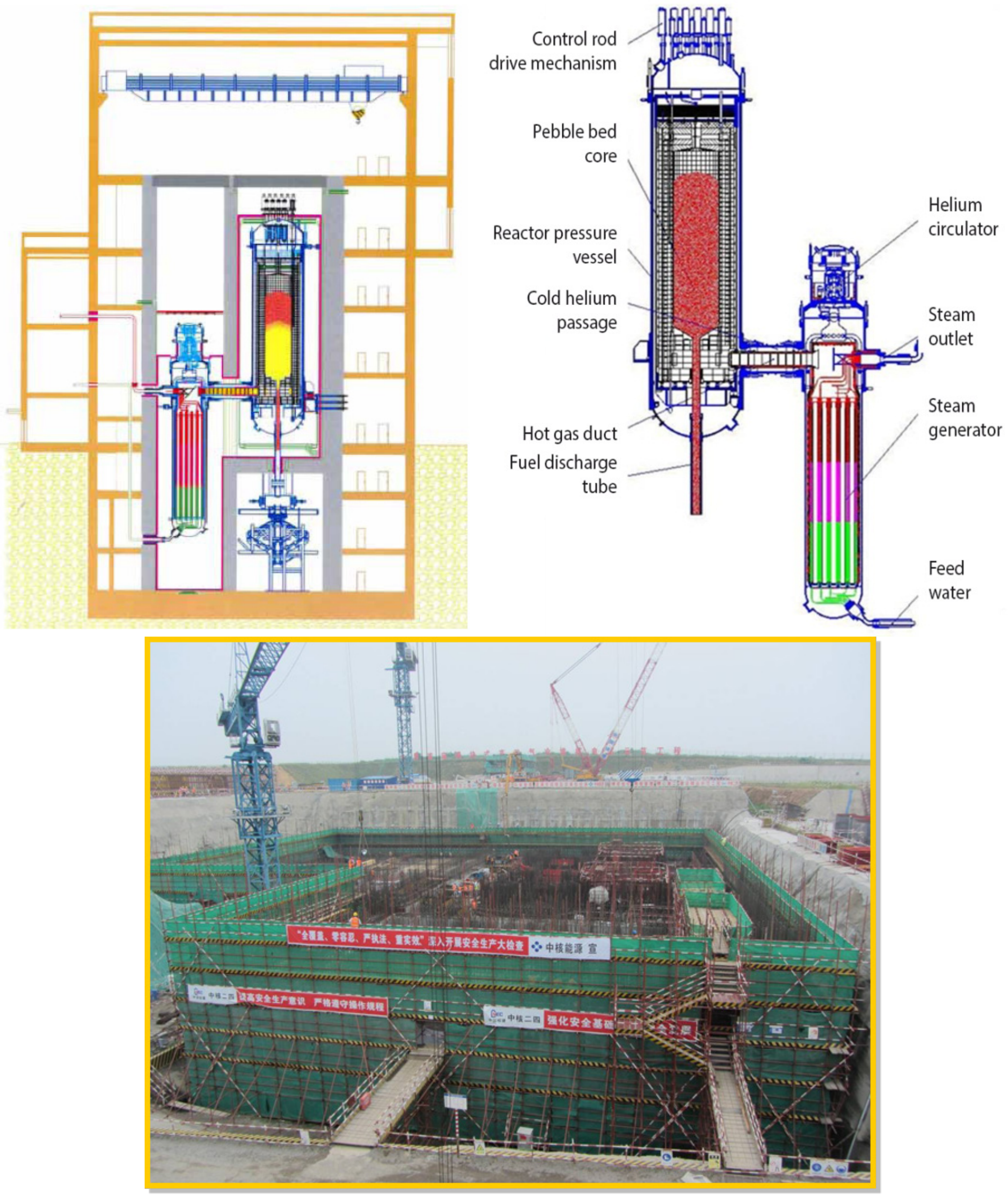
The materials PA, which addresses graphite, metals, ceramics and composites, was signed by implementing agents from Canada, France, Japan, the Republic of Korea, South Africa, Switzerland, the United States and Euratom by 16 September 2009, and is effective since 30 April 2010. China initiated the process for joining the PMB in 2010. South Africa's withdrawal from this PA became effective as of 21 November 2013. Canada withdrew at the end of 2012.

The hydrogen production PA became effective on 19 March 2008 with implementing agents from Canada, France, Japan, The Republic of Korea, the United States and Euratom. In 2010, China expressed its wish to join this PA. As a result, an amended Project Plan incorporating Chinese contributions and other countries' updated contributions was prepared by the PMB and submitted for approval to the System Steering Committee in 2011 October. The further update of the Project Plan is expected in early 2014.

The PA on computational methods, validation and benchmarking remains on hold due to factors discussed below.

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

Figure 3.2: HTR-PM reactor building/primary circuit (top) and photo from the construction site in Shidaowan (bottom)



3.1.2 R&D objectives

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

The VHTR system research plan describes the research and development programme to establish the basic technology of the VHTR system. As such, it is intended to cover the needs of the viability and performance phases of the development plan described in the Generation IV technology roadmap. While the SRP is structured into six projects; only three projects are now effective, and one has been delayed as discussed below:

- Fuel and fuel cycle (FFC) investigations are focusing on the performance of the TRISO coated particles, which are the basic fuel concept for the VHTR. The R&D aims at increasing the understanding of standard design (UO_2 kernels with SiC/PyC coating) and examining the use of uranium-oxicarbide UCO kernels and ZrC coatings for enhanced burnup capability, reduced fission product permeation and increased resistance to core heat-up accidents (above 1 600°C). This work involves fuel characterisation, 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. The R&D also addresses 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.
- Materials (MAT) development and qualification, design codes and standards, as well as manufacturing methodologies, are essential for the VHTR system development. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in the operating environments. For core coolant outlet temperatures up to around 950°C, it is envisioned to use existing materials; however, the goal of 1 000°C, including safe operation under off-normal conditions and involving corrosive process fluids, requires the development and qualification of new materials. Improved multi-scale modelling 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 co-ordination and support modelling to predict damage and lifetime assessment.
- For hydrogen production (HP), two main processes for splitting water were originally considered: the sulfur/iodine thermo-chemical cycle and the high-temperature steam 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 efforts in this PMB address feasibility, optimisation, efficiency and economics evaluation for small and large scale hydrogen production. Performance and optimisation of the 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 also be investigated and design-associated risk analysis will be performed covering potential interactions between nuclear and non-nuclear systems. Thermo-chemical or hybrid cycles are examined in terms of technical and economic feasibility in dedicated or cogeneration hydrogen production modes, aiming to lower operating temperature requirements in order to make them compatible with other Generation IV nuclear reactor systems.
- Computational methods validation and benchmarks (CMVB) in the areas of thermal-hydraulics, thermal-mechanics, core physics, and chemical transport are major activities

needed for the assessment of the reactor performance in normal, upset and accident conditions. Code validation needs to 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.

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

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

Milestones

In the near term, lower-temperature demonstration projects (from 700°C) are being pursued to meet the needs of industries interested in early applications. Future operation at higher temperatures (1 000°C and above) requires development of high temperature alloys, qualification of new graphite type and development of composite ceramic materials.

Lower temperature version of VHTR (from 700°C) will enter the demonstration phase around 2017, with the start of operation of HTR-PM. Higher temperature versions of VHTR (1 000°C and above) will require more research.

The major milestones for the VHTR defined in the *Technology Roadmap Update* are:

- Viability stage/preliminary design and safety analysis: 2010.
- Performance stage/final design and safety analysis: up to 2025.
- Demonstration stage/construction and preliminary testing: from 2025.

3.1.3 Main activities and outcomes

Fuel & fuel cycle (FFC) project

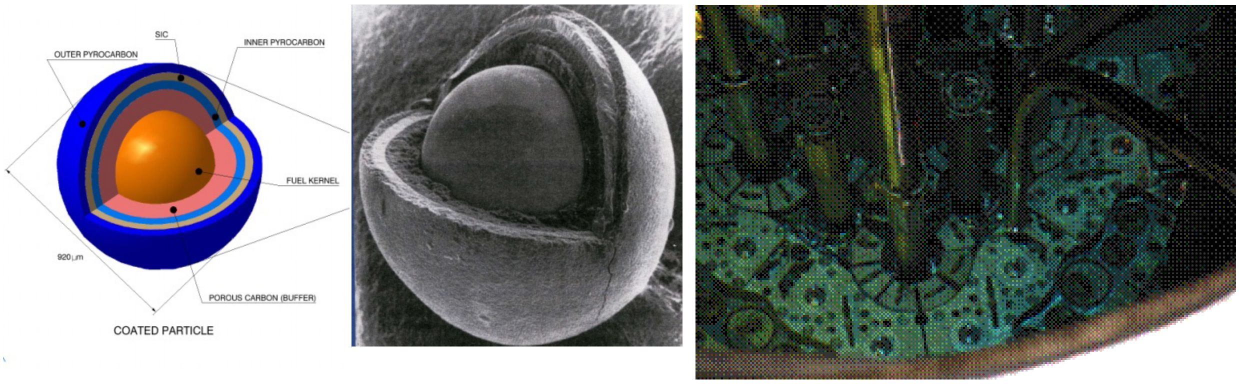
Several irradiation programmes are ongoing.

In the United States, the advanced gas reactor (AGR-2) irradiation that began in June 2010 was completed in October 2013. The capsule contains US UCO and French, South African, and US UO_2. At the time of this writing, the capsule is cooling down and PIE is expected to start in 2014. The project parties have presented general information on these irradiation tests at the High-Temperature Reactor 2012 conference (October 2012 in Tokyo) and will complete them with detailed reports in 2014.

In the United States, PIE of AGR-1 is nearing completion. Most work has been completed, but some final accident safety testing is anticipated for 2014. The PIE of high-flux reactor (HFR) European Union (EU-1) containing Chinese and German fuel irradiated at typical pebble bed conditions is also nearing completion in 2014.

In the Republic of Korea, an irradiation of TRISO fuel began in the High-Flux Advanced Neutron Application Reactor (HANARO) in July 2013. It will continue into 2014.

Figure 3.3: VHTR fuel – TRISO particle and ATR core



The VHTR FCC PMB has begun planning for the third SiC workshop to be held on Jeju Island in South Korea in September 2014. A call for presentations was sent out to the high-temperature reactor fuel research community.

In the EU, the pyro carbon irradiation programme to evaluate creep and swelling/shrinkage (PYCASSO I and II) are performed using surrogate particles fabricated in France, Japan and the Republic of Korea. X-ray tomography of PYCASSO I is underway. Other PIE activities will depend on availability of equipment. Funding for PIE on JAEA and KAERI samples is yet to be found but discussions are underway to define PIE scope.

China has performed extensive characterisation of an oxidised SiC layer on TRISO fuel.

Plans for continuing the successful round robin on fuel characterisation conducted under the IAEA auspices from 2006 to 2010 was discussed among PMB members. The need for a leach burn-leach round robin was discussed and is now part of the next five year programme. A kick-off meeting was held in December 2013. Shipping of natural uranium containing TRISO fuel particles among participants and subsequent analysis will be performed.

The need for an accident benchmark for TRISO fuel performance codes as a follow-up activity to similar work conducted under the IAEA CRP was discussed and the basic framework was established. Data from this PMB will be used to benchmark fuel performance codes. The European Union, the People's Republic of China, the Republic of Korea, the United States, and perhaps Japan are expected to participate. China has continued to develop its fuel performance code.

In the EU, accident safety testing of HFR EU-1 pebbles is planned at 1 700°C and 1 800°C. In the Republic of Korea and China, the conceptual design of accident heating furnaces is underway. A workshop on lessons learnt about the design and operation of such furnaces was held in Idaho Falls (United States) in June 2013.

The AGR 3/4 irradiation started in December 2012. In this experiment, 12 separate capsules containing designed-to-fail fuels are being irradiated over a spectrum of burnup, temperature, and fast fluence parameters to understand fission product release from failed fuel and retention of fission products in fuel matrix and fuel element graphite. Particle failures occurred as planned within two weeks after the experiment began, and data on fission product release have being gathered. Fission gas release is being correlated to temperature and half-life parameters.

Both the Republic of Korea and Japan are performing out-of-pile oxidation experiments with several graphite materials and SiC TRISO coated (surrogate) fuel particles under air ingress accident conditions for high-temperature gas-cooled reactor. Oxidation rate studies have been performed in the Republic of Korea on fuel matrix material in 2013 and surrogate TRISO fuel compact tests are planned for 2014. China has focused on the study of the effect of SiC grain size on the oxidation behaviour of SiC.

In Europe, experiments are underway to study dust transport and resuspension in two experiments (TUBE and TANK) at the University of Dresden. In addition, air and moisture ingress effects on graphite are being studied in the *Naturzug im Core mit Korrosion* (NACOK) facility.

In the area of advanced fuels, both the Republic of Korea and China are continuing to develop production routes for UCO, based in large part on the successful performance of this advanced high burnup fuel in the AGR-1 experiment. In 2013, the Republic of Korea has focused on different methods of carbon dispersal. China is interested in developing UCO ZrC TRISO and has been evaluating ZrC coating layers.

The area of waste management and other fuel cycle options covers three issues:

- spent VHTR fuel management;
- irradiated graphite management;
- transmutation using a VHTR.

In the EU, the final Carbonaceous Waste (CARBOWASTE project) workshop was held in March 2013, and the project was successfully completed in May 2013 with a possible new project planned for 2014. Opening of results to Generation IV International Forum is foreseen. In the European project ARCHER, three tasks are ongoing:

- Corrosion of coatings under waste disposal conditions;
- Model development for long term performance of TRISO coated particle fuel;
- Safety case for waste management.

Some documents concerning fuel storage in the framework of ARCHER will be made available to the VHTR FFC PMB in 2014. No activity has yet started regarding the assessment of the VHTR thorium fuel cycle.

Materials

Although the original Materials PP extended until 2012, the Materials PMB continued its work in 2013 while simultaneously pursuing an explicit extension of the PP through 2015. Changes in participation of the PMB to reflect the new expected signatories of the PA were necessary prior to establishing a new PP. Canada withdrew unconditionally from the PA, effective 31 December 2012, at its own request, reflecting changes in its internal programmatic priorities. The conditional withdrawal agreement for Pebble Bed Modular Reactor LTD (PBMR) from the PA became effective on 21 November 2013. Contributions for the extension of the PP through 2015 were developed by the remaining six Signatories (European Union, France, Japan, Republic of Korea, Switzerland, and United States), as well as China that will be joining the PA. The extended contributions are being compiled into a revised PP that will be reviewed by the PMB for recommended approval by the VHTR System Steering Committee, and subsequent signature.

By the end of the year, approximately 300 technical reports describing contributions from all signatories had been uploaded into the Gen IV Materials Handbook, the database used to share materials information within the PMB. This was well over twice as many reports as originally scheduled within the PA, reflecting the outstanding technical output of the membership. Uploads of the supporting materials test data are proceeding well for metals, but have been slower for graphite, for which international standardisation of testing is less developed (e.g. ASTM, ISO, etc.). The PMB's Graphite Working Group has been working with the Handbook Manager to resolve technical ambiguities and develop a viable common data upload template with the result that graphite data uploads have now begun on an evolutionary basis.

Technical liaison between the PMB and the ASME Code continues and is helping to ensure an actionable understanding of international codification needs for high-temperature gas-cooled reactors. In 2013, the Code's efforts to extend allowable stresses for alloy 800H to the higher temperatures and longer times, the implementation of chemistry restrictions on allowable stresses for certain stainless steels, and the establishment of an improved definition of creep-fatigue behaviour were examples of specific items needed for HTGRs. An area of common active interest and co-operation between the PMB and the ASME is the development of the technology and related design and construction rules for compact heat exchangers. The inclusion of rules on the use of graphite within ASME's new Division 5 of Section III on High Temperature Reactors reflects another critical intersection of the PMB's and ASME's technical interests.

In 2013, research activities continued to focus on near- and medium-term projects needs, i.e. graphite and high-temperature metallic alloys. Characterisation of selected baseline data and its inherent scatter of candidate grades of graphite were performed by the different members. Graphite irradiations continued to provide data on properties changes and irradiation creep behaviour, while related work on oxidation examined both short-term and chronic exposure effects on graphite. Examination of environmental and inelastic high temperature alloys, (800H and 617) provided very useful information for their use in heat exchanger and steam generator applications. Irradiation effects on advanced pressure vessel steels (9Cr and oxide dispersion strengthed steels) was continued. In the near/medium term VHTR projects, targeting temperature below about 850°C, metallic alloys are considered as the main option for control rods, instead of ceramic composites which are intended for future projects at temperatures up to and above 1 000°C. Ceramics are also still of interest as thermal insulation materials and for gas fast reactor fuel cladding and limited work continued to develop testing standards and examine irradiation effects and fabrication methods on ceramic composites.

Hydrogen production

The hydrogen production project arrangement was originally signed by Canada, France, Japan, the Republic of Korea, the United States and Euratom. China has been a candidate for joining the PA since 2010, expressing interest especially in the sulfur-iodine thermochemical cycle, and the high temperature steam electrolysis.

The main activities overseen by the PMB deal with thermochemical cycles (SI and copper-chlorine), and with HTSE. Studies on the SI process are mainly driven by the Japanese (HTTR-IS) and Korean (NHDD) programmes. Concerning the SI process, the Republic of Korea installed in 2012 a 3-section module integration test facility for the production of 50 NL·H2/h. Performance tests of each module are being carried out by POSCO, KIST and KIER. The integrated tests have been carried out at the operation pressure of 5 bar in 2013. Japan concentrated its efforts on the integrity tests of IS process key components made of industrial materials. A test facility on the Bunsen section of 150 NL·H2/h equivalent was completed and flow tests of the process solution are underway to examine the integrity of corrosion-resistant lining materials. China also fabricated SI components to erect an integrated test facility of 100 NL·H2/h.

Regarding the development of HTSE, the United States demonstrated pressurised operation of a 10-cell advanced technology stack at 1.5 MPa. HTSE at 4kW scale was demonstrated for 1 000 hours with an advanced technology. A public report *System Evaluation and Life-Cycle Cost Analysis for HTSE Hydrogen Production Facilities* was published in April 2012. With NGNP funding ending, additional near-term progress on this technology are in the hands of industry and non-US research organisations. Canada has conducted HTSE studies since 2007, in the field of modelling of HTSE plant, integration of HTSE plant with nuclear reactors and economic analysis of hydrogen production processes. It is expanding into studies of solid oxide electrolysis cells, including hydrogen production, synthesis gas production, experimental test systems and CFD modelling of SOEC. China also has a HTSE development programme for the improvement of SOEC stacks, including a lab-scale HTSE system design.

The third most studied process in the Canadian programme is the hybrid copper-chlorine (Cu-Cl) thermochemical cycle, which could be operated at lower temperatures, in agreement with the temperature targeted for other Generation IV systems such as the SCWR. Unit operation of a lab-scale Cu-Cl cycle was successfully performed in 2011 and a conceptual process flow diagram for a 4-step Cu-Cl cycle with solid feed and a crystalliser has been developed. It can be operated at high temperature and high pressure. The future vision on the integrated Cu-Cl plant will be finally expanded to the large commercial plant of 1 000 ton H_2/day through two step intermediate scale tests of 500 kg H_2/day and 50 ton H_2/day, such as demonstration of electrolyser for CuCl/HCl and thermal decomposition of copper-oxychloride. Future experimental plans for the improvement of component performance and the integrated operation test of the Cu-Cl cycle have been established.

Computational methods validation and benchmark

The list of provisional signatories to the computational methods validation and benchmark (CMVB) PA evolved in 2013, reflecting the national programmes of participants in the VHTR System Agreement. Provisional members are now China, Euratom, the Republic of Korea, Japan and the United States. While much work was done over several years to develop a draft project plan, the withdrawal of South Africa from VHTR R&D activities left a large leadership void for several key CMVB activities. The project plan is currently on hold until these issues can be resolved.

HTR-2012 conference

Finally, the 6th International Topical Meeting on High Temperature Reactor Technology HTR-2012, Nuclear Energy for the Future was held in Japan in 2012 (28 October-1 November). The conference was successful, with over 120 papers presented. The next conference, HTR-2014 will be held in Weihai, Shandong Province, China, in October 2014.

Publications from the HTR-2012 Conference

Bae Y., S. Hong, Y. Kim, “Scaling Analysis of Reactor Cavity Cooling System for PMR 200”.

Chang J. et al., “A Dynamic Simulation Code for the Sulfur-Iodine Process Coupled to a Very High Temperature Gas-Cooled Nuclear Reactor”.

Fütterer M. A. et al., “The Potential Impact of HTR and Nuclear Cogeneration in the European Energy Policy Context”.

Hamamoto S. et al., “Chemical Characteristics of Helium Coolant of HTR (High Temperature engineering Test Reactor)”.

Hania P.R. et al., “Post-irradiation Examination of ZrC based TRISO Coatings from the PYCASSO Experiments”.

Harp J.M., P. A. Demkowicz, S. A. Ploger, “Post-irradiation Examination and Fission Product Inventory Analysis of AGR-1 Irradiation Capsules”.

Kim W., G. Lee, S. Hong, J. Park, “Temperature Effect on Creep behaviour in Air and Helium Environments of Alloy 617”.

Kubo S. et al., “R&D Progresses on Thermochemical Water-splitting Iodine-sulfur Process at JAEA”.

Li F., D. Wang, C. Hao, Y. Zheng, “Solution of Multiple Circuits of Steam Cycle HTGR Systems”.

Park J., H. Kim, S. Hong, Y, Kim, “Materials Development for Process Heat Exchange (PHE) in Nuclear Hydrogen Production System”.

Petti D. et al., “Compacting Scale Up and Optimization of Cylindrical Fuel Compacts for the Next Generation Nuclear Plant”.

Terada A. et al., “Conceptual Design of Hydrogen Supply System for HTGR Hydrogen Production System”.

Ueta S. et al., “R&D Plan for Development of Oxidation-Resistant Graphite and Investigation of Oxidation Behavior of SiC Coated Fuel Particle to Enhance Safety of HTGR”.

Wang Y., Y. Zheng, F. Li, L. Shi, "Analysis of Blown-down Transient in Water Ingress Accident of High Temperature Gas-Cooled Reactor".

Zheng Y., F. Chen, L. Shi, "Analysis of Diffusion Process and Influence Factors in the Air Ingress Accident of the HTR-PM".

Zhou Y. et al., "Thermal Hydraulic Analysis for Hot Gas Mixing Structure of HTR-PM".

3.2 Sodium-cooled fast reactor (SFR)

3.2.1 Main characteristics of the system

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

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

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

Much of the basic technology for the SFR has been established in former fast reactor programmes, and is being confirmed by the Phenix end-of-life tests in France, the restart of Monju in Japan, the lifetime extension of BN-600 and, and the start-up of the China Experimental Fast Reactor.

- The SFR is an attractive energy source for nations that desire to make the best use of limited nuclear fuel resources and manage nuclear waste by closing the fuel cycle. Fast reactors hold a unique role in the actinide management mission because they operate with high energy neutrons that are more effective at fissioning transuranic actinides. The main characteristics of the SFR for actinide management mission are: Consumption of transuranics in a closed fuel cycle, thus reducing the radiotoxicity and heat load which facilitates waste disposal and geologic isolation.
- Enhanced utilisation of uranium resources through efficient management of fissile materials and multi-recycle.

High level of safety achieved through inherent and passive means also allows accommodation of transients and bounding events with significant safety margins.

The reactor unit can be arranged in a pool layout or a compact loop layout. Three options are considered in the GIF SFR System Research Plan:

- A large size (600 to 1 500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially minor actinides, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors as shown in Figure 3.4.
- An intermediate-to-large size (300 to 1 500 MWe) pool-type reactor with oxide or metal fuel as shown in Figure 3.5 and Figure 3.6.

- A small size (50 to 150 MWe) modular-type reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor as shown in Figure 3.7.

The two primary fuel recycle technology options are (1) advanced aqueous and (2) pyrometallurgical processing. A variety of fuel options are being considered for the SFR, with mixed oxide the lead candidate for advanced aqueous recycle and mixed metal alloy the lead candidate for pyrometallurgical processing.

Status of co-operation

The system arrangement (SA) for the international research and development of the SFR nuclear energy system became effective in 2006 and the present signatories of the SA are recalled in table 1.2.

Three project arrangements were signed in 2007: Advanced Fuel (AF), Component Design and Balance-of-Plant (CD&BOP), and Global Actinide Cycle International Demonstration (GACID). The latter was extended for two years in 2012. A further extension is under discussion. The updated Project Arrangement of CDBOP including the contribution of a new member, Euratom, has been drafted and signature is expected in 2014. The Project Arrangement for Safety and Operation (SO) was signed in 2009 and amended in 2012 to include the contributions of Euratom, the People's Republic of China and the Russian Federation. The Project Arrangement for System Integration and Arrangement (SIA) is in the process of being signed.

Figure 3.4: JSFR (loop-configuration SFR)

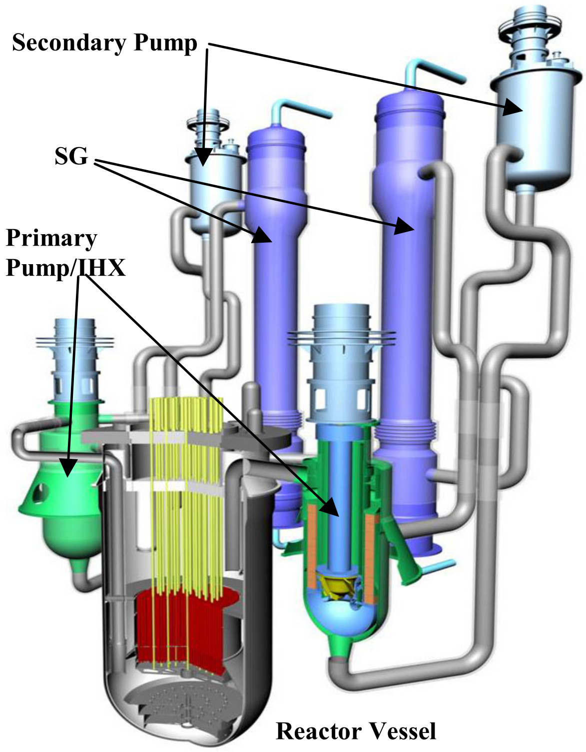


Figure 3.5: ESFR (pool-configuration SFR)

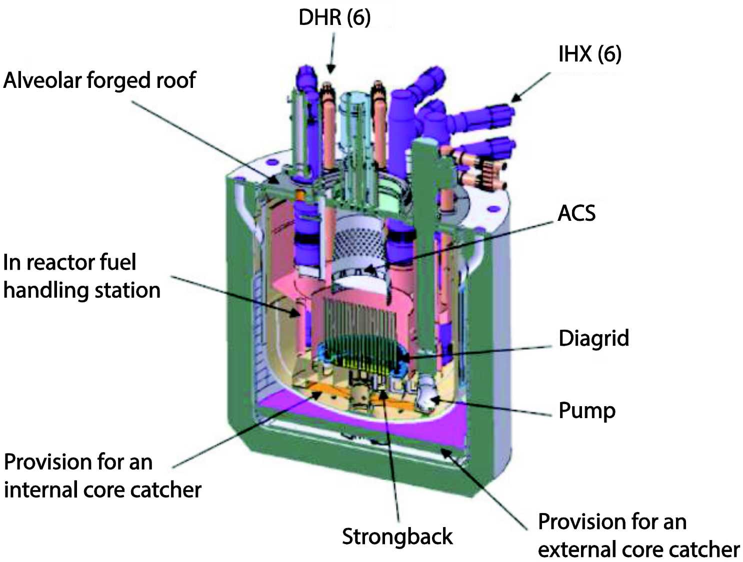


Figure 3.6: KALIMER (pool-configuration SFR)

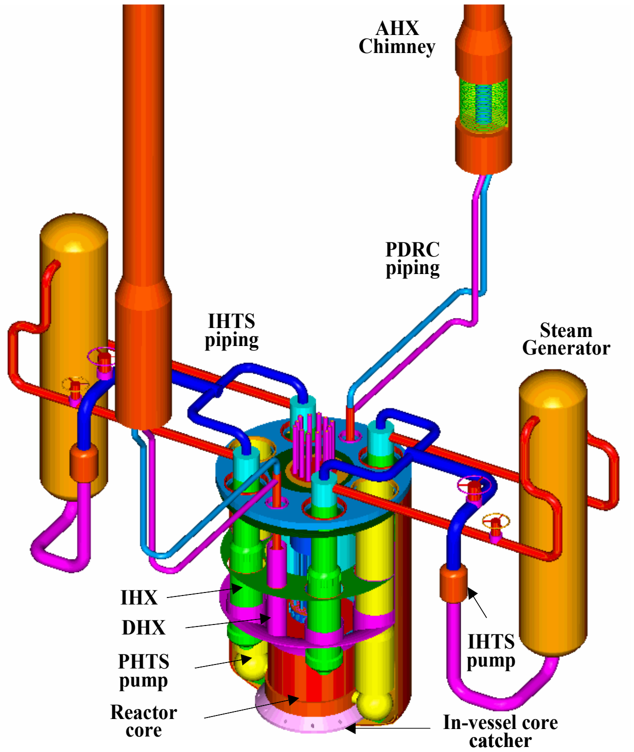
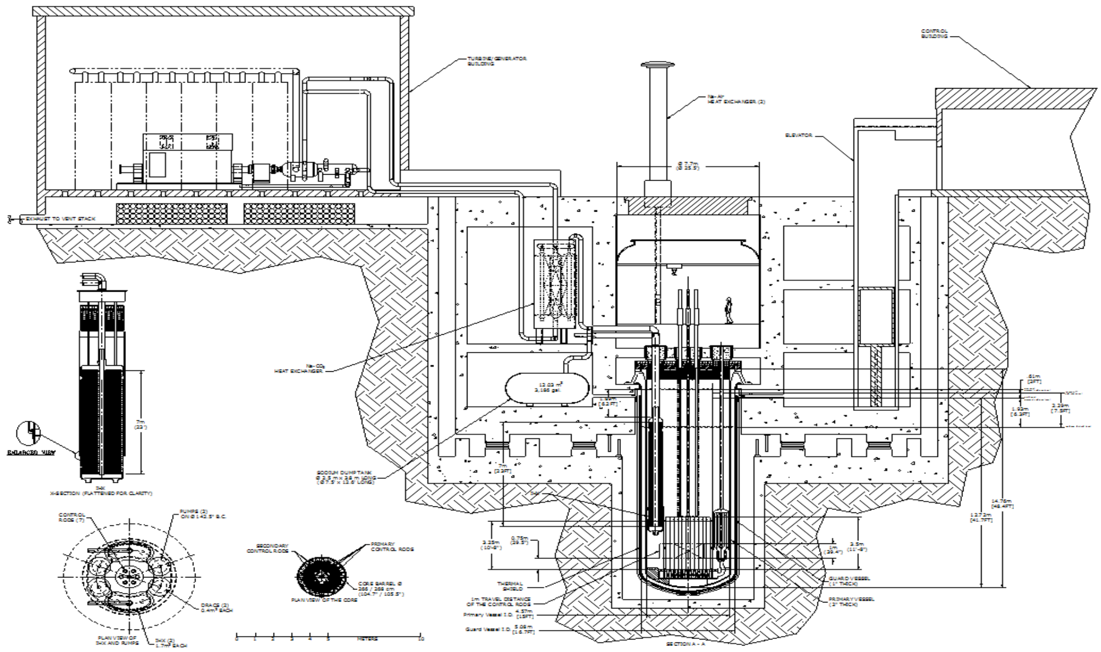


Figure 3.7: SMFR (small modular SFR configuration)



3.2.2 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 summarised below.

System integration and assessment project (SIA)

The main objectives of system integration and assessment are: to maintain and refine system options, reflecting continuous trade-off studies and technology development; to identify R&D needs and ensure 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 project (SO)

The SO project is arranged into three Work Packages (WPs) which consist of WP SO 1 “Methods, models and codes” for safety technology and evaluation, WP SO 2 “Experimental programmes and operational experience” including the operation, maintenance and testing experience in the experimental facilities and existing SFRs (e.g. Monju, Phenix, BN-600 and CEFR), and WP SO 3 “Studies of innovative design and safety systems” related to the safety technology for the Gen IV reactors such as passive safety systems.

Advanced fuel project (AF)

Fuel related research aims at developing high burnup MA bearing fuels as well as claddings and wrappers withstanding high neutron doses and temperatures. It includes: research on remote fuel fabrication techniques for fuels that contain minor actinides and possibly traces of fission products as well as performances under irradiation of fuels, claddings and wrappers. Candidates under consideration are: oxide, metal, nitride and carbide for fuels, alternate fast reactor fuel forms and targets for special applications (e.g. high temperature), and Ferritic/Martensitic & ODS steels for core materials.

Component design and balance-of-plant project (CD&BOP)

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, steam generators and development of alternative energy conversion systems, e.g. using Brayton cycles. Such a system, if shown to be viable, would reduce the cost electricity generation significantly. The primary R&D activities related to the development of advanced BOP systems are intended to improve the capital and operating costs of an advanced SFR. The main activities in energy conversion system include: (1) development of advanced, high reliability steam generators and related instrumentation; and (2) the development of advanced energy conversion systems based on Brayton cycles with supercritical carbon dioxide as the working fluid. In addition, the significance of the experience that has been gained from SFR operation and upgrading is recognised.

Global actinide cycle international demonstration project (GACID)

The GACID project aims at conducting collaborative R&D activities with a view to demonstrate, at a significant scale, that fast neutron reactors can indeed manage the actinide inventory to satisfy the Generation IV criteria of safety, economy, sustainability and proliferation resistance and physical protection. The project consists of MA bearing test fuel fabrication, material properties measurements, irradiation behaviour modelling, irradiations in Joyo, licensing and pin scale irradiations in Monju, and post-irradiation examinations, as well as transportation of MA raw materials and MA bearing test fuels.

Milestones

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

- SIA project:
  - Definition of SFR system options.
    - **2011**: initial specification of SFR system options and design tracks.
  - Definition of SFR R&D needs.
    - **2009:** review and refine SFR R&D needs in the SRP.
  - Review of assessments of SFR design tracks.
    - **2012:** Compile existing self-assessment results for SFR design tracks.
    - **2012:** Solicit economics assessment using ESWG methodology.
    - **2013:** Solicit proliferation assessment using PRPP methodology.
    - **2014:** Solicit safety assessment using RSWG methodology.
- SO project:
  - Methods, models and codes.
    - **2008-2011:** Research collaboration on methods, models and codes for safety technology and evaluation among four countries of France, Japan, Republic of Korea and United States.
    - **2012:** Research collaboration between China, France, Japan, Republic of Korea, Russia, United States and Euratom.
  - Experimental programmes and operational experience.
    - **2008-2011:** Research collaboration on the experimental programmes and operational experience including the operation, maintenance and testing experience in the existing SFRs (e.g. Monju, Phenix, BN-600 and CEFR) between

France, Japan, Republic of Korea and United States. (Collaboration with Korea started in 2009).

  - **2012:** Research collaboration between China, France, Japan, Republic of Korea, Russia, United States and Euratom.

- Studies of innovative design and safety systems.

  - **2008-2011:** Research collaboration on the studies of innovative design and safety systems related to the safety technology for the Gen IV reactors such as passive safety system among France, Japan, Republic of Korea and United States.
  - **2012:** Research collaboration between Euratom, China, France, Japan, Republic of Korea and United States**.**

- AF Project:
  - **2006-2015:** Preliminary evaluation of advanced fuels.
  - **2008-2015:** Evaluation of MA-bearing fuels.
  - **2008-2020:** High-burnup fuel behaviour evaluation.
  - **2021:** Demonstration and application of the selected advanced fuel.

- CD&BOP Project:
  - **2007-2012:** Viability study of proposed concepts.
  - **2009-2015:** Performance tests for detailed design specification.
  - **2014-2016:** Demonstration of system performance.

- GACID Project:
  - **2007-2013:** Preparation for the limited MA-bearing fuel irradiation test.
  - **2007-2013:** Preparation for the licensing of the pin-scale curium-bearing fuel irradiation test.
  - **2007-2013:** Programme planning of the bundle-scale MA-bearing fuel irradiation demonstration.
  - **2014-2018:** Under discussion among the participants to achieve the objectives of the project.

3.2.3 Main activities and outcomes

Safety and operation project

Work Packages (WPs) of the SO project were rearranged in 2012 into three WPs which consist of WP SO 1 “Methods, models and codes”, WP SO 2 “Experimental programmes and operational experiences” and WP SO 3 “Studies of innovative design and safety systems”. The major developments in these three areas in the 2012 Annual Work Plan have been summarised as follows:

WP SO 1 – Methods, models and codes

Concerning core safety, unprotected loss-of-flow and overpower transient analyses in CEFR using SAS4A/SASSYS-1 code were evaluated. Outcomes of the analyses will allow deeper understanding on CEFR core behaviour under severe postulated transient circumstances. Basic experiments for model development were conducted on molten fuel discharge behaviour in order to clarify the mechanism of upward fuel discharge in the FADUS system in JSFR (Figure 3.8). As a result, it is concluded that coolant vapour can behave as a driving force for the upward discharge in the inner duct structure. A PIRT study was performed on the core evolution and in-vessel structures behaviour in transients of the design basis domain and on core degradation phenomenology. The main phenomena occurring during a set of accident scenarios under both design-basis and design-extension conditions were identified and ranked from a safety point view. Further, a study was performed on safety goals and general safety principles for future reactors including a synthesis of safety requirements applicable to the ESFR. As one of the core safety evaluation method, an advanced code development was initiated for joint multidimensional calculation of neutronic-physical and thermal-hydraulic processes in the SFR core under transient and accidental conditions.

For the safety system analysis method on natural convection, coupled system and CFD codes (CATHARE x TRIO_U) were used for the Phenix natural convection test (Figure 3.9). These results validate some important assumptions of the present methodology (low order convection schemes, implicit time marching, use of wall functions, single-phase and non-compressible models).

PSA studies are carried out in support of the design of the ASTRID prototype of French Gen IV Sodium Fast Reactors. First modelling options have been assessed and preliminary fault trees and event trees have been built for ASTRID. Level-1 PSA method developments for JSFR were also performed considering the passive safety reliability including the phenomena identification and ranking table (PIRT) method. To support level-2 PSA analysis, a code was developed to evaluate the released energy and its mechanical loading on surrounding walls during HCDA (hypothetical core disruptive accident) of KALIMER-150.

A simulator platform was developed for visualisation and demonstration of innovative design concepts and safety features. It is used to interactively run transients and display the plant response to demonstrate the inherently safe response of a liquid-metal reactor to upset events.

WP SO 2 – Experimental programmes and operational experience

New sodium experimental facility for safety evaluation: the construction of an experimental facility was completed for the study and evaluation of performance of a passive decay heat removal circuit. Major components were manufactured and installed to compose the STELLA-1 facility in KAERI, and experimental works were scheduled to be conducted. The CFD analysis assessing the thermal-hydraulic performance of the major heat exchangers, e.g. DHX (sodium-to-sodium decay heat exchanger) and AHX (sodium-to-air heat exchanger) were performed (Figure 3.10).

Existing SFR: the measurement and analysis of sodium void reactivity of CEFR was performed, and the measurement method, experiment data and analysis results were described. The results of the analysis will be compared with measurement data, which will demonstrate the core design with large negative sodium void activity in the whole core.

Existing SFR data: the analysis of most recent metal fuel transient pin disruptive tests at TREAT was performed using the SAS4A code to gain insight into metal fuel behaviour during fast reactor transients for estimates of margins to cladding failure and accident progression. The results of SAS4A analyses for thermal-mechanical response of the metal-alloy fuel elements have been compared with data from TREAT Tests M5-7.

Maintenance technology development for existing SFR: improvement of numerical simulation code has been conducted for eddy current testing of steam generator tubes in Monju. And also, improvement of corrosion hazard evaluation code “PSYHCE” was performed.

WP SO 3 – Studies of innovative design and safety systems

The development of a small, modular sodium-cooled fast reactor concept, AFR-100, continued to identify the key R&D needs and challenges, pursue utilisation of advanced technology options, and confirm feasibility of innovative features (Figure 3.11). To demonstrate the safety response of AFR-100, the SAS4A/SASSYS-1 code has been used to simulate unprotected (without scram) loss of heat sink (ULOHS) and unprotected total loss of power (ULOF) scenarios for different pump coast-down assumptions.

The safety architecture was assessed to master the safety functions and design measures for consequence mitigation of seismic loads. Guidelines and recommendations of techniques and methods for the reduction of seismic vulnerability and evaluation of the consequences of mitigation are described. Further, a study was performed to provide guidance for ESFR safety studies with respect to design strategy and approach with respect to confinement function.

Figure 3.8: Molten fuel discharge experiment

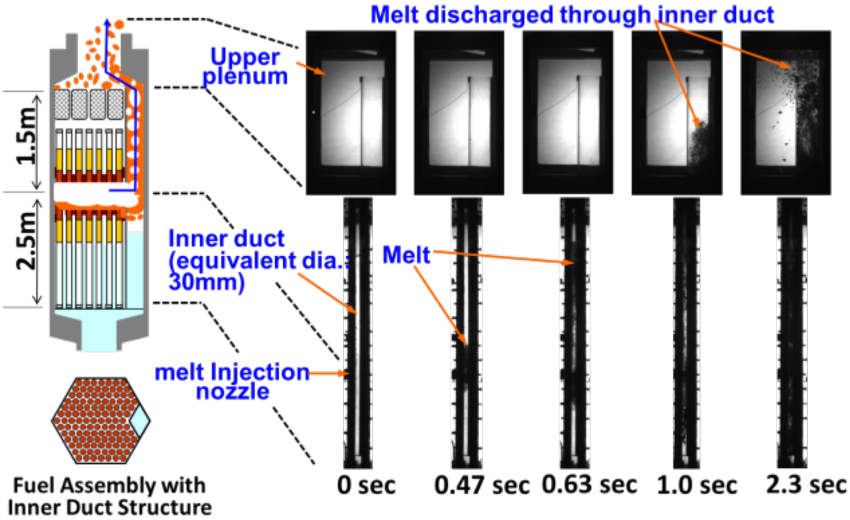


Figure 3.9: CFD for the natural convection test

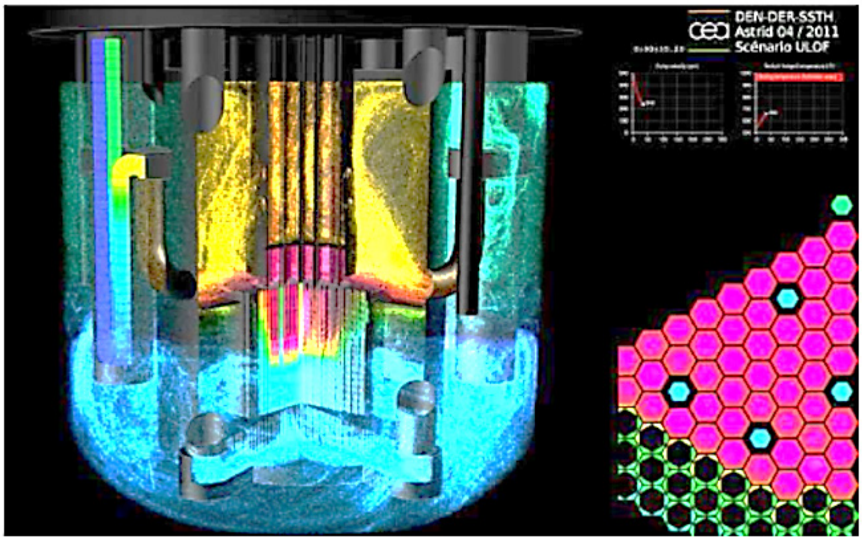


Figure 3.10: STELLA-1 facility

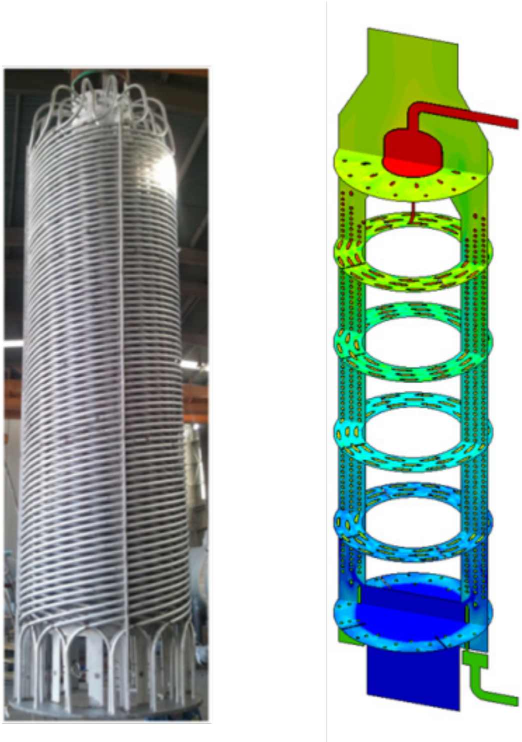
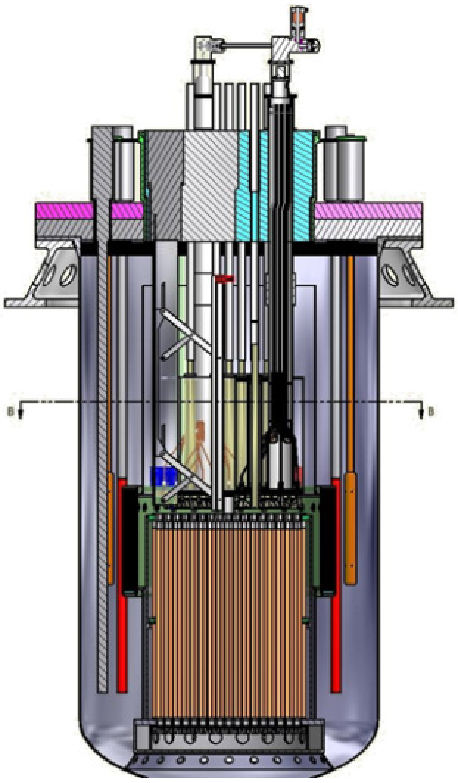


Figure 3.11: AFR-100 reactor concept

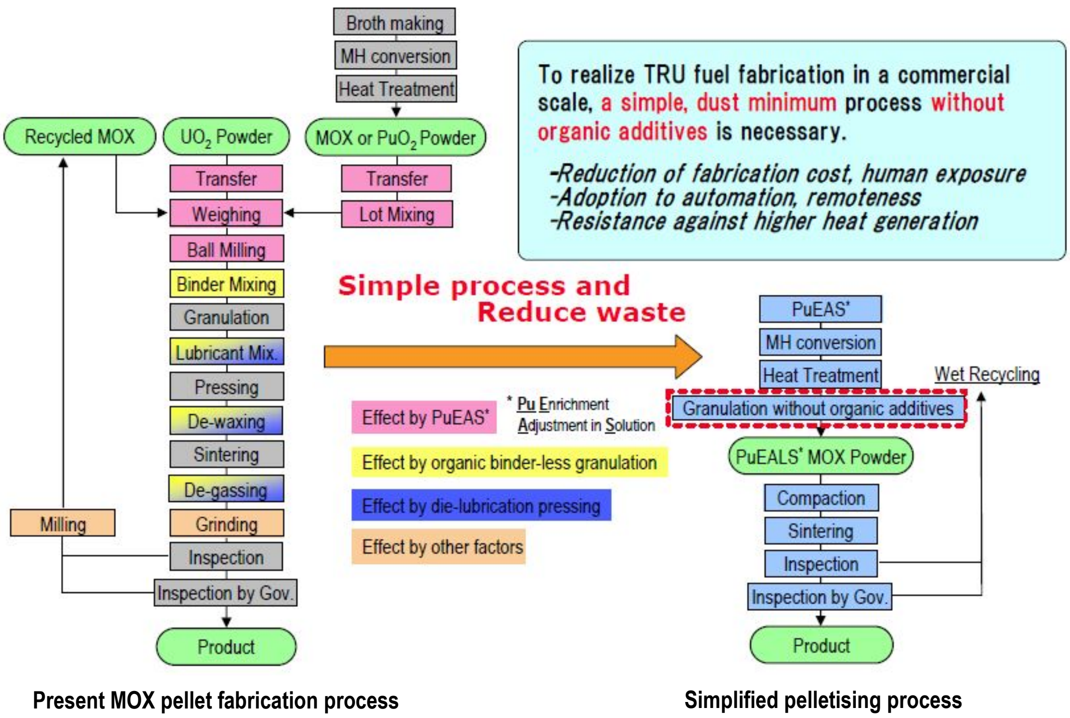


Advanced Fuel Project

A first technical evaluation based on historical experience, knowledge of fast reactor 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 quickly meet the goals. Regarding core materials, promising candidates are Ferritic/Martensitic and ODS steels. Fuel investigations have been enlarged since 2009 to include the heterogeneous route for MA transmutation, for which MA are concentrated in dedicated fuels located at the core periphery, as identified in the SIA project.

In 2013, irradiation test preparation and implementation as well as post-irradiation examinations (PIE) have been continued regarding MA bearing oxide, metal, nitride and carbide fuels. In particular, PIE for Am bearing oxide fuel irradiated in Joyo and PIE for MA bearing oxide and metal fuels irradiated in ATR have been performed. Preparation work continued for irradiation tests. Oxygen potential for PuO2 was evaluated and sintering parameters and mechanisms for (Am,U)O2-x have been investigated. The melting point of UN and (U, Pu)N was determined. Processes for MA bearing fuel fabrication and the granulation technology for simplified pelletising method in hot cell by remote operation were developed (see Figure 3.12).

Figure 3.12: Schematic flow of simplified pelletising method (top) and HT9 cladding tube (bottom)



Regarding cladding development, fabrication and characterisation of Ferritic/Martensitic cladding tubes were continued. The effects of heat treatment on the properties of HT9 cladding tube have been investigated. Preparation of fuel pin with ODS cladding for irradiation in Joyo was continued.

Component design and balance-of-plant project

The CD&BOP project started in October 2007 when the Project Arrangement was signed by France, Japan, the Republic of Korea and the United States. Euratom joined the CD&BOP PA in 2012. The CD&BOP activities include in-service inspection and repair technologies, LBB assessment technology and sodium heated steam generators. Supercritical CO_2 Brayton cycle is also addressed as an advanced energy conversion system which is an alternative to the conventional steam Rankin cycle system.

Two methods of "under sodium viewing" (USV) have been investigated in the inspection technology study:

- Ultrasonic viewing technique using phase array probes in sodium. Heat-resistance ultrasonic transducers such as EMAT and piezo-element were examined in high temperature conditions. The USV techniques were tested in water test first with the CIVA code simulation, and then in-sodium test equipment was prepared for the next step.
- 10m long multi-array ultrasonic wave-guide sensor module for various applications (see Figure 3.13). The inspection sensor was fabricated and the differential signal process method was introduced to improve its S/N ratio. A new sodium test facility was also set up to investigate the wetting performance of the sensor.

In parallel with R&D on inspection sensors, a possible improvement of RCC-MRx (Design and Construction Rules for Mechanical Components of Nuclear Installations) was discussed to adjust to the needs required by innovative sodium fast reactors' design options.

The remote sodium removal and welding using Laser technique has been studied since 2012 as an advanced repair technology.

Fatigue crack growth (FCG) tests were conducted for Gr.91 specimens to obtain its mathematical model as part of the LBB assignment technology study. Creep crack growth (CCG) tests on base metal, welding area, and HAZ were also conducted to be compared.

The steam generator (SG) study in 2013 covered the water flow instability computer analysis, main parts trial fabrication of a double-walled-tube (DWT) SG, tube inspection technique development and heat transfer performance test equipment for double-walled tubes.

In the water flow instability analysis, the computer code was validated by using the existing 1MWth DWT-SG (see Figure 3.14) experimental data, and the code clarified the characteristics on the water flow instability of a large-sized DWT-SG.

Essential parts such as double-walled tubes, tube-sheet and tube-to-tube-sheet junction were fabricated to confirm their manufacturing process and required performances.

Heat transfer tubes need to be inspected in order to prevent tube failures leading to sodium/water reactions. The performance of the remote field eddy current testing (RF-ECT) technique has been developed for ferromagnetic G91 single-walled tubes. Magnetic sensor testing is considered as a back-up technique for the RF-ECT.

The thermal resistance at the interface of DWTs is a fundamental property to evaluate the heat transfer performance of DWT-SGs. Sodium-to- sodium heat exchanging test equipment for DWTs was constructed in 2013 to measure their thermal resistance at the interface.

Because sodium/water reaction is one of the significant issues of sodium-heated SGs, high reliable SGs such as DWT-SG have been developed to reduce the possibility of tube failure. In addition to this approach, CD&BOP has dealt with S-CO_2 Brayton cycle systems as well to exclude

Na/water reaction and to obtain further economy. In 2013, CFD analyses of flow in a S-CO_2 compressor (see Figure 3.15) and Control/Dynamics analysis on an SFR with S-CO_2 cycle (see Figure 3.16) were performed to obtain thermo-dynamical performance data in various conditions and to develop the active control strategy and the autonomous SFR control. A sodium-CO_2 heat exchanger is such a key component of SFRs with S-CO_2 system that several tests have been planned and performed. For example, sodium freezing/melting and drain/fill experiment revealed the fundamental flow phenomena in small paths of heat exchangers. Na/CO_2 reaction test facilities were also constructed to understand the reaction and to confirm that the reaction has no significant effect on the system's safety. Use of all the data provided by the CD&BOP members clarified material corrosion mechanism in S-CO_2 environment and allowed corrosion development prediction.

Figure 3.13: Multi-array wave guide sensor module

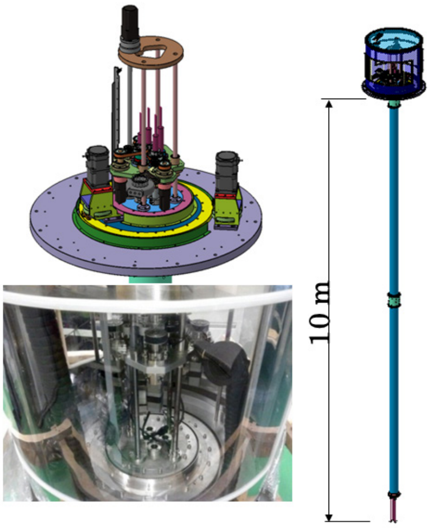


Figure 3.14: 1MW$_{th}$ Double-walled-tube SG test model

Figure 3.15: S-CO_2 compressor CFD analysis

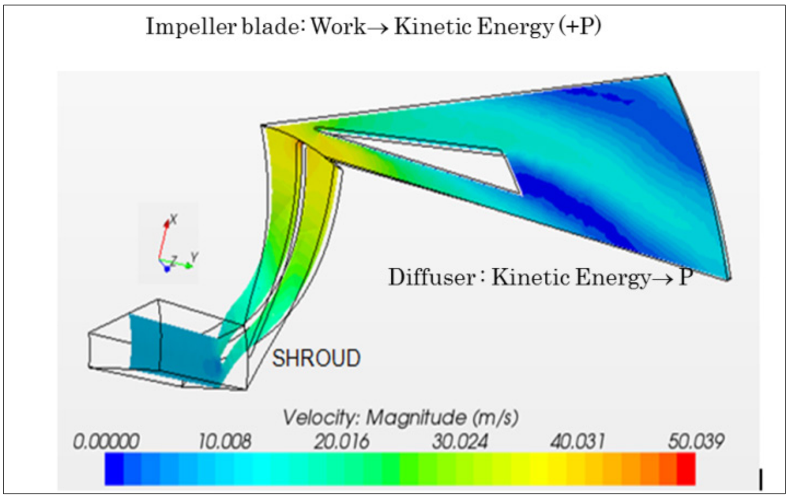
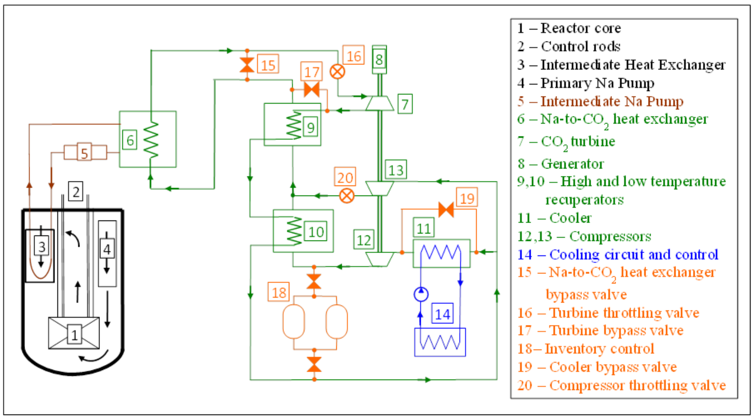


Figure 3.16: Plant dynamics code model on SFR with S-CO_2



Global actinide cycle international demonstration project

The Global actinide cycle international demonstration project 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 behaviour modelling and post-irradiation examinations (PIEs), as well as transportation of MA raw materials and MA-bearing fuels. Bundle-scale demonstration will be included.

The irradiation behaviour of the Am-1 test in the Joyo reactor, such as americium migration, was analysed and investigated in detail based on the PIE results for irradiation behaviour modelling. The Joyo irradiation experiment is currently suspended. According to JAEA, the irradiation experiment will resume after completion repairs.

The irradiation of AFC-2C and 2D has been performed by DOE in the ATR material testing reactor in Idaho. Preliminary irradiated fuel characterisations have been realised and presented to the GACID members.

R&D on fabrication is in progress and the specifications of (U, Pu, Am, Np)O_X, have been established at CEA. The overall programme on property measurements was defined and split between several laboratories.

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3.3 Supercritical-water-cooled reactor (SCWR)

3.3.1 Main characteristics of the system

The Supercritical Water-Cooled 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 grouped into two main categories: pressure vessel concepts proposed first by Japan and more recently by a Euratom partnership, and pressure tube concepts proposed by Canada, generically called the Canadian-SCWR. Other than the specifics of the core design, these concepts have many similar features (e.g. outlet pressure and temperatures, thermal neutron spectra, steam cycle options, materials, etc.). Therefore, the R&D needs for each reactor type are common; 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. Improvements in the areas of safety, sustainability, and proliferation resistance and physical protection are also possible and are being pursued by considering several design options using thermal and fast spectra, including the use of advanced fuel cycles.

Status of the co-operation

There are currently four Project Management Boards (PMBs) within the SCWR System: 1) System Integration and Assessment (provisional), 2) Materials and Chemistry, 3) Thermal-hydraulics and Safety, and 4) Fuel Qualification Testing (provisional). Table 1.1 and 1.2 list the signatories of the System Arrangements and Project Arrangements. China, which hosted the 6th International Symposium on SCWR as well as the SSC meeting in Shenzhen in March 2013, expressed its interest to participate in GIF SCWR activities.

3.3.2 R&D objectives

The following critical-path R&D projects have been identified in the SCWR System Research Plan:

- System integration and assessment: Definition of a reference design, based on the pressure tube and pressure vessel concepts, that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance.
- Thermal-hydraulics and safety: Gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed validating thermal-hydraulic codes. The design-basis accidents for a SCWR have some similarities with conventional water reactors, but the difference in thermal-hydraulic behaviour 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: Qualification of key materials for use in in-core and out-core components of both pressure tube and pressure vessel designs. Selection of a reference water chemistry which minimises materials degradation and corrosion product transport will also be sought based on materials compatibility and an understanding of water radiolysis.
- Fuel qualification test: An important collaborative R&D project is to design and construct an in-reactor fuel test loop to qualify the reference fuel design. As a SCWR has never been operated before, such generic testing is considered to be mandatory before a prototype reactor can be licensed.

3.3.3 Main activities and outcomes

System integration and assessment

Significant progress has been achieved in 2013 in particular in Canada. Canada is focusing on the development of a pressure-tube type SCWR concept, which is evolved from the well-established CANDU® reactor. The Canadian SCWR is designed to produce electrical energy as the main product, plus process heat, hydrogen, industrial isotopes, and drinking water (through the desalination process) as supplementary products, all within a more compact reactor building.

The pre-conceptual Canadian SCWR maintains a modular configuration with separated coolant and moderator, as in current CANDU reactors. It is developed to generate 2 540 MW of thermal power and about 1 200 MW of electric power (assuming a 48% thermodynamic cycle efficiency of the plant). A batch refuelling strategy is adopted as the current CANDU practice of on-line refuelling is extremely challenging at the proposed higher operating pressure and temperature. This has led to a simplified vertical core design with vertical fuel channels, each containing a fuel assembly. Figure 3.17 illustrates schematically the pre-conceptual Canadian SCWR core.

The pre-conceptual Canadian SCWR core consists of 336 fuel channels, each housing a 5-m long fuel assembly. The average fuel channel power is 7.56 MWth and the core radial power profile factor is estimated to be 1.32. The lattice pitch of the channels is selected to be 250 mm based on recent optimisation results for the fuel-to-moderator ratio to achieve a negative void coefficient, and high fuel burnup. Some fuel channels at the outer region of the core could be used for a reheat option.

The fuel-channel configuration in the fuel-assembly region consists of the pressure tube and an insulator encapsulated with a stainless-steel jacket (Figure 3.19). This fuel channel design is called the high-efficiency channel (HEC). The pressure tube is designed to withstand the high coolant pressure at a low temperature (~100°C), achieved by direct contact of the pressure tube with the moderator. This allows the use of the zirconium alloy Excel for the pressure tube.

The insulator thermally protects the pressure tube from the high temperature coolant flowing through the fuel assembly. It is made of yttrium-stabilised zirconia, which is refractory, has low neutron absorption properties and excellent resistance to neutron damage. The insulator is encapsulated with a stainless-steel jacket, which will minimise damage to the insulator by the movement of the fuel assembly and prevent any fragments of the insulator from getting into the coolant stream and damaging the turbines should the insulator fragment. One of the expected benefits of the HEC concept is that in the event of a loss of coolant accident (LOCA) without emergency core cooling, the fuel will not melt because of passive heat rejection through the insulator into the moderator.

The fuel assembly consists of 64 elements, distributed into two rings (i.e., 32 elements in each ring). Elements in the inner ring have an outer diameter of 9.5 mm and those in the outer ring have an outer diameter of 10.5 mm. The overall fuel-assembly length is 5 metres. Wrapped-wire spacers are used to maintain the gap size between elements in the same ring. CANDU-type bearing pads are used to maintain the gap size between outer-ring elements and the insulator cladding and that between inner-ring elements and the inner flow tube. In addition, CANDU-type spacers are used to maintain the gap size between elements in the outer and inner rings. A stainless steel or nickel-based alloy will be used as the cladding material. Five candidate cladding materials are currently undergoing for in-depth material testing.

The calandria vessel holds the low-pressure and low-temperature heavy-water moderator surrounding the fuel channels. It is a low-pressure tank containing the fuel channels, moderator, reactivity control mechanisms and emergency shutdown devices. Control and shut-down rods are installed from the side of the calandria vessel. A second shut-down system would also be installed providing gadolinium injection at various levels of the calandria vessel.

The thorium fuel cycle is implemented into the Canadian SCWR to meet the GIF technology goals for enhanced safety, resource sustainability, economic benefit and proliferation resistance.

Thorium fuel has only been used in research reactors and demonstration irradiations were performed in power reactors. Recent studies of thorium-based fuel cycles in contemporary CANDU reactors demonstrate substantial reductions in natural uranium (NU) requirements of the fuel cycle via the recycle of ^{233}U bred from thorium.

As thorium itself does not contain a fissile isotope, reactor-grade plutonium is selected as the fissile material to provide neutrons (the use of enriched uranium or ^{233}U bred from an earlier thorium cycle has also been considered). The plutonium (on average 13%) is mixed uniformly with the thorium in the fuel pellets, which are inserted into the elements of the fuel assembly. A 3-batch refuelling scheme is adopted (see Figure 3.20). The exit burnup of the fuel is about 50 MWd/kg Heavy Metal.

Figure 3.17: Schematic diagram of the pre-conceptual Canadian SCWR core

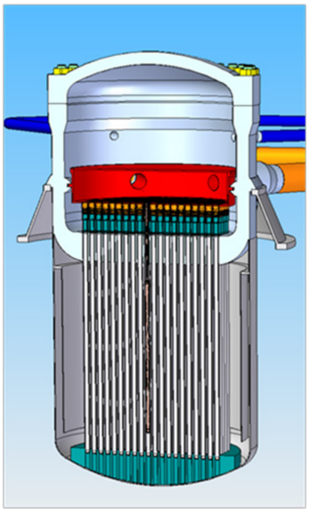


Figure 3.18: Schematic diagram of the pre-conceptual Canadian SCWR reactor building configuration

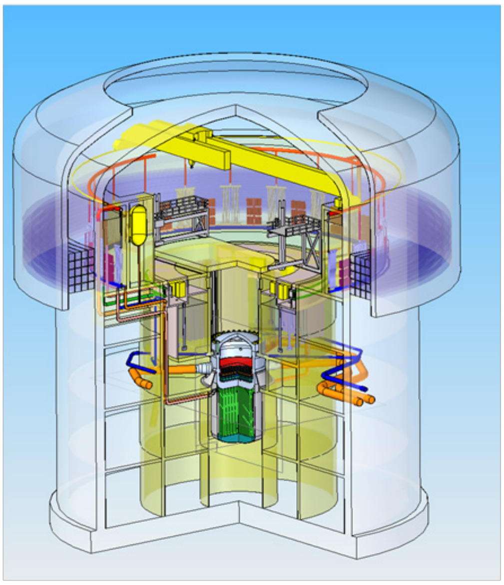


Figure 3.19: Schematic diagram of the high efficiency channel of the pre-conceptual Canadian SCWR

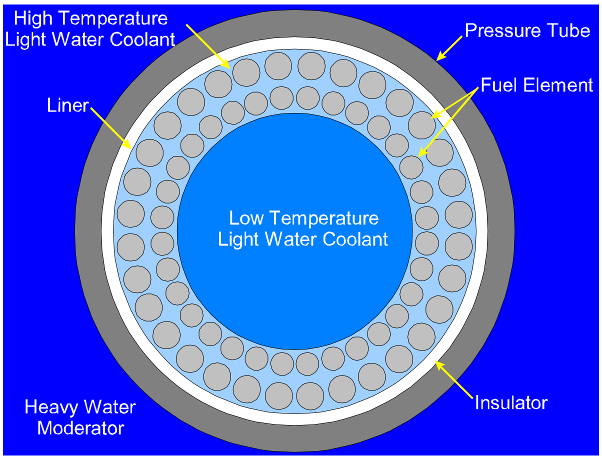


Figure 3.20: Fuel-loading map for the Pu-Th fuel in the pre-conceptual Canadian SCWR

	1	2	3	4	5	6	7	8	9	10
A							20_2	1_1	21_3	2_1
B					3_1	9_2	4_1	3_3	5_1	8_3
C			6_1	4_3	12_3	7_1	20_3	8_1	2_3	9_1
D			18_3	10_1	5_2	2_2	17_2	5_3	13_2	8_2
E		11_1	15_3	24_2	10_3	12_1	16_2	13_1	7_2	13_3
F		27_2	14_1	26_2	15_1	9_3	16_1	17_3	16_3	17_1
G	25_2	18_1	25_3	28_2	19_2	19_1	27_3	12_2	20_1	7_3
H	21_1	11_3	22_1	24_3	23_1	28_3	15_2	6_2	4_2	1_2
J	1_3	24_1	26_3	23_2	14_2	19_3	25_1	18_2	6_3	3_2
K	26_1	22_3	27_1	22_2	23_3	28_1	14_3	21_2	11_2	10_2

i_1	Fresh Fuel
i_2	Once Burned
i_3	Twice Burned

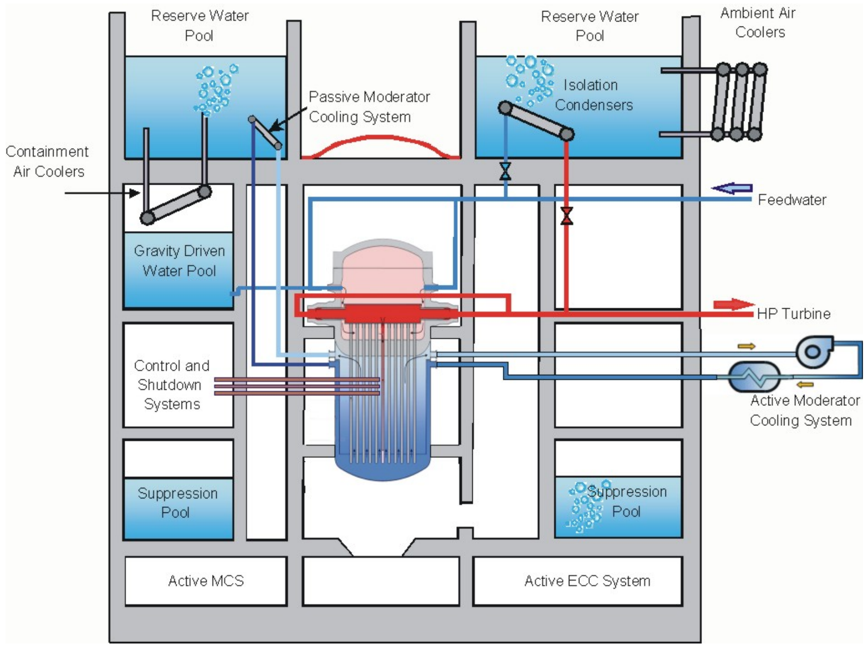
i = assembly label

The safety concepts for the Canadian SCWR are generally similar to those developed for modern nuclear reactors, but specific considerations are necessary to cover the transition through the pseudo-critical temperature. Figure 3.21 illustrates schematically the safety system features. Passive safety concepts have been incorporated to support the inherent safety goals required in next generation nuclear reactors:

- The Canadian SCWR fuel is designed to exhibit a negative coolant void reactivity coefficient throughout its residence time in the core.
- As in the CANDU reactor, the Canadian SCWR maintains the separation of the primary coolant from the moderator, providing a large heat sink (moderator) in case of a LOCA within the HTS.
- In the event of a LOCA without emergency core cooling, heat from the fuel will be rejected passively through the HEC to the moderator, maintaining the fuel and cladding temperatures below melting.

To ensure the effectiveness of long-term cooling, a passive moderator cooling system is introduced to remove decay heat from the fuel.

Figure 3.21: Schematic diagram of the safety systems of the pre-conceptual Canadian SCWR

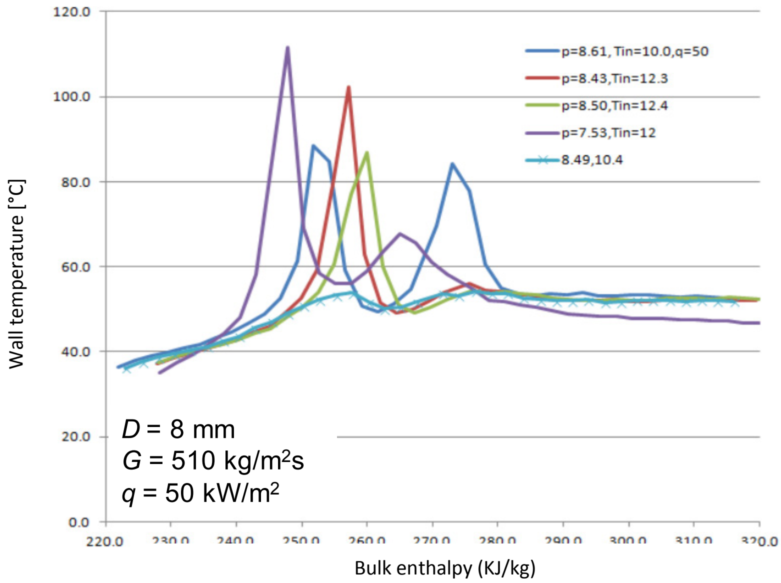


Thermal-hydraulics and safety

Supercritical water is a single phase fluid having liquid-like properties below the pseudo-critical temperature (384°C at 25 MPa) and steam-like properties above this temperature. Heat transfer of supercritical water differs fundamentally from ordinary fluids in the vicinity of the pseudo-critical point, where the fluid properties vary significantly with temperature. Heat transfer in this range can be enhanced at low heat flux compared with ordinary fluids, or deteriorated at high heat flux and low mass flux, causing local hot spots on the heated surface. Prediction of such hot spots still remains a challenge. Up to now, simple heat transfer correlations cannot predict these phenomena properly and computational fluid dynamics (CFD) or even large eddy simulations are taken instead. Similar questions arise with critical flows through orifices or breaks and with stability limits of supercritical fluids in heat exchangers if the pseudo-critical point is located in the computational domain. New physical models and codes describing these phenomena need to be validated by experiments with supercritical water or at least with surrogate fluids having similar properties, like supercritical CO_2 or refrigerants.

The Canadian programme has generated a large amount of experimental heat-transfer data in support of the SCWR concept development. At this point, the majority of data were obtained with tubes (which were installed for loop commissioning) and annuli. Figure 3.22 illustrates the wall temperature measurements obtained with the carbon dioxide flow inside an 8-mm tube at various pressures and inlet fluid temperatures for the mass flux of 510 kg/m^{2}s and the heat flux of 50 kW/m^2. Deteriorated heat transfer has been observed at some conditions, but not the others. Figure 3.23 compares experimental heat-transfer coefficients obtained in this study and those of Fewster and Jackson (2004) at similar test conditions. Very good agreement between the two sets of experimental data, hence improving the confident on the new data.

Figure 3.22: Wall-temperature measurements obtained with carbon dioxide flow in an 8-mm tube



A separate heat-transfer experiment has been completed with a 12.5-mm tube in supercritical Refrigerant-134a flow. Figure 3.24 illustrates the variations of wall-temperature measurements with bulk-fluid enthalpy and heat flux. Deteriorated heat transfer (with a sharp wall-temperature rise) has been observed at high heat fluxes but not at the low heat flux.

The supercritical water heat-transfer experiment using an annulus test section has been extended to down flow conditions. Figure 3.25 compares wall-temperature measurements between up flow and down flow. Wall-temperature measurements are about the same between up flow and down flow at the heat flux of 200kW/m^2 but are generally higher for up flow than down flow at the heat flux of 1 000 kW/m^2.

Figure 3.23: Comparison of heat-transfer coefficients for carbon dioxide flow between this study and Fewster and Jackson (2004)

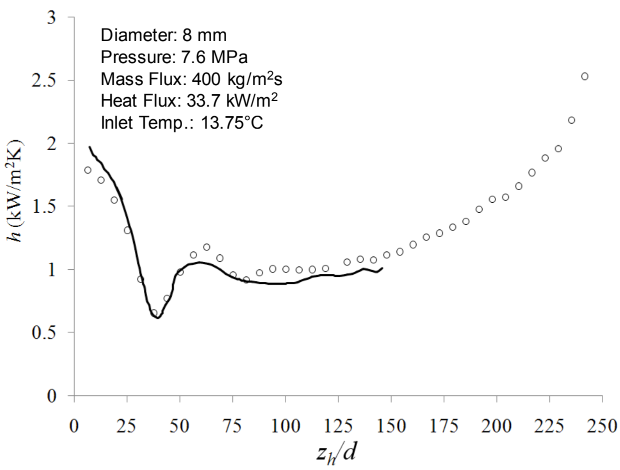


Figure 3.24: Wall-temperature measurements obtained with refrigerant-134a flow in a 12.5-mm tube

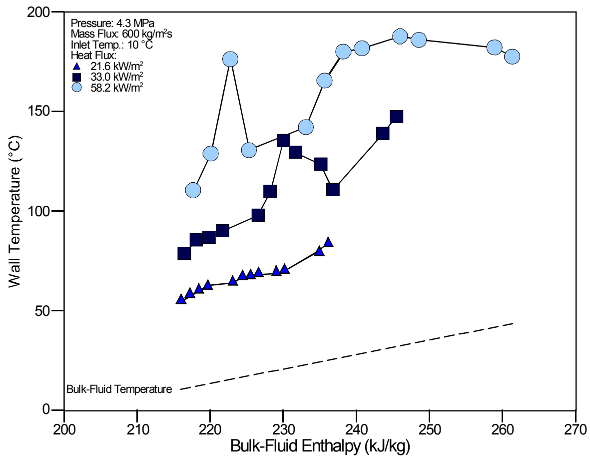
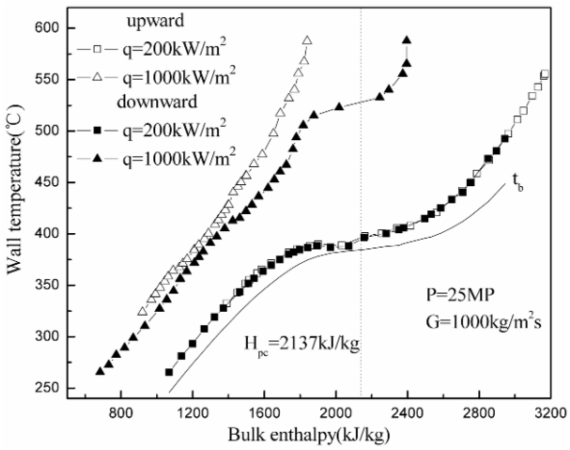


Figure 3.25: Comparison of wall-temperature measurements between upward and downward flows of supercritical water in an annulus

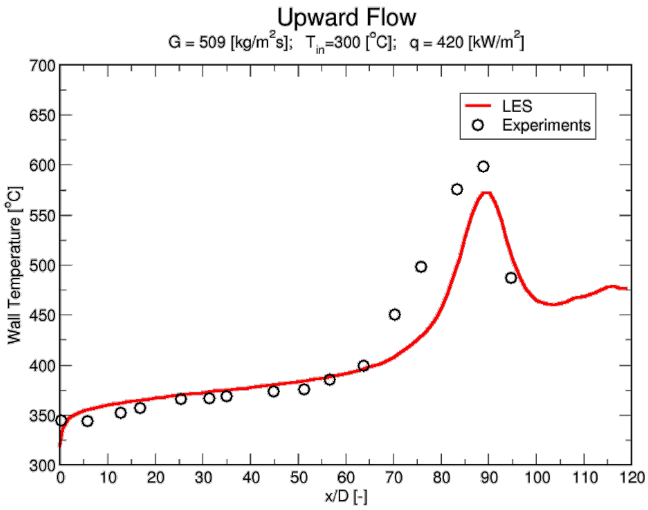


Since the large project on the European SCWR design (the HPLWR) was finalised in 2010, most of the activities in the EU have been focusing on heat transfer in supercritical water and other fluids and system stability. Heat transfer in supercritical water (and other fluids) is a challenge to model, as properties such as density and specific heat capacity drastically change near the pseudo-critical point. Such advanced models, which can be implemented in CFD codes, are of paramount importance to accurately predict the wall temperatures of the HPLWR fuel assemblies.

In 2010, the thermal-hydraulics of innovative nuclear systems (THINS) project was initiated. One of its work packages aims to study the performance of existing turbulence models and improve them (or even develop new ones). On top of this, a reference database will be created and used, based on LES/DNS simulations and experiments. One of the recent results is an LES simulation of supercritical water flow in upward direction, showing good correspondence with experiments (see Figure 3.26).

An experimental setup, based on Freon HFC23 has been built to obtain detailed flow data of annular flow and of three jets impinging on a wall (representing a flow that can be found in the upper plenum of the HPLWR). Laser techniques (laser doppler anemometry and particle image velocimetry) and a fast infrared camera will be used to measure near wall velocities and temperatures of the wall (see Figure 3.27). The latter experiment aims to study thermal fatigue in supercritical flows. Results are to be expected in the beginning of 2014.

Figure 3.26: Wall temperature of a large eddy simulation (LES) of upwards supercritical water flow. Experimental data from Pis'menny et al. (2006), LES by Niceno and Sharabi (2013)



The University of Stuttgart has developed an algebraic model for the buffer layer, thereby taking the roughness of the wall into account. It has been found that roughness plays a significant role above a certain critical Reynolds number. KIT has been working on transient heat transfer during depressurisation from supercritical to sub-critical pressure with the help of a one-dimensional model. The model has been validated using transient heat transfer data of an electrically heated tube. This model will be applicable to system codes to describe the temporary boiling crisis right after depressurisation. The latter institute has been working on LES model development (dynamic Pr_{SGS}) as well. A good correspondence has been found with a DNS study performed by Bae et al. (2005) at Re=5 400.

A 5-year Dutch research programme was initiated in 2012 by the Delft University of Technology, aiming at a deeper understanding of heat transfer in supercritical water flows. Experiments as well as numerical work will be performed. Various geometries (annular flow, impinging jets, Rayleigh Benard cell) will be used.

In Hungary, both the Institute of Nuclear Techniques (Budapest University of Technology and Economics) and the Centre for Energy Research (Hungarian Academy of Sciences) have been working on SCWRs. They make use of research facilities such as the Dynamic Radiography Station of the Budapest reactor and the ANCARA closed loop with natural circulation (see Figure 3.28). The latter is used to study the steady-state characteristics of such a loop (power-flow map) and loop stability.

Loop stability has been studied by the Delft University of Technology as well with the help of the DELIGHT facility (a scaled version of a natural circulation-driven HPLWR) and numerical codes. Amongst other things, it has been found that at least two different stability modes exist, i.e. a natural circulation driven mode (low frequency) and a friction driven one (high frequency). The University of Pisa has been working on the simulation of the stability of heated channels with supercritical fluids. Ledinegg instabilities as well as dynamics instabilities have been studied.

Recently, the Russian Federation proposed to contribute to the thermal-hydraulic and safety activities by a few experimental works. The Experimental and Design Bureau "Gidropress" plans to design, mount and operate a small supercritical water (SCW) rig "MUV" for experimental simulation of heat transfer of SCW flow in tubes, in annular channels formed by a fuel rod simulator inside a tube, and in a carved channel with a bundle of 2-3 fuel rod simulators. Heat transfer for water flows at subcritical pressure of 18-22 MPa is envisaged to be conducted in the MUV facility as well. Measurements of the pressure loss in the heated and cooled SCW flows (with special attention to the area of pseudo-critical transition) will be conducted at the MUV rig. Based on the results of these experiments, some new (or renewed) correlations for heat-transfer coefficient and friction factor will be developed for sub-critical and supercritical flows in simple channels and rod bundles.

Figure 3.27: LDA of annular, supercritical flow at the Delft University of Technology, the Netherlands. A special measurement section of glass has been built that can withstand the pressure of 5.7 MPa (HFC23) and limits the amount of refraction of the laser beams.

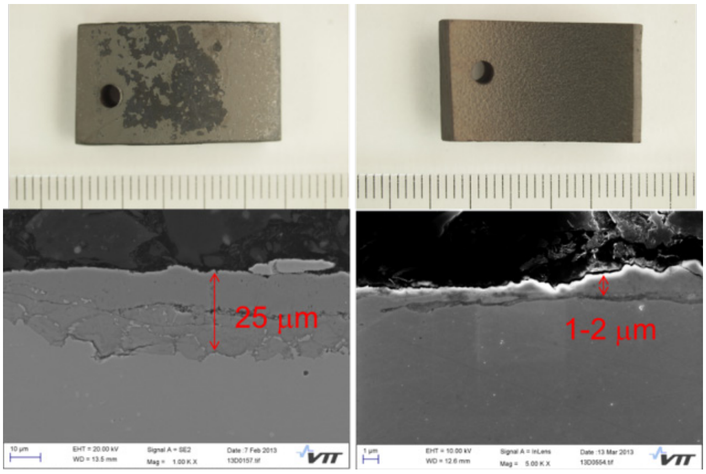
Figure 3.28: The Hungarian ANCARA supercritical water loop with natural circulation, consisting of 4x600 W heater elements, a flow metre and a range of pressure sensors and thermocouples. The heated length amounts to 1 000 mm.

In the field of the problem of thermal-hydraulic stability, the Russian Federation plans to conduct experimental investigations of the limits of onset of instability regimes and amplitudes of oscillations of pressure and SCW flow rate. The phenomena accompanying mixing of superheated SCW as pseudo-vapour with sub-cooled SCW as pseudo-liquid in a nozzle mixing chamber is planned to be explored at the MUV rig as well. Preparation of a data bank for SCW tubes goes on now in a few institutions of the Russian Federation

Materials and chemistry

Low Cr austenitic stainless steels are very attractive for the near-term purposes due to their proven track record in nuclear field applications. Studies have shown that cold working austenitic stainless steels is beneficial in reducing the oxidation rate under SCW conditions. Shot peening is one method to produce deformation and compressive stresses at the sample surfaces. Based on promising results obtained fora machined 316L tube specimen, the effect of shot peening on oxidation resistance of low Cr austenitic stainless steels (347H, 347HFG, 316NG) was studied, including thin wall 316L tube samples. Shot peening was found to significantly enhance the oxidation resistance of the materials at 550 °C and 650 °C with ~150 ppb or 8 ppm dissolved oxygen in the inlet flow. Figure 3.29 shows the preliminary results of a commercially shot peened 347HFG tube section after 1 000 h of exposure at 550°C/25MPa. Based on the results, shot peening can be seen as a promising surface treatment method to improve low Cr austenitic stainless oxidation resistance in SCW. However, the effect of higher dissolved oxygen content (~8 ppm) at higher temperatures as well as long term exposure tests (> 5 000 h) must be studied further.

Figure 3.29: The appearance (top) and SEM cross-sections (bottom) of commercially shot peened 347HFG section after 1 000 h of exposure at 550°C/25MPa with 8 ppm dissolved oxygen in the inlet flow. Outer surface (left) not shot peened and inner surface (right) shot peened surface.



The M&C PMB Project Plan recognises that coatings could be of value for SCWR, and would also have applications in existing reactors. A possible path forward for fuel cladding development, may be coating of austenitic stainless steels or F/M steels if ODS or austenitic steels are found to be unsuitable. Small coated samples (AISI 316L and Inconel 600) were exposed to SCW (650°C/25MPa) up to 1 000 h. Whereas the performance of the TiAlN and the ZrO_2 coatings was not satisfactory, the CrN coating was found to provide a stable and protective corrosion barrier. The CrN coated Alloy 600 base metal showed that chromium creates approximately 0.5 µm wide stable natural passivation oxide layer protecting the surface from further corrosion attack during exposure to SCW. Fe-22Cr-6Al-0.6Y and Ni-20Cr-5Al model coating alloys were tested in SCW (500°C, 25 MPa) for over 6 000 h. The weight change was very sensitive to the surface preparation method, with a ground surface showing the lowest weight gain. The Ni-20Cr-5Al material showed the best corrosion resistance.

VTT, in co-operation with JRC-IET, has been working in recent years on a new type of miniature autoclave with a bellows-based loading device. Miniature loading devices are attractive due to their small size, which is expected to decrease costs and at the same time enables future irradiation-assisted stress corrosion cracking (IASCC) testing to be performed in-pile. First calibration and SCC crack growth rate tests were performed in SCW at VTT and the results were compared to those gained in subcritical water at 288°C using the same loading parameters and device. Very good accuracy and stability during the approximately two-month long test were achieved.

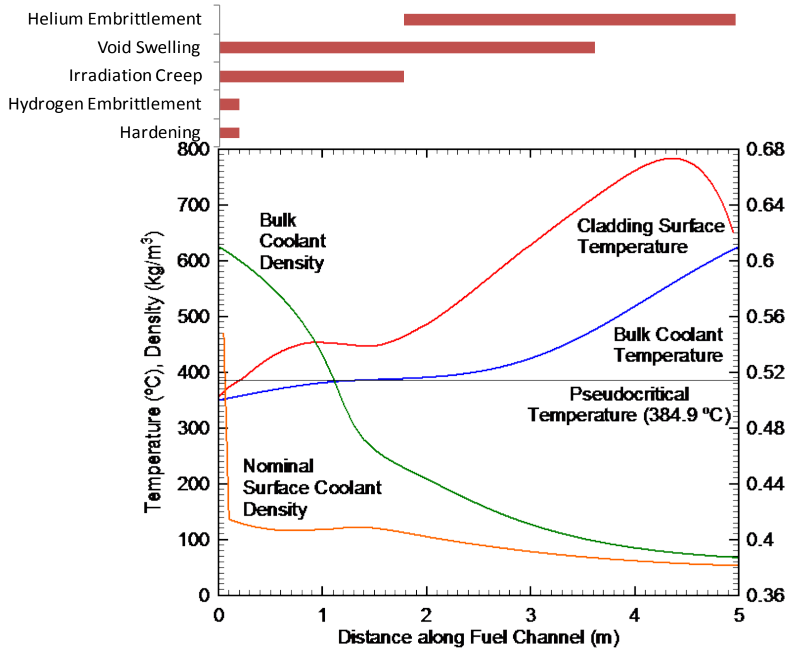
A major activity of the M&C PMB has been the organisation of a Round Robin corrosion test exercise between PMB partners to compare the results of corrosion tests in different test facilities in different laboratories using a standard test protocol and coupon preparation method with materials from the same batch. The tests started in 2012 and are now underway and will be completed by the end of 2013.

A summary of the results will be made available to the PMB members in early 2014.

In Canada, a major step forward in fuel cladding material selection was made in 2013 by the adoption of a collapsible fuel cladding concept. A collapsible stainless steel fuel cladding was evaluated in tests performed at the Vallecitos BWR in 1961 using a fuel element constructed with a 0.71 mm thick 304 stainless steel cladding, and exposed to superheated steam at 6.9 MPa for a total of 617 h, including 80 h at a maximum steam temperature of about 480°C. A key conclusion of the test was that a fuel element with a thin-walled cladding supported by the UO_2 fuel could be made workable for nuclear superheat applications. This decision significantly reduced the mechanical properties requirements for the fuel cladding; as a result, five candidate alloys were selected for further assessment. A major review of the available data on these five alloys was performed and major gaps identified. These five alloys are now undergoing detailed testing to close these gaps, including corrosion and SCC tests in superheated steam at 800°C, the current predicted peak cladding temperature.

A major knowledge gap is in the behaviour of candidate alloys under irradiation. However, compared to other Gen IV concepts, the expected radiation doses for the SCWR concepts do not differ substantially from those experienced by Generation II and III water-cooled reactors, although the range of temperatures at which the materials are irradiated is much larger (Figure 3.30). In Canada, calculations of radiation damage and helium production were performed for several alloys of interest; for Alloy 800HT, dpa and He production values of 9 dpa and 50 appm He respectively were obtained. The use of a Pu-Th fuel in the Canadian SCWR concept reduces the thermal neutron flux, reducing both the damage and He production compared to Generation II/III reactors. The large body of data on radiation damage available from the US nuclear superheat development programme in the 1960s is currently being reviewed. For example, during this programme, Alloy 800 was irradiated in a number of test reactors including the EVSBR, where fuel assemblies with 0.406 mm thick fuel cladding were exposed to superheated steam at temperatures up to 738 ± 83°C for 10 292 hours. An extensive post-irradiation examination EVSBR bundle KB-40, which failed after 6 188 hours at power at 493°C, showed the cause was most likely low-cycle fatigue. The strength of the Alloy 800 cladding material was not significantly affected by irradiation but the ductility was reduced markedly. Combined with targeted modelling, these data will enable reasonable predictions of effects such as swelling, He embrittlement and irradiation creep to be made, and confirmed once suitable in-reactor test facilities are available.

Figure 3.30: Bulk and surface coolant densities along the heated section of the Canadian SCWR fuel channel. The nominal surface coolant density is calculated assuming that the surface has no effect on the structure of the adjacent fluid. The figure also indicates the dominant irradiation damage mechanisms throughout the core.



At the University of Sherbrooke, a number of advances in the modelling of water radiolysis were made that are improving our ability to model this phenomenon, including determination of the temperature dependence of the rate constant of the self-reaction of the hydrated electron, characterisation of the importance of the reaction: $H^{\bullet} + H_2O \rightarrow H_2 + {}^{\bullet}OH$ as a possible source of molecular hydrogen and $^{\bullet}OH$ radicals in the low-LET radiolysis of water above 200°C, and understanding the influence of the geminate recombination of subexcitation-energy electrons prior to thermalisation on the density dependence of the yield of hydrated electrons in the low-LET radiolysis of SCW at 400°C. In ~500 h tests of candidate alloys using the SCW convection loop of the Canada-Ukraine Electron Irradiation Test Facility with an irradiation cell coupled to a 10 MeV, 10 kW linear electron accelerator at temperatures in the vicinity of the critical point, the outlet sample water was found to contain 3-5 $\mu g{\cdot}kg^{-1}$ of oxygen; due to the long sample line the actual oxygen concentration in the test sections was likely higher. The water conductivity increased systematically up to ~23 $\mu S{\cdot}cm^{-1}$ indicating release of metals into the coolant due to corrosion, and the water contained Cr at concentrations up to 54 $\mu g{\cdot}L^{-1}$; there was no detectable Cr in the water before exposure to the electron beam. These tests clearly demonstrate the potential for radiolysis-enhanced corrosion in an SCWR core.

Fuel qualification test

A fuel qualification test facility, required for the licensing of a nuclear facility operated with supercritical water, is planned to be installed in the LVR-15 research reactor in Řež, Czech Republic. The fuel qualification test is planned to be performed under evaporator conditions of the Euratom SCWR concept. A pressure tube with 57-mm outer diameter and 9-mm wall thickness will replace one fuel assembly of the LVR-15 reactor. The pressure tube will contain 4 fuel rods with 8-mm diameter at 9.44-mm pitch, like the HPLWR assembly concept, inside a square assembly box. The rod length will be limited to 600 mm to match the core height of the research reactor. With a ^{235}U enrichment of almost 20%, these 4 fuel rods can reach a fissile power of more than 60 kW. A recuperator inside the pressure tube, situated right above the fuel rods, will be used to boost the feed-water temperature of 300°C to typical evaporator conditions. A single recirculation pump will drive the primary loop, operating at around 25 MPa system pressure.

Valuable contributions to the fuel qualification test project will be coming from material and chemistry research, such as information on water radiolysis, for which a model is being developed and will be useful for the fuel tests. Three candidate cladding materials for initial corrosion tests have been selected. The tests results are expected to provide the necessary information to the Czech regulator for conducting tests at 550°C coolant temperature, emphasising the need to have the information 2-3 years in advance for material qualification. Materials and chemistry information specific to the fuel tests (e.g. test data on the candidate alloys, water chemistry data specific to the loop) are planned.

Little is known about the behaviour of fuel and fission products in an SCWR, and there is a need to investigate the interaction of SCW and fuel and to study the solubility and transport of fuel contaminants. A test plan has been developed to obtain some of this information in leaching tests of SIMFUEL, which contains natural UO_2 and non-radioactive additives to simulate the fission products present in irradiated fuel. SIMFUEL can be used to estimate the thermal, mechanical, and chemical characteristics of fresh and irradiated Canadian SCWR fuel. The leaching tests are being performed in a Parr 250 mL autoclave at a temperature of 400ºC and a pressure of 25 MPa, and the results will be available in early 2014.

The proposed project plan includes a first test phase at around 400°C coolant temperature with qualified cladding alloys to commission the test facility, followed by tests with elevated coolant temperatures up to 500°C using the advanced low carbon alloy 310S for fuel claddings.

The following work progressed in 2013:

- The supercritical water loop has been designed, including the pressure tube and its test section to be inserted into the LVR-15 research reactor after commissioning tests, the block with auxiliary systems with recirculation pumps, measurement and cleaning systems, and including its safety systems needed in case of loss of coolant or of power supply. Figure 3.31 shows the planned arrangement of these systems inside the reactor building. The auxiliary systems are located in a new, neighbouring experimental hall shielded from the working area in the reactor hall. A shielded duct bridge from the reactor to the experimental hall carries coolant supply lines and emergency cooling lines to the reactor.
- Construction work of the new experimental hall has been started in 2013 and is planned to be completed in spring 2014.
- Figure 3.32 shows details of the in-pile test section inside the pressure tube. The flow inside the pressure tube has been analysed with computational fluid dynamics and the expected cladding surface temperature distribution of the fuel rods has been predicted. Results show that the peak cladding temperature will stay well below 550°C, such that the temperature limit for the selected stainless steel claddings will be met.
- Safety systems for the fuel qualification tests have been designed, including passive accumulators supplying coolant during depressurisation to sub-critical pressures, two active emergency cooling systems for residual heat removal, and a passive long term heat removal system to the reactor pool by flooding the insulation gap. Transient analyses with the system codes APROS and ATHLET have been performed, covering loss of coolant accidents, anticipated blockage or bypass of coolant for the test section, trip of the primary pump, loss of power supply, accumulation of radiolysis gas in stagnant coolant line and the effect of reactivity insertion by change of coolant temperatures. Mechanical and radiological consequences of a postulated failure of the high pressure system have been determined to assess the risk of the test loop.
- Corrosion tests at the maximum allowable temperature of 550°C have been performed in an autoclave at supercritical pressure up to 3 500 hours exposure time. These tests cover all envisaged test conditions of the first phase, confirming that the selected stainless steels will be applicable. Slow strain rate tests (SSRT) and SCC tests have been performed in supercritical water at the upper temperature limit to qualify the selected cladding

alloys. For fuel claddings, SS316L has been identified as the most appropriate candidate material for the first test phase. A fuel rod mock-up has been built to test the end fittings of the fuel rods in an out-of-pile autoclave prior to construction of the real fuel rods.

- A bilateral agreement between Euratom project partners and a Chinese consortium enabled to contribute results of the Chinese project SCRIPT to the fuel qualification project. Additional thermal-hydraulic analyses, material tests and tests of fluid-structure interaction in the test section have been a valuable support of the project in 2013. An out-of-pile test with electrically heated rods of same design as the planned in-pile tests has been designed in China. It shall be built until spring 2014 to serve as a pre-qualification test at full pressure, temperature and heat flux, confirming thermal-hydraulic and mechanical predictions.
- Experts contributing to design and analyses of the fuel qualification tests have been trained in a quality management workshop in Shanghai as well as in a workshop in Budapest on codes and experimental methods to analyse SCWRs.

Design and analyses have been completed in order to present the planned fuel qualification test to the international advisory board.

Figure 3.31: Arrangement of the SCWR fuel qualification test loop in the LVR-15 reactor and in the new experimental hall in Řež

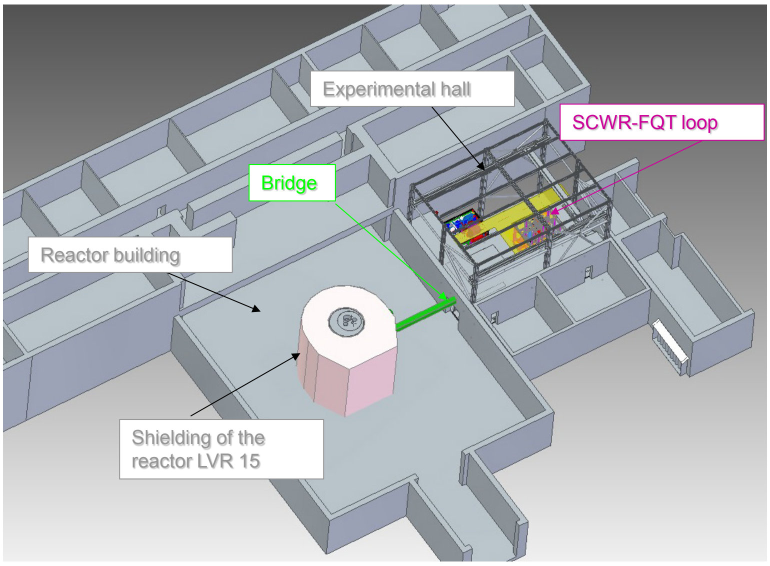
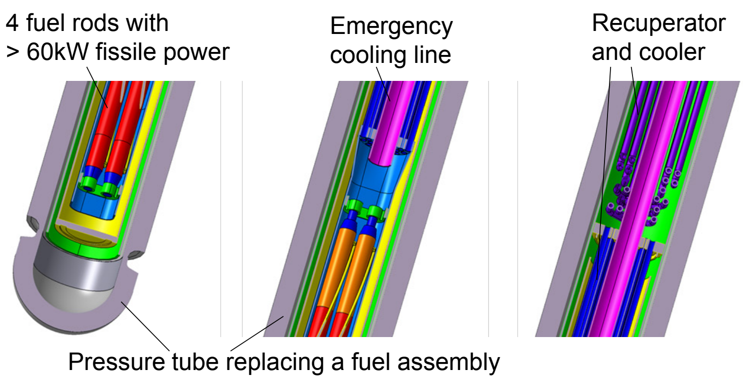


Figure 3.32: Details of the in-pile test section for SCWR fuel qualification test



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3.4 Gas-cooled fast reactor (GFR)

3.4.1 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 minimisation (through fuel multiple reprocessing and fission of long-lived actinides), with those of high-temperature systems (high thermal cycle efficiency and industrial use of the generated heat, 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.

The reference design for GFR is based around a 2 400 MW_{th} reactor core contained within a steel pressure vessel. The core consists of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fuelled pins contained within a ceramic hex-tube. The favoured material at the moment for the pin clad and hex-tubes is silicon carbide fibre reinforced silicon carbide. Figure 3.33 shows the reactor core located within its fabricated steel pressure vessel surrounded by main heat exchangers and decay heat removal loops. The whole of the primary circuit is contained within a secondary pressure boundary, the guard containment. The coolant is helium and the core outlet temperature will be of the order of 850°C. A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle (Figure 3.34) containing a helium-nitrogen mixture which, in turn drives a closed cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plant so represents an established technology, with the only difference in the GFR case being the use of a closed cycle gas-turbine.

Figure 3.33: GFR reference design

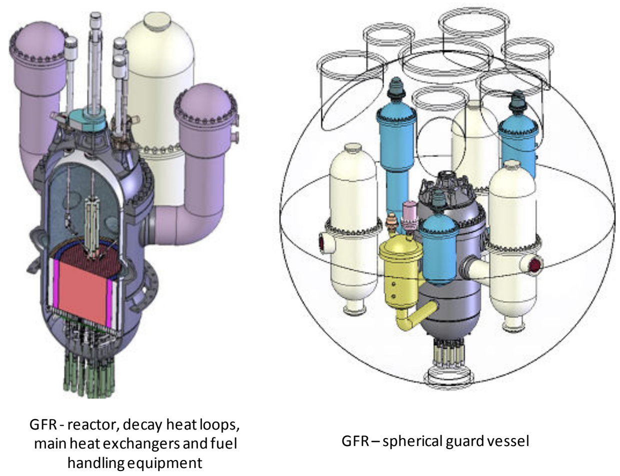
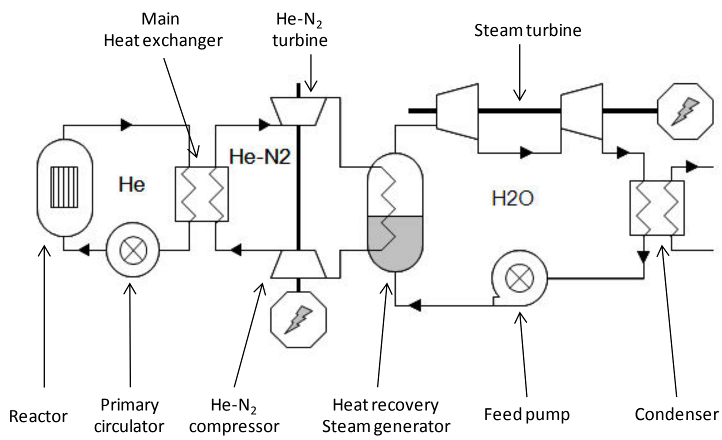


Figure 3.34: GFR indirect combined cycle power conversion system



The proposed experimental reactor ALLEGRO (formerly ETDR) could become the first gas-cooled fast reactor to be constructed. Being a small experimental reactor (75 MW_{th}), the objectives of ALLEGRO are to demonstrate the viability and to qualify specific GFR technologies such as the fuel, the fuel elements and specific safety systems in particular, the decay heat removal function, together with demonstrating that these features can be integrated successfully into a representative system. So far, ALLEGRO development has been driven by the French national programme with significant contributions from Euratom and Switzerland. In 2010 a memorandum of understanding was signed between the Czech Republic, Slovakia and Hungary as partners to support each other in bidding for one of them to host ALLEGRO, with assurances that the two other partners would provide technical and administrative support to the successful host nation. In 2012 the consortium was joined by Poland and subsequently became a legal entity led by VUJE from Slovakia in 2013. A centre of excellence was established within the four countries of the consortium, the Visegrad 4 (or V4), to act as a focal point for the development of ALLEGRO and GFR and has been named the V4G4 Centre.

Status of co-operation

The system arrangement was signed at the end of 2006 by Euratom, France, Japan and Switzerland. It is to be noted that, while France and Japan have been very active in the development of the GFR concept, providing regarding conceptual design, safety assessment and fuel development in the previous years, in 2010 French research priorities were re-focused on sodium-cooled fast reactors, which led to a reduction of effort on the GFR system. Further, the Fukushima Daiichi accident in 2011 further refocused priorities away from GFR in Japan, and to a lesser extent in Switzerland. Switzerland has reduced its funding for Generation IV activities and its main input into Generation IV now comes from its contribution to Euratom activities.

Two projects were discussed at the origin of the SA, dealing with conceptual design & safety (CD&S), and fuel and core materials (FCM). The conceptual design & safety project arrangement was signed in 2009 by Euratom, France and Switzerland, and is effective as of 17 December 2009 which ran until the end of 2012. All of the agreed deliverables for the period 2009 and 2012 from all of the partners have been delivered to and registered by the GIF Secretariat. The Fuel and core materials project arrangement remains unsigned and the participants including the Institute of Advanced Energy of Kyoto University, Japan, which newly joined in March 2013, have agreed to continue their collaboration on an informal basis. New project plans covering the period 2013 to 2015 are now required for both the CD&S PMB and the FCM PPMB. Establishing these new plans is proving to be difficult because of the funding situation now affecting all of the signatories to the SA.

France and Switzerland continued to be very active members within Euratom, with a number of organisations in France and PSI in Switzerland being members of the GoFastR project (Euratom FP7), which provided the main contribution from Euratom to the GIF GFR system development until the project finished in February 2013. Euratom's ALLIANCE project started in 2012 and is aimed at developing the case for implementing ALLEGRO in one of the V4 countries, and whilst the Coordinator of this project has been appointed to be a member of the GFR SSC, the project does not have the resources to provide Euratom's future technical contributions to the GFR projects. A further Euratom FP7 project was launched in September 2013 called ESNII+ and its objective is to provide administrative and technical support to develop the European Sustainable Nuclear Industrial Initiative (ESNII) to be in a future position to manage the development of fast reactor demonstration plants within Europe. ALLEGRO is one of the ESNII systems so there is some R&D work with ESNII+ that is of relevance, specifically cross-cut work on the development and characterisation of MOX fuels and determination of uncertainties in ALLEGRO core physics calculations. Again, unfortunately, ESNII+ is a wide-ranging project that does not have the resources to make a technical contribution on behalf of Euratom into the projects of the GIF GFR system.

A number of papers were presented at the IAEA's FR13 conference, held in Paris in March 2013; papers on the GFR system were presented by France, Switzerland and Euratom.

3.4.2 R&D objectives

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

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

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

- Define 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);
- Identify and study alternative design features (e.g. lower temperatures, pre-stressed concrete pressure vessel, diverse decay heat removal systems);
- Define appropriate safety architecture for the reference GFR system and its alternatives;
- Define the ALLEGRO conceptual design and its safety architecture, in coherence with that of the GFR;
- Develop and validate computational tools needed to analyse performance and operating transients (design basis accidents and beyond).

The goals of the fuel and other 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 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.
- Dense fuels with good thermal conductivity are required to achieve a high power density and a high-temperature. 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 programme.

3.4.3 Main activities and outcomes

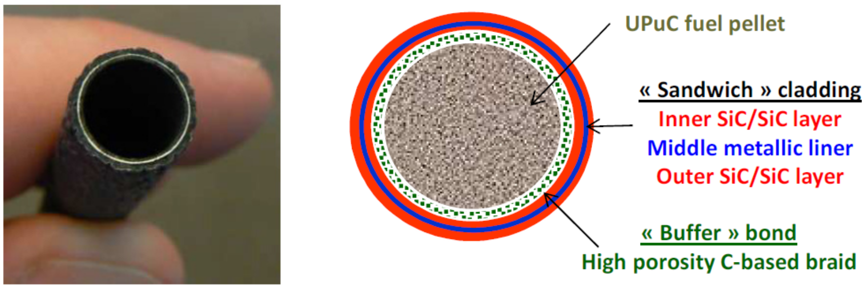
The Euratom FP7 GoFastR project finished in February 2013. Consequently the main activities in Euratom at this time were focused on reporting through completion of the internal project deliverables and those deliverables destined for the GIF. No significant R&D on the CD&S project was undertaken within Euratom during 2013. Some work continued on GFR fuel through cross-cut actions in the FP7 ASGARD project.

GFR core and fuel design

Recent advances on the fuel concept developed by CEA were described by Poette et al. and Zabiego et al. They include (see Figure 3.35):

- A ceramic matrix composite cladding comprising a sandwich of SiC cladding and a thin internal metallic liner to ensure the leak tightness of the pin.
- A "buffer", porous carbon structure placed between the pellet and the cladding allowing higher heat exchanges and moderate clad/pellet mechanical interaction.

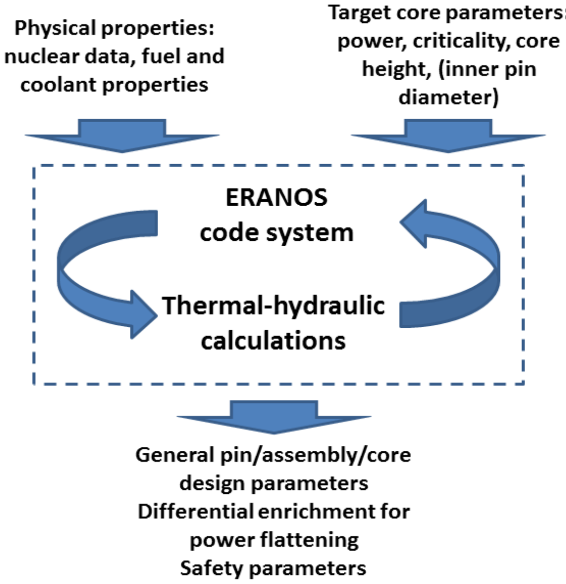
Figure 3.35: Sandwich cladding and buffer bond



GFR neutronic core studies were resumed by CEA starting from optimised cores studied in the frame of Xavier Ingremeaux PhD work using the FARM methodology. These PhD core studies were using a SiC buffer and an internal Tungsten-Rhenium liner stick to the SiC clad. The new studies considered same core geometries, but included a carbon buffer and an internal Tantalum liner (part of the sandwich cladding concept) and were compared to the previous ones. Globally, the impact of the Carbon buffer is weak, whereas the impact of the Tantalum liner is a bit penalising (+ 1.2% on the Plutonium enrichment, and -25% on the Doppler constant), all cores remaining nevertheless acceptable.

The Paul Scherrer Institut (PSI, Switzerland) has developed its own procedure for the design and optimisation of GFR cores. It couples the ERANOS code system for FR analysis with a dedicated script that evaluates the thermal-hydraulic characteristics of a reactor core (see Figure 3.36).

Figure 3.36: Schematic illustration of the procedure developed at the PSI for GFR core design and optimisation



The procedure has been conceived for the design of sub-critical GFRs loaded with annular pins and sphere-pac fuel. However it can be used for critical reactors, as well as for core designs employing standard pins. In addition to the physical properties of the system, the procedures accepts as input a set of target core parameters presently consisting of core power, criticality level, inner pin diameter (for annular pins), and core height.

It gives as a result a core design fulfilling these design objectives and meeting the constraints on maximum fuel and clad temperatures. In case of annular pins, it also equalises the temperature rise inside and outside of the core average pin. The procedure considers the possibility of two-zone cores and adjusts the fuel enrichment in the two zones to achieve an optimal radial power distribution. Finally, it can evaluate safety parameters and fuel cycle characteristic both at beginning-of-life and at equilibrium.

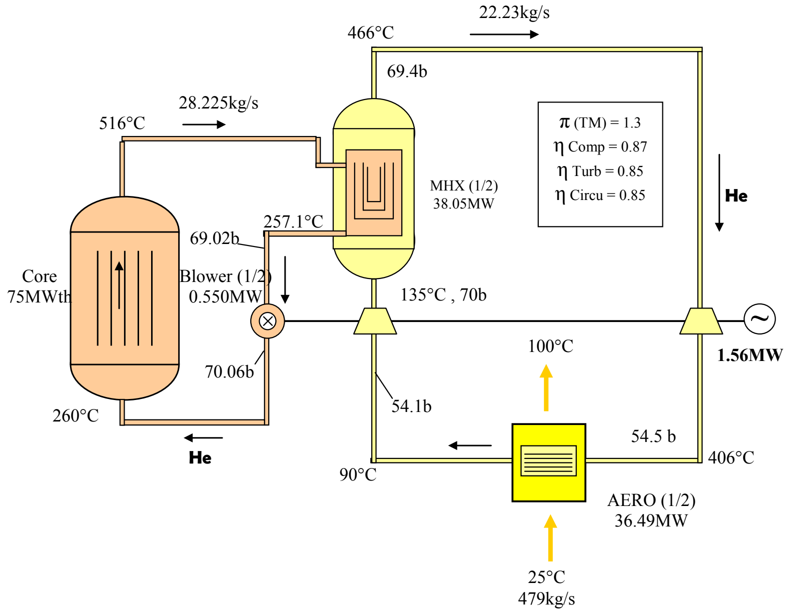
Using this procedure, a preliminary design has been conceived of a 400 MWe subcritical GFR core employing inert-matrix sphere-pac fuel and annular SiC pins. During the design process, the newly developed procedure revealed itself as a powerful tool for parametric analysis aimed at the design and optimisation of GFR cores.

ALLEGRO system design

In the current preliminary viability studies, ALLEGRO primary compression system relies on two centrifugal blowers (operating in parallel on each primary loop) driven by two electrical motors. An alternative design of the primary compression systems, assuming that the two primary blowers are driven by the 2 turbo-generators of the secondary circuits was studied. For this purpose, the secondary circuit is modified; in particular the coolant is no more water but helium (see Figure 3.37). This new system is derived from the alternative cycle already considered for GFR, and named indirect coupled cycle.

Similar advantages as those found for GFR were found: no requirement for external energy for driving the compression system (except for start-up) and safety qualities: suppression of the Loss of Flow Accident due to a primary motor failure, possibility to use this new cycle to improve the grace delay of the reactor using the turbo-machinery to drive the primary blowers during the beginning of accidental situations, and minimisation of water ingress risk.

Figure 3.37: Thermodynamic cycle of the ALLEGRO coupled cycle



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3.5 Lead-cooled fast reactor (LFR)

3.5.1 Main characteristics of the system

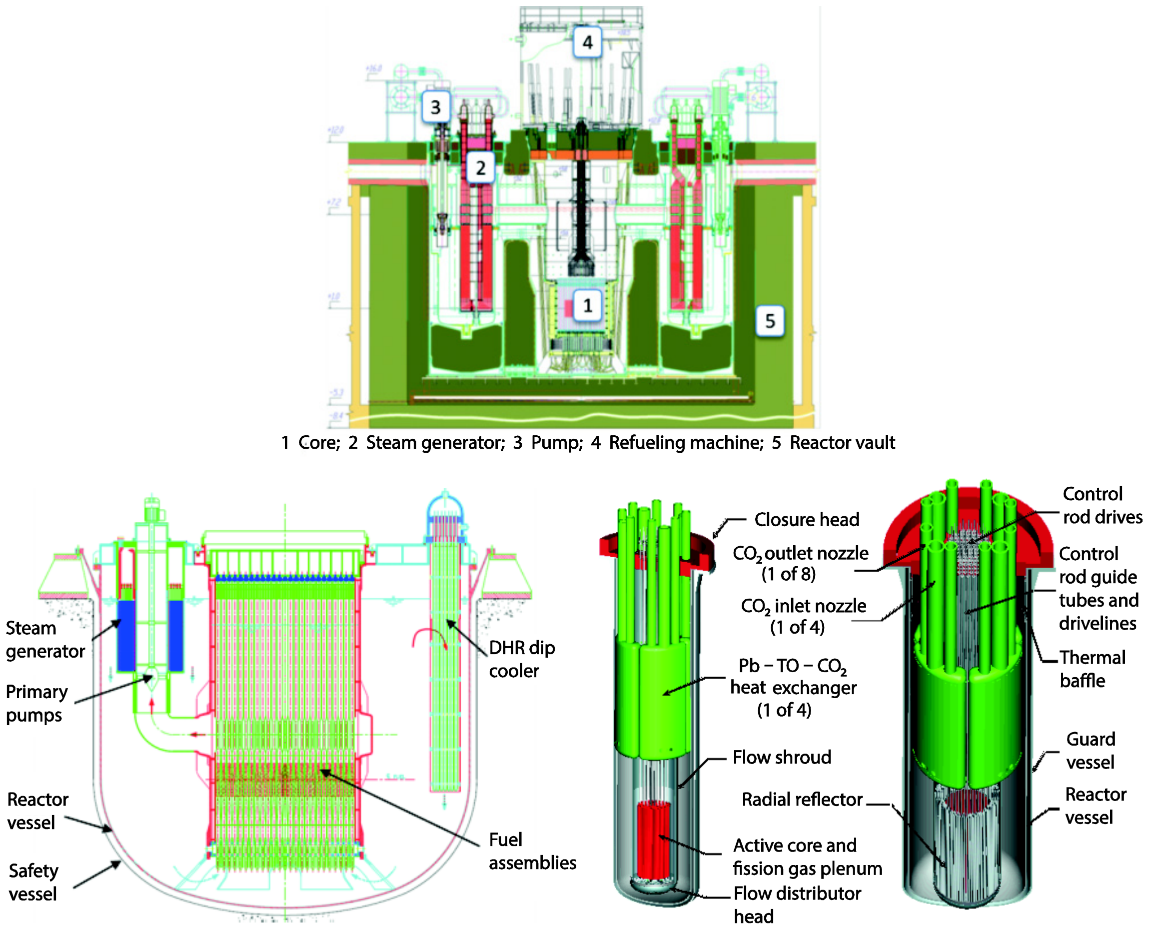
The LFR features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of minor actinides, both self-generated and from reprocessing of spent fuel from light water reactors (LWR), and as a burner/breeder with thorium matrices. An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

The system identified by GIF includes three reference systems. The options considered are a small transportable system of 10-100 MWe size (SSTAR-US) that features a very long core life, a system of intermediate size (BREST 300 Russia), and a larger system rated at about 600 MWe (ELFR EU), intended for central station power generation. In the United States, some activity is carried out for the conceptual design of the small LFR. The expected secondary cycle efficiency of the LFR system is above 42%. It can be noted that the reference concepts for GIF-LFR systems covers the whole full range of powers, from the small to the intermediate and large size. Important synergies exist among the different systems so that a coordination of the efforts carried out by participating countries has been one of the key points of LFR development.

The conceptual configuration of ELFR has been under development since April 2010 within the 7th Euratom framework programme. The LEADER project has just completed its activities on 30 September 2013. The main goal of the project was to reach an LFR configuration using as much as possible proven solutions and limit the technology development needed for deployment at the industrial level. The LEADER project has been performed by a consortium consisting of sixteen organisations from Europe. ELFR features hexagonal wrapped fuel assemblies and an extensive use of passive technology for decay heat removal. The system size (600 MWe) has been chosen to demonstrate the possibility of designing a competitive and safe fast critical reactor using simple engineered technical features while fully complying with the mission identified in the GIF Roadmap of minor actinides recycling in a closed fuel cycle with particular attention devoted to safety aspects from the beginning of the design activities.

A second, but not less important, goal of the LEADER project has been the development of a scale-downed LFR demonstrator called ALFRED which has been already included in the ESNII Road Map.

Figure 3.38: The three reference systems of GIF-LFR – BREST (top), ELFR (left), SSTAR (right)



BREST-OD-300 is a reactor facility of pool-type design, which incorporates within the pool the reactor core with reflectors and control rods; the lead coolant circulation circuit with steam generators and pumps; equipment for fuel reloading and management; and safety and auxiliary systems. The reactor equipment is arranged in a steel-lined, thermally insulated concrete vault. BREST has a widely-spaced fuel lattice with a large coolant flow area, resulting in low pressure losses, favouring the establishment of primary natural circulation for decay heat removal. It shares with other designs the absence of uranium blankets, replaced by lead reflector with the proper albedo improving power distribution, providing a negative void and density coefficients, and ruling out the production of weapons-grade plutonium. The BREST decay heat removal systems are characterised by passive and time-unlimited residual heat removal directly from the lead circuit by natural circulation of air through air-cooled heat exchangers, with the heated air vented to the atmosphere. The fuel type considered for the first core of the BREST fast reactor is nitride of depleted uranium mixed with plutonium and Minor Actinides (MA), whose composition corresponds to that of irradiated (spent) fuel from PWR's following reprocessing and subsequent cooling for about 20 years. The characteristics of lead allow for the operation with such fuel at an equilibrium composition. This mode of operation is characterised by full reproduction of fissile nuclides in the core (core breeding ratio (CBR) is about 1) with irradiated fuel reprocessing in the closed fuel cycle. Reprocessing is limited to the removal of fission products without separating Pu and minor actinides (MA) from the mixture (U-Pu-MA). One of the notable characteristics of the BREST plant is that a reprocessing plant is co-located with the reactor, eliminating in principle any accident or problem due to fuel transportation.

The reference design for the SSTAR is a 20 MWe natural circulation reactor concept with a small transportable reactor vessel. 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 utilising a supercritical carbon dioxide Brayton cycle power converter.

The typical design parameters of the GIF-LFR systems are briefly summarised in Table 3.1.

Table 3.1: Key design parameters of GIF LFR concepts

Parameters	ELFR	BREST	SSTAR
Core power (MWt)	1 500	700	45
Electrical power (MWe)	600	300	20
Primary system type	Pool	Pool	Pool
Core inlet T (°C)	400	420	420
Core outlet T (°C)	480	540	567
Secondary cycle	Superheated steam	Superheated steam	CO_2
Net efficiency (%)	42	42	44
Turbine inlet pressure (bar)	180	180	20
Feed temperature (ºC)	335	340	402
Turbine inlet T (ºC)	450	505	553

3.5.2 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.

Following the reformulation of GIF-LFR-pSSC in 2012 the SRP was completely revised and has been issued at the end of 2013.

The approach taken in the SRP is to consider the research priorities of each member entity, and to propose a co-ordinated research programme to achieve the objectives of each member while avoiding unnecessary duplication of effort. The integrated plan recognises three representative reference systems to address the principal technology objectives of the members:

- a small, transportable system with very long core life, and
- a system of intermediate size;
- a system for central station power generation.

The committee notes that there are significant potential commonalities in research and design among these three system thrusts. The plan proposes co-ordinated research along parallel paths leading to a single pilot facility that can serve the research and demonstration needs of the reference concepts while reducing the unnecessary expense of duplicate major facilities and research efforts.

The needed research activities are identified and described in the SRP. It is expected that co-ordinated efforts can be organised in four major areas and formalised as projects once an SA agreement will be signed: system integration and assessment; lead technology and materials; system and component design and fuel development. The goals and activities of these four R&D projects are summarised below.

System integration and assessment (SIA) project

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

System and component design project

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

Fuel development project

The LFR fuel development project is a continuing long term process consisting of tasks designed to meet progressively more ambitious requirements. It includes efforts in the areas of core materials development, fuel fabrication, fuel irradiation and tests aimed at fuel qualification. It is also important to note that strong synergies exist with parallel SFR fuel development.

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

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

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

Lead technology and materials project

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

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

3.5.3 Main activities and outcomes

Following the signature of an MOU between Euratom and Japan (2010) and the signature by the Russian Federation of the Memorandum (2011), the GIF-LFR-pSSC was completely reformulated in 2012. Being the collaboration based on the MOU signatures and observers contributions the LFR-pSSC activities are essentially based on a fruitful exchange of information between partners taking place at the meetings that have been scheduled each 6 month in order to follow closely the research developments of each country. To give a complete picture of the activities carried out it is convenient to follow their evolution on the basis of the scheduled meetings.

The first meeting of the pSSC was held in Pisa (Italy) on April 2012. This was the first meeting of the reformulated pSSC and the main decision was to start a complete revision of the LFR SRP and to accept USA as observer of the activities.

The second meeting took place in Japan on November 2012 hosted by the Tokyo Institute of Technology. A main decision of the meeting was to expand the initiative to three thrusts (large, medium and small LFRs): ELFR, BREST, SSTAR and to continue accordingly the SRP revision. A paper was also written and presented at the GIF Symposium is San Diego.

The third meeting was kindly hosted by OECD/NEA in Paris in March 2013, just after the FR13 conference organised by IAEA. The meeting had a large participation from MOU countries as well as additional observers. China and the Republic of Korea were invited and, on their request, accepted as observers. Additionally, on the basis of a RSWG request, it was decided to start the work on a White Paper on Safety as well as on Safety Design Criteria for LFR.

The LFR pSSC met again at OECD/NEA in October 2013 and the strategy for the development of the White Paper and SDC were decided. Concerning the White Paper the participants observed that the content of the white paper as proposed by RSWG is detailed and specific information are needed for application of ISAM. The pSSC noted that for none of the reference systems of GIF-LFR activities it is possible to provide the requested amount of information. On the other hand, in the frame of FP7 LEADER project, the ISAM methodology was at least partially applied to ALFRED design. The pSSC decided to use the material available from LEADER project and use ALFRED as an example of LFR design in the white paper in strict collaboration with RSWG. At the same meeting it was decided to develop LFR SDC on the basis of the SDC report already developed for SFR. Taking into account the similarities between the two systems it was in fact realized the SFR SDC as a very useful starting point in terms of a template and guide for the development of LFR-SDC. The main idea behind is to delete not applicable parts and to add specific topics for the LFR. The goal is to have at least a common basis for Gen IV LM cooled SDC report. A preliminary draft was generated and distributed to members and observers during October meeting.

During 2013 the revision of the LFR SRP has been completed and the final report issued. Also a paper dedicated to GIF LFR activity was submitted as requested by the Technical Director to a special issue of *Progress in Nuclear Energy*. Finally the LFR-pSSC was invited to present at both the EG and PG meetings held in Brussels on November 2013.

Main activities in the Russian Federation

In the frame of the development of the LFR, the Russian Federation plays a key role. *The Technology Roadmap Update for Generation IV Nuclear Energy Systems* to be issued in the beginning of 2014 reports a transition date from "performance" to "demonstration" in 2021. This date is based on the announced date by ROSATOM for an expected start of operation of the BREST reactor. The operation of SVBR-100 is also expected to be as early as 2018, for the LBE 100 MWe reactor.

The activities carried out in the Russian Federation in 2013 are widely spread over many technological aspects of the LFR technology and are here below very briefly summarised:

Safety: In order to identify initiating events challenging the safety functions of the system, PSA study and fault tree analysis carried out to determine most challenging initiators and sequence of events have been performed. Deterministic analysis are carried out for the above identified accident initiators.

Thermal-hydraulics: For the purpose of detailed experiments for heat transfer coefficients of SGs, experimental facility for SGTR simulation is under construction. Design of Experiment methodology is widely used to help design activities. The experimental program includes both the evaluation of leakages from SG tubes, tube rupture as well as tube breaks caused by tube defects subjected to high pressure. Detailed experiments within the Reynolds numbers of interest are under execution to determine precisely the heat transfer coefficients and mixing properties of the sub-channels in a fuel element. Numerical verification of Relap5 models of the SGs have been carried in transient conditions. The model is being used to compute additional transients of interest for the design, performance, safety analysis.

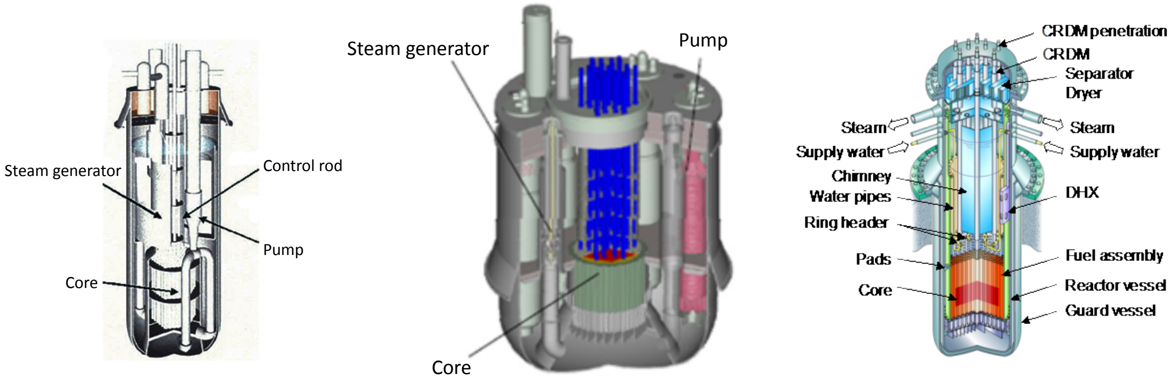
Coolant: The scheme of the system to receive, prepare, and transfer the coolant as well as to provide quality control of the coolant have been developed. The combined scheme includes the basic equipment of system: melting ovens, where carry out non-stop melting of lead ingots; accumulative tanks, where fulfil deposit of liquid lead; filter for finishing polishing of coolant, which fuse in circulation loop; pumping equipment. The scheme also includes lay-out of basic equipment and equipment to control coolant quality (filter, mass exchange apparatus, disperser, sensors of control of thermo dynamical activity of oxygen). Design of external loop systems has been carried out for: cover gas and lead with provision for purification of impurities (including Polonium) and fission products.

Materials: Material irradiation has been carried out during two micro campaign of reactor BOR-60. The main goal of the investigation is to study the mechanical characteristics of steel samples. The irradiation of samples has been fulfilled (small Gagarin's samples, samples for test on fracture strength with incision, has been sharpened by fatigue crack, basic metal and metal of joint weld of steel of pearlite class). On samples of basic metal damage doses (0,63-0,69) dpa have been obtained, on samples of metal of joint weld (0,87-0,96 dpa). Temperature during period of reactor tests was in the range (436-464°C). Reactor tests of models for absorbers on the basis of tablets of boron carbide and dysprosium hafnate are on-going. The maximum integral value obtained in steel was 24 dpa.

Main activities in Japan

In early 1990s, LBE-cooled long-life Safe Simple Small Portable Proliferation Resistant Reactor (LSPR, 50 MWe) shown in Figure 3.39 (a) was proposed by Tokyo Institute of Technology. The innovative features of LSPR are long life core, small reactors constructed in factories and transported to the site and deployed, sealed reactor vessel without being opened at the site and excellent proliferation resistance because of long refuelling interval. On the other hand, in the feasibility study of FR cycle various types of LFR were studied by Japan Nuclear Cycle Development Institute (JNC) and Japan Atomic Power Company (JAPC) and finally the forced convection cooling type of LFR (750 MWe) shown in Figure 3.39 (b) was selected in 2006. At the same time, the concepts of a small and simplified LFR with long life core, the Pb-Bi-Water Direct Contact Boiling Water Fast Reactor (PBWFR, 150 MWe) shown in Figure 3.39 (c) was formulated by Tokyo Institute of Technology from 2002 till 2004. In the early 2000s, the concept of CANDLE burning was proposed, and has been applied to the LFR core. The design concepts of LFRs and related studies in Tokyo Tech. were presented at GLOBAL 2011, Osaka, December 2011.

Figure 3.39: Concepts of LFRs studied in Japan



(a) Tokyo Institute of Technology LSPR (50MWe)

(b) JNC/JAPC LFR (750MWe)

(c) Tokyo Institute of Technology PBWFR (150MWe)

In these years, mainly fundamental studies for the LFR concepts mentioned above were continued primarily in Tokyo Institute of Technology. The studies covered the conceptual study of CANDLE burning, and material and thermal-hydraulic studies. The overview of recent studies related to LFRs in Tokyo Institute of Technology was presented at International Conference on Advances in Nuclear Science and Engineering (ICANSE 2011), Denpasar, Indonesia, November 2011.

In order to inform various people in Japan of LFR activities for GIF, they were presented at the special session on the status of Generation IV International Forum at the fall meeting of Atomic Energy Society of Japan (AESJ), Kitakyushu, Japan, September 2011. The activities in CRINES for GIF LFR, were also presented at The 4th International Symposium on Innovative Nuclear Energy Systems (INES-4), Tokyo Institute of Technology, Tokyo, Japan, November 2012.

For the support to the activities of CRINES for GIF LFR, a CRINES GIF-LFR special committee has been setup since March 2012, and the committee meetings have been held in March, June, and October 2012.

Main activities in the Euratom

The Euratom activities have been focused on the development of the ELFR and ALFRED designs as part of the FP7 LEADER project. Sixteen European organisations carried out the conceptual design of the ELFR (substantially an evolution of the previous ELSY design) and of a small scale demonstrator called ALFRED (Advanced Lead Fast Reactor European Demonstrator). Several experimental facilities have been put in operation by LEADER partners for material testing, lead corrosion and fretting in lead, as well as thermal-hydraulic and steam generator tube rupture experiments. The project was terminated in September 2013 but, following the request by the Romanian Government to host ALFRED in the Mioveni site, in September 2013 a new FP7 project called ARCADIA was started. The ARCADIA project, coordinated by ICN (the Romanian Institute of Nuclear Research), collected contributions from 26 partners, including many East European country organisations, with the main goal to pave the way for the implementation of ALFRED in Romania.

In parallel to the above activity the FP7 CDT project was completed in 2013. The CDT project was dedicated to the conceptual design of the MYRRHA facility, a lead-bismuth cooled $100MW_{th}$ with both critical and subcritical modes of operation. Following the end of CDT the EC approved the FP7 MAXSIMA project dedicated to safety analysis, fuel aspects and innovative passive technology system for lead, lead bismuth cooled facilities and reactors. In October 2013 the FEED contract from SCK●CEN for MYRRHA was awarded to a consortium formed by AREVA, ANSALDO, EMPRESARIOS AGRUPADOS and GROMTJ. The activities of the consortium are centred on the design of the ancillary systems of the facility primary side including: secondary side, DHR systems, cover gas, lead- bismuth purification, pressure suppression system.

Following a memorandum of understanding signed in February 2012, ANSALDO, ENEA and INR signed a consortium agreement for ALFRED development and implementation in Romania. The consortium is presently based on in-kind contributions of the three organisations and will be opened to the participation of other European Industrial/Research/Educational organisations at the beginning of the next year. The aim of the consortium is to coordinate all efforts of the different partners in one direction, to share the knowledge generated by the partners and prevent any activity duplication.

Last but not least, SNE-TP issued its Strategic Research Agenda in 2013 which identify four projects of interest of ESNII: ASTRID, ALLLEGRO, MYRRHA and ALFRED. Following this line of activities the ESNII+ project started in November 2013, and its activities mainly dedicated to cross cutting aspects between the ESNII projects.

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3.6 Molten salt reactor (MSR)

3.6.1 Main characteristics of the system

Molten salt reactors (MSRs) have two main subclasses. In the first subclass, which corresponds to the R&D path developed under the MOU, fissile material is dissolved in the molten fluoride salt and it serves both as fuel and coolant in the primary circuit. In the second subclass, the molten fluoride salt serves as the coolant to a coated particle fuelled core similar to that employed in very high-temperature Reactors (VHTRs). In order to distinguish the reactor types, the solid fuel variant is typically referred to as a fluoride salt-cooled high-temperature reactor (FHR).

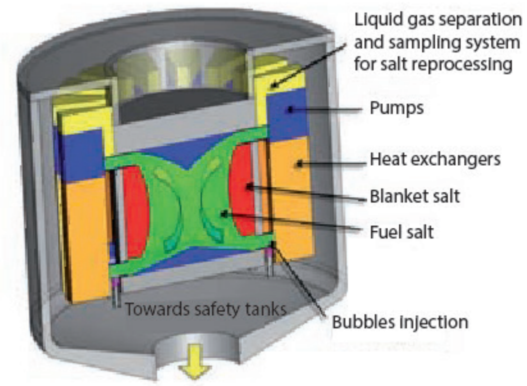
The molten salt fast reactor concept

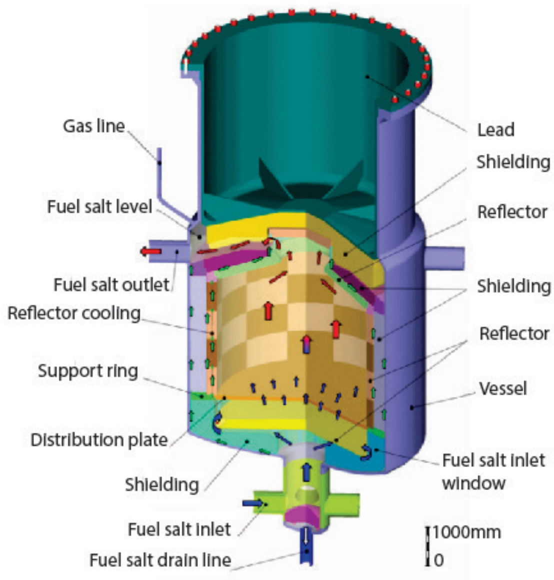
In the beginning MSRs were mainly considered as thermal-neutron-spectrum graphite-moderated reactors. Since 2005 liquid fuelled MSR R&D has focused on fast-spectrum MSR options combining the generic assets of fast neutron reactors (extended resource utilisation, waste minimisation) with those related to molten salt fluorides as both fluid fuel and coolant (low pressure and high boiling temperature, optical transparency).

Recent MSR developments in Russian Federation on the 1 000 MWe molten-salt actinide recycler and transmuter (MOSART) and in France on the 1 400 MWe non-moderated thorium molten-salt reactor (MSFR) address the concept of large power units with a fast neutron spectrum in the core (see Figure 3.40). The fast neutron spectrum molten salt reactors open promising possibilities to exploit the ^{232}Th-^{233}U cycle and can also contribute, in the transmuter mode, to significantly diminishing the radiotoxic inventory from current-reactor used fuel in particular by lowering the masses of transuranic elements (TRU).

Fast MSRs have large negative reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors. Compared with solid-fuelled reactors, MSFR systems have lower fissile inventories, no radiation damage constraints on attainable fuel burnup, no used nuclear fuel, no requirement to fabricate and handle solid fuel, and a homogeneous isotopic composition of fuel in the reactor.

Figure 3.40: MSFR and MOSART concepts



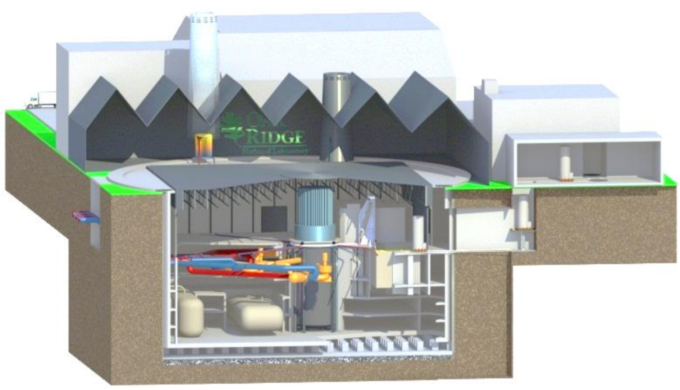


Fluoride salt-cooled high-temperature reactor (FHR)

Fluoride salt-cooled high-temperature reactors (FHRs) that are currently outside the scope of the MOU are a nearer-term molten salt reactor option. FHRs by definition feature low-pressure liquid fluoride salt cooling, coated particle fuel, a high-temperature power cycle, and fully passive decay heat rejection. FHRs have the potential to economically and reliably produce large quantities of electricity and high-temperature process heat while maintaining full passive safety. Leveraging the inherent reactor class characteristics avoids the need for expensive, redundant safety structures and systems and is central to making the economic case for FHRs. Moreover, their high-temperature increases FHR compatibility with low- or no-water cooling. FHRs will have a near thermal neutron spectrum, and first-generation FHRs are intended to operate on a once-through low-enrichment uranium fuel cycle.

The most mature FHR design concept currently available is for the advanced high-temperature reactor (AHTR). The AHTR is a design concept proposed in the US for a first-generation, large power output [3 400 MW(th)], central station type FHR (Figure 3.41). FHRs are a broad reactor class that maintains strong passive safety at almost any scale and features significant evolutionary potential for higher thermal efficiency (through higher temperatures), process heat applications, online refuelling, thorium use, and alternative power cycles.

Figure 3.41: AHTR reactor building layout



3.6.2 R&D objectives

Partners of the MSR PSSC are involved in the Euratom-funded Evaluation and Viability of Liquid Fuel Fast Reactor Systems (EVOL) project. A complementary ROSATOM project called MARS (Minor Actinides Recycling in Molten Salt) between Russian research organisations is being carried out in parallel. The common objective of these projects is to propose a conceptual design of MSFR as the best system configuration – resulting from physical, chemical and material studies – for the reactor core, the reprocessing unit and wastes conditioning. The mastering of MSR technically challenging technology will require concerted, long-term international R&D efforts, namely:

- studying the salt chemical and thermodynamic properties;
- system design: Development of advanced neutronic and thermal hydraulic coupling models;
- development of a safety approach dedicated to liquid fuelled reactors;
- studying materials compatibility with molten salt;
- development of efficient techniques of gaseous fission products extraction from the coolant;
- salt reprocessing: reductive extraction tests (actinide-lanthanide separation) and He bubbling (gaseous fission products).

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

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

3.6.3 Main activities and outcomes

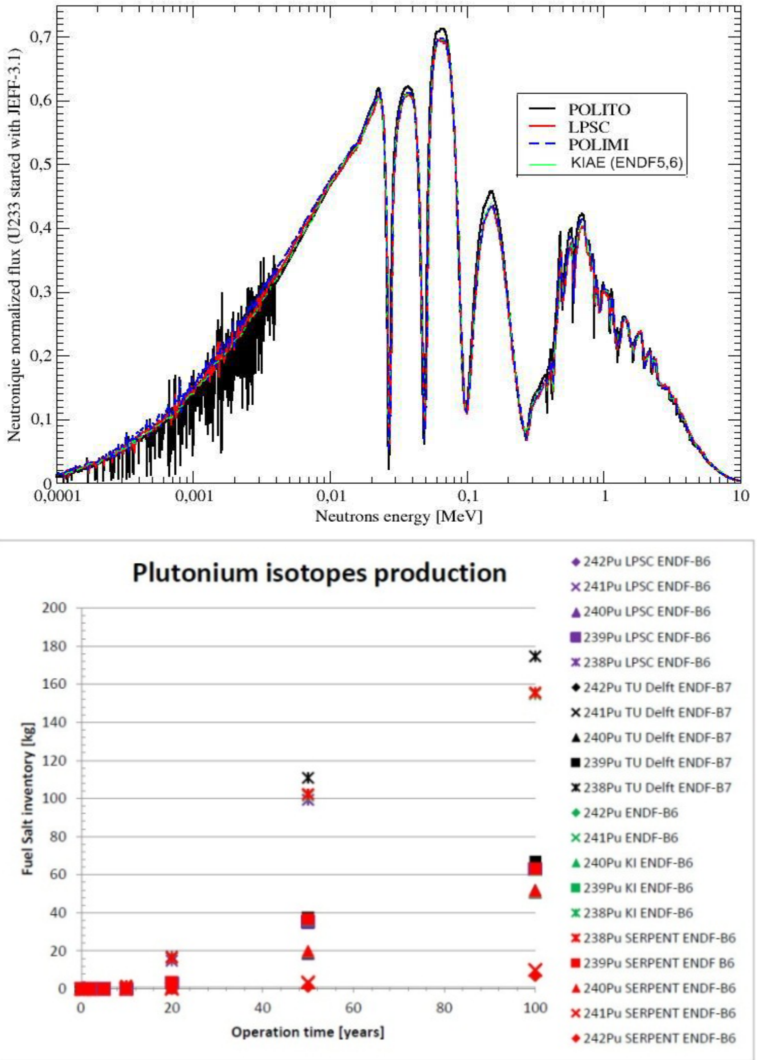
Between 2011 and 2013, the Euratom EVOL project of FP7, associated to the Rosatom minor actinides recycling in molten salt (MARS) project, were the main frame of international scientific co-operation on Th-U MSFR and MOSART concepts. Some results obtained within this collaborative work are given hereafter.

Studies around the concept of MSFR were focused on neutronic and safety issues. A neutronic benchmark has been realised to validate tools and methods developed for the optimisation of the MSFR physical specifications. Its residual heat removal has also been calculated and analysed in different incidental situations. Due to the differences with classical solid fuelled nuclear reactors, a novel methodology for the safety assessment for liquid fuelled reactors is under development. Finally a multi-physics approach to simulate such a reactor has been initiated, based in a first phase on a neutronic – thermal-hydraulics coupling.

Neutronic benchmark

Both static and transient calculations of criticality, neutron spectrum, delayed neutron fraction, feedback coefficients, heavy nuclei inventories (see Figure 3.42) were carried out by the partners of both EVOL-and MARS projects. A MSFR core geometry was first agreed to allow a neutronic benchmark using various simulation codes (REM or ORIGEN coupled to MCNP, Dalton, Serpent, Eranos, Helios). ENDF-B6/7 and JEFF-3.1 data bases were used to run the calculations by the following institutions: CNRS-IN2P3-LPSC Grenoble, France; Delft University of Technology, the Netherlands; Politecnico di Torino and Politecnico di Milano, Italy; HZDR, Dresden, Germany; Kurchatov Institute, Moscow, Russia.

Figure 3.42: Example of results of the neutronic benchmark of the MSFR in the frame of the EVOL-MARS collaborative projects: neutron spectrum of the MSFR (top) and Pu inventory evolution (down)



Safety study

A novel methodology for the safety assessment of the liquid fuelled reactors is under development. This work has been initiated in a recent PhD study (M. Brovchenko, Grenoble Institute of Technology, October 2013), through the transposition of classical safety methods and a systematic risk analysis. The integrated safety analysis methodology (ISAM) developed by the GIF RSWG and a more general systemic risk analysis have been the basis of this PhD work. Both were used in parallel at this early stage of system design as screening means for any accidental situations. Some dynamic situations were also simulated in Grenoble and Karlsruhe by coupling neutronics and thermal hydraulics. A classification of the accidents specific to the MSFR has been defined and some accidental transients will be evaluated in the frame of the EVOL project.

Multiphysics modelling

The MSFR core design analyses require an evaluation of the fuel salt flow distribution in the reactor cavity since it influences the distribution of the delayed neutron precursors (and thus the reactor reactivity), the fuel irradiation, the reactivity feedback coefficients, the reactor core wall temperature, etc. The prediction of the fuel salt flow distribution involves the use of coupled neutronics and thermo-hydraulics numerical simulations. In a first step, three partners of the EVOL project (INOPRO LAO, France; KIT, Germany, and LPSC-IN2P3-CNRS, France) have participated to a thermal-hydraulic benchmark of the MSFR (see Figure 3.43). A multi-physics approach is necessary to simulate the MSFR behaviour: a neutronics – thermal hydraulics coupling is currently under development.

Thermodynamic data

At JRC-ITU the ongoing activities on determination of physico-chemical properties of molten salt reactor fuel have continued and with the new experimental data obtained using the calorimetric facilities the full thermodynamic assessment of the LiF-ThF_4-PuF_3 system has been made. The calculated liquidus projection of the LiF-ThF_4-PuF_3 phase diagram is shown in Figure 3.44 indicating the lowest melting temperature at 818 K and the LiF-ThF_4-PuF_3 (74.9-22.3-2.8 at %) composition. The melting behaviour of the phase diagram in the region of the ternary eutectic was checked by series of experiments performed by differential scanning calorimetry on CeF_3-containing samples (CeF_3 as analogue compound to PuF_3) confirming the lowest melting temperature as indicated above, with uncertainty of ±10 K.

LiF-ThF_4-PuF_3 is a key system for the fuel of the MSFR concept and the established database was used to optimise the fuel with respect to its properties with emphasis on the melting behaviour. The study was done within the frame of the FP7 EU project EVOL and was coupled with neutronic calculations performed in CNRS to guarantee that the selected composition would sustain the chain reaction while having enough breeding gain. As a conclusion of this study two options of fuel composition have been determined: the LiF-ThF_4-UF_4-PuF_3 (78.6-12.9-3.5-5 at%) and the LiF-ThF_4-UF_4-$(TRU)F_3$ (77.5-6.6-12.3-3.6) compositions. They both are characterised by sufficient concentration of fissile PuF_3 to keep positive breeding ratio of the reactor with fairly low melting temperatures of 873 K and 857 K respectively. The enrichment of $^{235}UF_4$ is in the former case calculated to 20% and in the latter one for 13%, thus low enough to fulfil the non-proliferation criteria. The drawback of both compositions can be in potentially increased melting temperature of the fuel solvent during chemical reprocessing (>1 000 K). It must be however noted that the demand on low temperature is much higher in case of the reactor compared to the reprocessing vessel, as other structural materials (e.g. graphite or ceramics for coatings etc.) or e.g. cold crucible technology may be used during reprocessing. Furthermore, the exact reprocessing scheme is not yet exactly defined and it may be possible that the clean-up process may not require such dramatic composition shift of the fuel.

Figure 3.43: Example of results of the thermal-hydraulic MSFR benchmark of the EVOL project: Velocities (m/s) in a vertical plane with the full exchanger (left) with simplified exchanger for the benchmark (middle from INOPRO IAO and right from LPSC Grenoble)

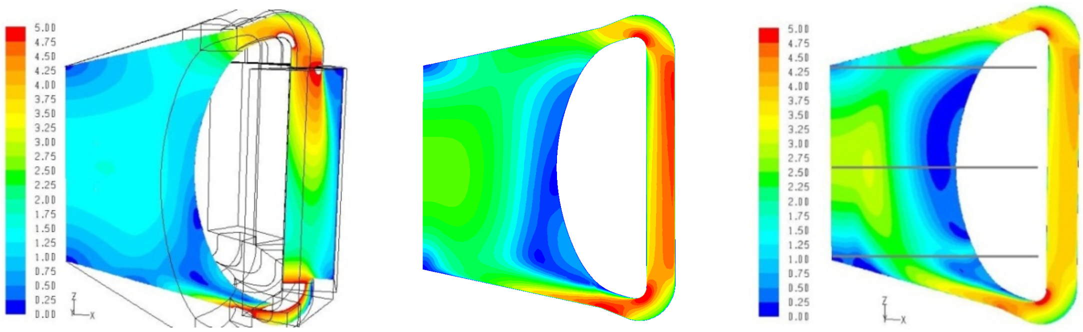
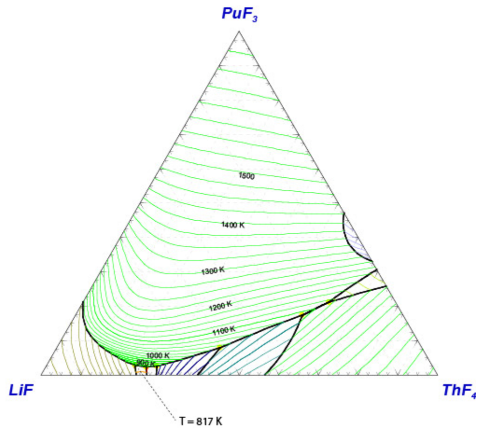


Figure 3.44: A calculated liquidus projection of the LiF-ThF4-PuF3 system indicating the lowest melting temperature at 817 K and LiF-ThF4-PuF3 (74.9-22.3-2.8) composition



MOSART fuel cycles

The MOSART system without essential design changes is capable of operating in different modes:

- as transmuter for a wide range of possible feedings;
- as TRU–^{233}U converter;
- as self-sustainable system with CR = 1 at transition to U–Th fuel cycle;
- as breeder with conversion ratio CR > 1.

Interesting possibilities may be demonstrated by the use of MOSART with the reduced dimensions as self-sustainable system with CR=1. For transition to this mode of operation a strategy of gradual increase of thorium concentration in the fuel salt is required. Single fluid 2.4 GWt Li,Be/F MOSART core (radius -1.4 m, height – 2.8 m) containing as initial loading 2 mole % of ThF_4 and 1.2 mole % of $TRUF_3$ with the rare earth removal time 1 EFPY after 12 years can operate without any $TRUF_3$ make up basing only on Th support as a self-sustainable system (see Figure 3.45). The maximum concentration of $TRUF_3$ during this transition does not exceed 1.7 mole %. At equilibrium molar fraction of ThF_4 in the fuel salt is near 6% and it is enough to provide the system with CR=1 up to 50 years of the reactor operation. The reactivity temperature coefficient of the homogeneous core is not only essentially negative, but also practically has no inertion. In the case of self-sustainable MOSART its value on equilibrium is – 6.7 pcm/K. The use of the Th – containing blanket permits to reduce the transition to self-sustainable mode of operation down to 3-4 EFPY, but of course makes the system more complicated from technical point of view. Any moment of self-sustainable mode of operation can be used for transition to breeder mode with CR> 1 due to increasing of thorium concentration in the fuel salt. So self-sustainable mode demonstrates the MOSART abilities as transforming system and can be used for starting U-Th fuel cycle on the base of first TRU loading from LWR SNF.

TRU trifluorides solubility

Actinides and lanthanides fluorides solubility in molten salts were measured by local γ-spectrometry, isothermal saturation and reflectance spectroscopy within MARS project. Data on PuF_3 solubility obtained in this study are in excellent agreement with those interpolated from results of ORNL and BARC, who determined PuF_3 solubility in same temperature range respectively in LiF-BeF_2 and NaF-LiF-BeF_2 (ORNL) as well as LiF-ThF_4, and LiF-BeF_2-ThF_4 (BARC). The data on solubility in molten salt fluorides appear to follow a linear relationship within the experimental accuracy of the measurements when plotted as logarithm of molar concentration of actinide trifluoride vs. $1/T$(K). Equations are given in Table 3.2 for molten LiF-BeF_2, LiF-ThF_4, LiF-NaF-KF, NaF-LiF-BeF_2 and LiF-BeF_2-ThF_4 salt mixtures. Particularly, it was found that two beryllium fluoride containing solutions LiF-BeF_2 and NaF-LiF-BeF_2 with BeF_2 concentration 27 mole % provide close values for solubility of PuF_3 in the temperature range of 825-1 000K. Solubilities of some other actinide fluorides were also measured, including AmF_3 in the molten

$LiF-BeF_2$ and LiF-NaF-KF salt mixtures (see Table 3.3). For two beryllium fluoride containing solutions ranging in BeF_2 concentration from 27 to 34 mole %, the ^{241}Am analysis showed that behaviour of americium is almost identical to that of plutonium. In case of LiF-NaF-KF eutectic, tests showed that the behaviour of americium is more close to that of thorium and uranium.

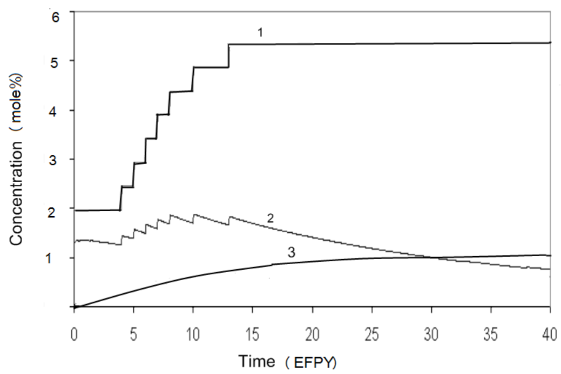
Table 3.2: Solubility of PuF_3 in molten salt fluoride mixtures: log S, mole% = A + B/T,K

LiF	NaF	KF	BeF_2	ThF_4	*T*, K	*A*	-*B*·10^{-3}	Method
46.5	11.5	42	0	0	823-973	5.59	3.949	isothermal saturation
73	0	0	27	0	825-1 000	3.927	3.099	isothermal saturation
66	0	0	34	0	800-900	3.231	3.096	γ-spectrometry
15	58	0	27	0	825-925	3.639	2.750	γ-spectrometry
17	58	0	25	0	800-900	3.253	2.578	γ-spectrometry
78	0	0	0	22	873-973	2.58	1.73	γ-spectrometry
75	0	0	5	20	873-1 023	2.06	1.34	γ-spectrometry
77	0	0	17	6	848-998	3.61	2.91	γ-spectrometry

Table 3.3: Solubility of AmF_3 in molten salt fluoride mixtures: log S, mole% = A + B/T,K

LiF	NaF	KF	BeF2	T, K	A	-B.10-3	Method
46.5	11.5	42	0	825-975	3.75	2.052	isothermal saturation
73	0	0	27	825-1 000	3.93	3.101	isothermal saturation

Figure 3.45: Transition to equilibrium of ThF_4 (1), $TRUF_3$ (2), UF_4 (3) in single fluid 2.4GWt MOSART (self-sustainable mode) with Li,Be,Th/F core (CR=1 gradual increase of thorium)



Tellurium corrosion

MARS studies with molten $75LiF-5BeF_2-20ThF_4$ salt mixture (mole %) fuelled with 2 mole% of UF_4 and containing additives of Cr_3Te_4, include five 250 hrs tests with exposure of Ni-based alloys specimens at temperatures from 993K to 1 023K and under mechanical loading from 0 to 50 MPa. Special Ni-based alloys selected for corrosion studies with Li, Be, Th, U/F fuel salts had the following mass compositions (in wt%): HN80M-VI (Mo–12, Cr–7.6, Nb–1.5), HN80MTY (Mo–13, Cr-6.8, Al–1.1, Ti–0.9), HN80MTW (Mo-9.4, Cr-7.0, Ti-1.7, W-5.5) and EM-721 (Cr-5.7, Ti-0.17, W-25.2). Materials compatibility tests confirmed that the HN80MTY alloy has the best corrosion and mechanical properties. It does not undergo tellurium intergranular corrosion (IGC) in the fuel salt with addition of about 2 mole % UF_4 mixture at [U(IV)]/[U(III)] ≤100 (see Figure 3.46). HN80MTY alloy can be recommended for further consideration as the main container material for the fuel circuit with operating temperature up to 1 023K required for both MSFR and MOSART design.

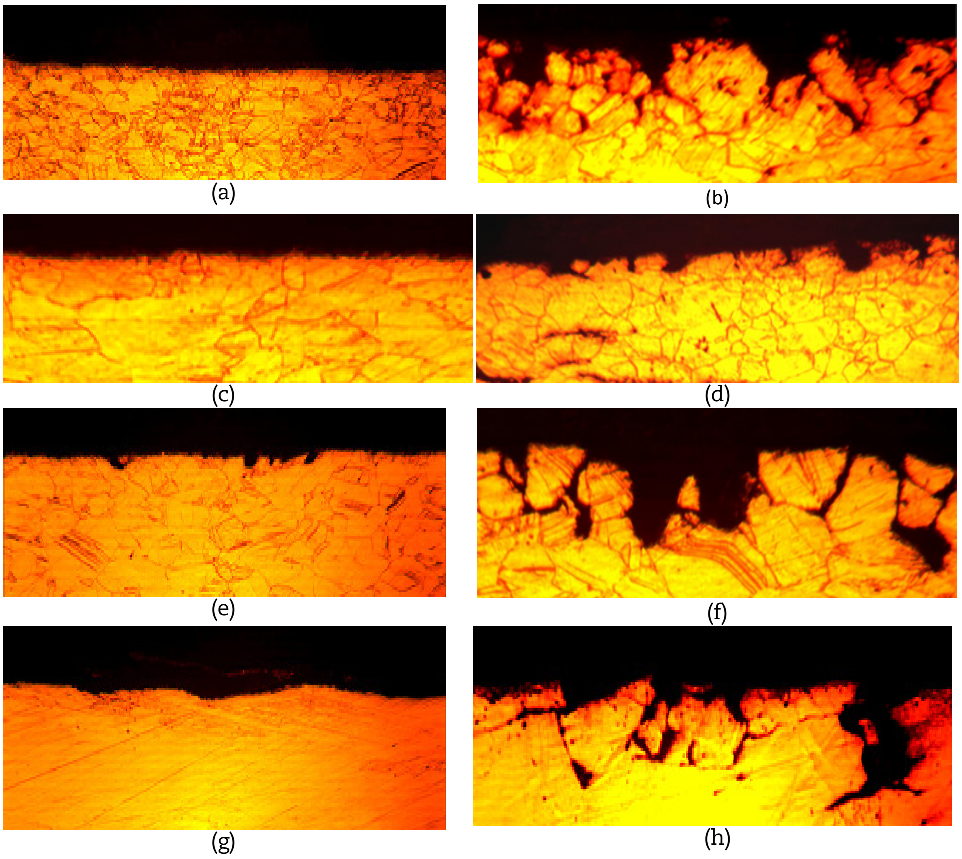
FHR studies

As a nearer term nuclear power plant (NPP) option, FHR development has more of an engineering focus than is the case for liquid fuelled MSRs. Developing FHRs into an economically preferable reactor class, however, requires overcoming a number of technical challenges. FHRs rely primarily on advancing and combining established technologies and no technical hurdles have been identified to-date that would prevent FHRs from being developed into economically advantageous nuclear power plants. China and the United States, observers in the MSR PSSC, are currently working on FHR concepts, as described below.

The Chinese Academy of Science through its Shanghai Institute of Applied Physics (SINAP) is actively pursuing FHR development. The SINAP project includes the full range of technologies necessary to design, evaluate, construct, approve, operate, and maintain an initial FHR test reactor. The United States and China have signed a Memorandum of Understanding on co-operation in nuclear energy science. Negotiations remain underway on establishing specific collaboration on joint US-China FHR technology development activities.

The US has been pursuing FHR technology development roughly the past decade. FHR development in the US is sponsored by the Department of Energy's Office of Nuclear Energy (DOE-NE). A broad-scope FHR technology development roadmap was recently completed. Additionally, DOE-NE provided a canister of prototypic primary coolant salt to the Czech Republic Ministry of Industry and Trade to enable criticality testing using the LR-0 test reactor at the Nuclear Research Institute near Prague. Also, DOE-NE recently completed construction of a liquid salt hydraulic test loop at its Oak Ridge National Laboratory (see Figure 3.47).

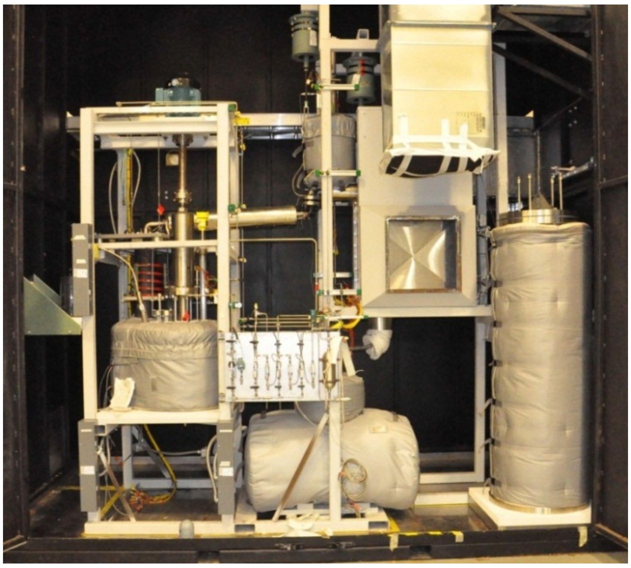
Figure 3.46: Surface layer of Ni – based alloy specimens after 250 hrs exposure under strain 25 MPa at 1 013K in fuel salt with [U(IV)]/[U(III)] ratios 100 (left) and 500 (right): (a and b) HN80M- VI, (c and d) HN80MTY, (e and f) HN80MTW, (g and h) EM-721; enlargement ×50)



(a) (b) (c) (d) (e) (f) (g) (h)

US FHR development activities include a broad and active university-based component. University lead activities include FHR concept development, reactor physics optimisation, surrogate fluid thermal and hydraulic modelling and simulation, advanced structural and functional material development and evaluation, economic performance evaluation, and advanced power cycle design. More generally, a number of the broad set of advanced reactor technology projects being pursued by DOE-NE support FHR development. In particular, the advanced gas reactor programme's development and testing of coated particle fuel, the joint DOE-NE and Nuclear Regulatory Commission initiative to develop a technology neutral advanced reactor licensing framework, and the ongoing advanced ceramic composite development and standardisation activities are each key elements to FHR development and deployment. Further, specific technology development projects such as high-temperature fission chambers, in-vessel optical instrumentation access, and salt compatible canned rotor magnetic bearing pump also support FHRs. The American Nuclear Society (ANS) has also initiated development of an FHR design safety standard that is co-chaired by US and Chinese representatives. Finally, ASME has initiated an effort to develop a high temperature.

Figure 3.47: Liquid salt test loop featuring 200 kW induction heating, an HF/H_2 based cleanup system and SiC test section, along with flow, level and temperature measurements



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Chapter 4. Methodology working group reports

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

4.1 Economic assessment methodology

The EMWG (Economic Modelling Working Group) was formed in 2003 to develop a cost estimating guidelines to be used for assessing GIF systems against the GIF economic goals. Its creation followed the recommendations of the Economics Crosscut Group (ECG) of the US DOE Nuclear Energy Research Advisory Committee (NERAC) and the Generation IV International Forum (GIF); see *A Technology Roadmap for Generation IV Nuclear Energy Systems* (December 2002). The ECG recommended that a standardised cost estimating protocol be developed to provide decision makers with a credible basis to assess, compare, and eventually select among nuclear energy systems, taking into account a robust evaluation of their economic viability; see *Generation IV Roadmap: Crosscutting Economics R&D Scope Report* (October 2002).

The methodology developed by the EMWG (Figure 4.1) 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 levelised unit cost of energy);
- to have a level of financial risk comparable to other energy projects (i.e., to involve similar total capital investment cost, equal to the "capital at risk" at the time of commercial operation);

The EMWG methodology consists of:

- *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems*, Rev. 4.2 (GIF/EMWG/2007/004). https://www.gen-4.org/gif/upload/docs/application/pdf/2013-09/emwg_guidelines.pdf;
- G4ECONS (Generation IV Excel Cost Calculation of Nuclear Systems) software package (see Figure 4.2);
- User's Manual for G4ECONS Version 2.0 (GIF/EMWG/2007/005).

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

Figure 4.1: Structure of the GIF cost estimating methodology

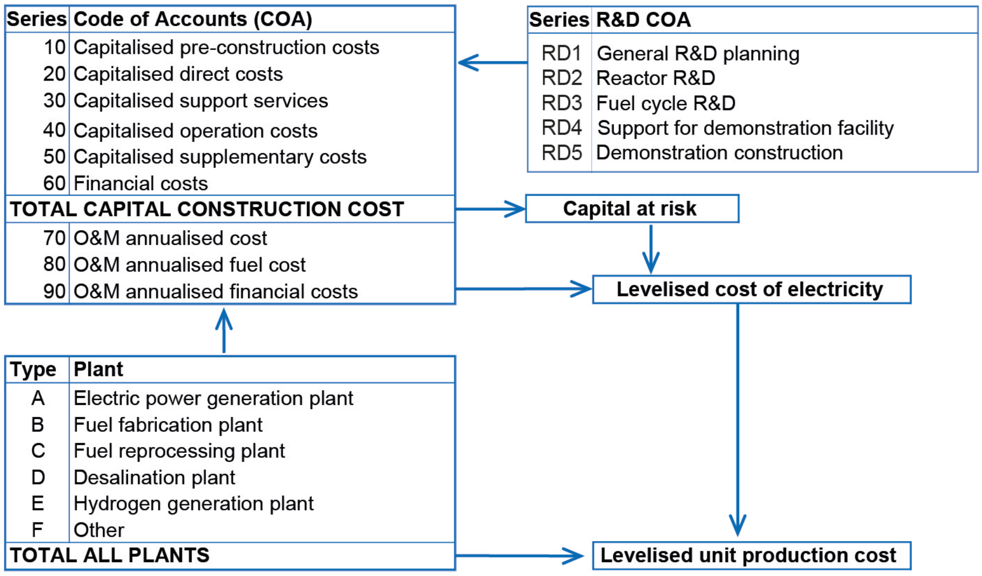
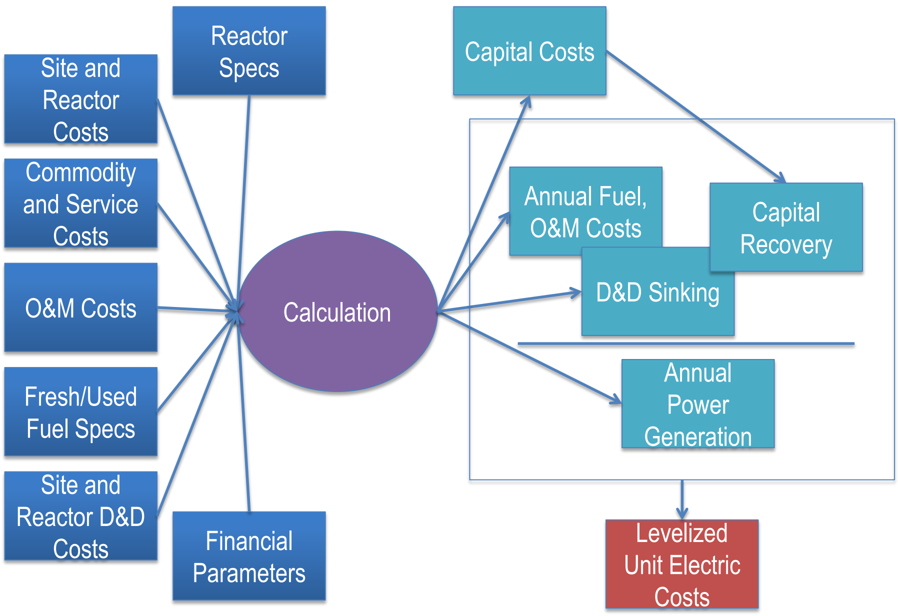


Figure 4.2: Overview G4ECONS version 2.0



Previous years

The EMWG, with the agreement of the GIF Expert and Policy Groups, released the methodology for public as well as GIF application. A CD is available from OECD/NEA containing the complete methodology. To date, over 160 copies of the methodology CD have been provided to those organisations requesting its use. In addition to GIF groups, the software has been requested by various IAEA groups, several universities, and a number of consulting companies.

The EMWG has also developed standard training presentations for the application of the methodology. The training presentations are modularised so as to be useful for presentation from a management level (Level-1: 45-minutes) to a detailed user's level (Level-3: all day with half a day devoted to calculating levelised cost of a particular technology). EMWG members are prepared to give this presentation to GIF groups as requested. Level-1 training presentations have been given to several GIF groups. See, for example, http://web.ornl.gov/~webworks/cppr/y2001/pres/125294.pdf.

The G4ECONS software has been extended to fuel cycle applications. Enhancement of the G4ECONS software has been suggested to better facilitate the analysis of heterogeneous fuel cycles that may be proposed for fast reactor systems for actinide management applications. Several studies were done to demonstrate an approach for estimating the cost of sectors of nuclear fuel cycles.

Applications of the methodology were done by the Japanese EMWG members to estimate the cost of the Japanese SFR and compared with other Japanese cost models. This and many other applications reviewed are proprietary and not available in the open literature.

2012-2013 activities

The EMWG had its 27th meeting on 14 June 2012 in Washington DC, hosted at the Oak Ridge National Laboratory DC office; participated in the GIF Symposium in San Diego, California, hosted by the American Nuclear Society; its 28th meeting on 8 March 2013 in Paris, France, hosted by the OECD/NEA; and its 29th meeting on 20 November 2013 in Brussels, Belgium, hosted by the European Commission.

Much of the discussion at these meetings has been focused on the comparison of the INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) cost calculation methodology and the EMWG Guidelines and software. For example, in the meeting in Brussels, it was decided that we would develop common sets of input data for Generation II, III, and IV reactors with which to compare software.

A Beta version of a new G4ECONS that can incorporate uncertainty has been prepared and reviewed by the EMWG. Development continues on the upgrade that will be finalised and tested in the near future with the common datasets being developed with a team from INPRO.

One of the EMWG members presented the results of a consulting report with E.ON New Build & Technology GmbH comparing the costs of 1) high-temperature gas pebble-bed reactor, based on the Chinese reactor under construction (with data from 2009); 2) the European sodium fast reactor (with data from 2009); 3) the European lead fast reactor (with data from 2009); 4) European fast gas reactor (with data from 2011); 5) the European super-critical water reactor (with data from 2008); and 6) molten salt reactor (with data from 1971). This paper was presented at the GIF Symposium and published in the *2012 Annual Report*.

The EMWG continues to monitor the use of the methodology and encourages feedback on its use and possible improvement with the Expert Group, the Policy Group, the Senior Industry Advisory Panel, and the Technical Working Groups on economic and cost matters.

4.2 Proliferation resistance and physical protection assessment methodology

According to its mandate the PRPPWG has developed a methodology for evaluation of the proliferation resistance (PR) and physical protection (PP) characteristics of GIF nuclear systems. The main objective of the PR&PP methodology is to facilitate the introduction of proliferation resistance and physical protection features into nuclear system concepts at the earliest possible stage of their design in order to optimise the effectiveness of PR and PP measures. The activities of the Group aim at ensuring that PR&PP issues are taken into account, together with safety and economics, in the research and development work undertaken within GIF and eventually in a broader context.

The latest version of this PR&PP methodology is described in a document released in 2011 for general distribution within and outside of GIF. The document, entitled "Evaluation methodology for proliferation resistance and physical protection of Generation IV nuclear energy systems", Revision 6, is available on the GIF public web site. Outputs from the PRPPWG also include a joint report with SSCs, entitled "Proliferation resistance and physical protection of the six Generation IV nuclear energy systems" (GIF/PRPPWG/2011/002) and a case study report describing an application of the PR&PP methodology to an Example Sodium Fast Reactor, entitled "PR&PP evaluation: ESFR full system case study final report" (GIF/PRPPWG/2009/002). Both reports are available on the GIF public web site.

Based upon those previous achievements, the activities of the Group now focus on:

- seeking collaboration with SSCs and other potential users of the methodology, with the mid-term to long-term objective of applications to GIF designs to illustrate its relevance and to enhance its user friendliness;
- communicating the methodology at GIF-sponsored meetings and as other opportunities arise;
- maintaining co-ordination with the IAEA especially in the framework of the INPRO project and the Agency's efforts in safeguards by design.

Contacts with SSCs were maintained through participation of representatives of the PRPPWG in GIF Experts and Policy Group meetings and exchange of information through electronic mail. This ensures that SSCs are aware of the PR&PP Methodology capabilities and know that it could be applied to their system whenever they will be ready for embarking on a study.

Following the International Seminar on the Nuclear Fuel Cycle in Russia and Proliferation Resistance and Physical Protection Evaluation (PR&PP) Methodology for Generation IV Nuclear Energy Systems, held in Moscow in October 2012 in connection with the 23rd meeting of the Group, lessons learnt from the exchanges between Russian experts and members of the Group were analysed. The main concern raised by experts outside the Group was the apparent complexity of the methodology, even for trained analysts but in particular for policy and decision makers. The Group is working on this issue through presenting and publishing illustrative examples of applications with the objective of providing potential users with guidance on the data required, the process to follow, and the results to be expected. In particular, studies by Japan and the European Commission are discussed below.

Responding to the demand from potential users of the methodology and from policy makers, the Group has prepared a synthetic two-page leaflet on frequently asked questions (FAQ) to the Group. The document, aimed at a broad audience, is an introduction to the methodology which aims to provide a rapid overview on its objectives and expected results. It gives insights on data and manpower required to apply the methodology and on the overall efforts and time needed to carry out a specific study. Reviewed by the GIF Experts Group, the document was cleared by the Policy Group at its November 2013 meeting for general distribution within and outside GIF and is posted on the GIF public web site.

Several members of the Group carried out studies using the Methodology and presented the outcomes in various *fora*. Within Euratom the methodology was used to explore proliferation resistance issues related to the European Sodium Fast Reactor concept. Results presented regularly at the PRPPWG meetings since 2011, were presented inter alia at the European Nuclear Society Conference (ENC) in December 2012 and at the 35th ESARDA Annual meeting in May 2013. In Japan, a study being carried out at JAEA focusses on applying the methodology to a fast reactor fuel cycle and preliminary results were presented at the 24th meeting of the Group in October 2013. The PR&PP methodology is also used for studies outside the group. The application of the PR&PP methodology to the MYRRHA research reactor is a prominent example.

Within GIF, the PRPPWG has maintained a close coordination with the Risk and Safety Working (RSWG) to ensure synergy and complementarity between the methodologies developed by the two Groups. A joint session was organised in connection with the respective meetings of

RSWG and PRPPWG held in Obninsk, Russia, in October 2012. Following the RSWG meeting held in October 2013, it was agreed that in due course the PRPPWG members will be invited to comment on the white papers on safety of GIF systems prepared jointly by SSCs and the RSWG. In addition, the Group is investigating the potential impacts on the PR&PP methodology of the 3 S (Safety, Safeguards, Security) concept under consideration in several circles of experts and policy makers in various countries.

Coordination and collaboration with INPRO and other IAEA projects, in particular in the field of safeguards, is especially important for PRPPWG in the light of the prominent role of IAEA in the area of proliferation resistance. Members of the Group are participating in some INPRO projects and in other IAEA activities such as the work on Safeguards By Design which raises high interest in countries where new advanced reactor concepts are in the design and development process.

The INPRO project on "Proliferation resistance and safeguardability assessment tools (PROSA)", which was completed at the end of 2013, illustrates the close coordination between GIF and INPRO in the domain of PR. The project, in which several members of the PRPPWG were involved, offered opportunities to identify commonalities and differences between the approaches and methodologies developed by the two groups and to exploit synergies between the two projects. As PROSA was reaching its completion, a joint meeting was held in October 2013 in conjunction with the 24th meeting of the PRPPWG. The session helped in assessing consistency in methodology and terminology, identifying commonalities and synergies, and exploring potential benefits of further joint activities.

Outside GIF, the Group monitors national and international activities and studies which could have an impact on future work aiming at adapting the methodology to new needs of analysts and policy makers and further development of evaluation tools. In this connection, the Group has followed the work of the US National Academies on methodologies for proliferation risk assessment and how they relate to the needs and questions of policy makers in this area, which was completed in June 2013. The US sponsors of the national academies agree with its conclusions and are developing a fact sheet on these conclusions. The PRPPWG will be informed accordingly.

4.3 Risk and safety assessment methodology

Activities of the risk and safety working group (RSWG) were focused on three key areas during 2013:

- the development of the GDI, the Guidance Document for ISAM (the Generation IV integrated safety assessment methodology);
- the review of the SFR Safety Design Criteria (SDC) report and associated activity as the 2nd Phase for developing the Safety Design Guideline (SDG);
- the white paper preparation as the interactions with the six reactor systems especially on the safety aspects and the application of the ISAM.

The ISAM* is an integrated safety assessment methodology developed by RSWG and consisting of a set of analysis tools selected for use at various stages of nuclear system design development throughout the Gen IV technology development cycle. The tools selected in the ISAM are: qualitative safety features review (QSR), phenomena identification and ranking table (PIRT), objective provision tree (OPT), deterministic and phenomenological analyses (DPA), and probabilistic safety analysis (PSA). In response to the feedbacks from the expert group (EG) and system steering committees (SSCs) on the "integration" of the different tools that characterise the methodology and practical guidelines on its use, the RSWG has identified the need to develop a guidance document for application of the ISAM (GDI) in the development of Generation IV systems.

* The document "An integrated safety assessment methodology (ISAM) for Generation IV nuclear systems" is available at: www.gen-4.org/gif/upload/docs/application/pdf/2013-09/gif_rsgw_2010_2_isamrev1_finalforeg17june2011.pdf.

Namely, the purpose and the needs for the GDI are: "how to apply the ISAM in design and assessment", "when to utilise ISAM during system development process", and "what are criteria/standards to be referred". The GDI is a document including a step-by-step description on how to apply ISAM: the users can recognise when to use the ISAM tools in the safety assessment process, can identify inputs and outputs of each ISAM tool, can follow the flow from one step to another, and can elaborate a flow chart in support. In practical terms, the GDI, as a specific guidance on ISAM application, is expected to aid the SSCs as they use the ISAM to develop and improve the respective systems. Likewise, the RSWG will be looking for opportunities to directly support more systematic, comprehensive and detailed applications. The RSWG is now updating the draft version of the document and including the internal comments by the RSWG members. The GDI will be, then, provided to the EG and policy group (PG) for review, and will be provided to the SSCs as the practical guidance document of the ISAM to utilise in their design and safety assessments processes.

Late in 2010, responding to the direction from the GIF Chair, a task force (TF) was formed to define and articulate safety design criteria for SFR systems. The TF is comprised of representatives of the RSWG, the SFR system steering committee, and other interested representatives of the GIF SFR community. The work of the TF began in 2011 and developed the SDC during two years, and the SDC Phase 1 Report was finalised and provided relevant international organisations and national regulatory bodies in GIF SFR member countries. The RSWG had reviewed the SDC, and the recommendations on the updates of the SDC had been included in the SDC Phase 1 Report. The RSWG's detailed reviews on the SDC text were mainly on "General part (Safety approach and safety assessment for the Gen IV reactor system)" and "Criteria" (of overall & specific plant system)". The feedback from the international organisations is foreseen in the coming years, and the RSWG will contribute to the feedback processes between the GIF community and the international/national organisations via general technical reviews on the safety as the Generation IV reactor systems. The RSWG will contribute to the SDC TF 2nd phase activity as well that are for developing the SDG via the interaction with the TF, especially with the high level, general, and comprehensive viewpoints.

In 2013 the RSWG further encouraged the development of the Gen IV reactor systems white papers on safety. The white papers are produced by the six Generation IV SSCs with the support of the RSWG and, and are presenting high level information about safety-related design issues and phenomena associated with each of the six Generation IV system concepts, as well as early thinking about safety assessment for these systems for possible improvement/upgrade of the safety features. The development of the white paper was envisaged to stimulate the RSWG interaction with the respective SSC on safety aspects and facilitate the pilot application of ISAM. Noting the different maturity of the various GIF concepts, the progress achieved so far with the development of the white papers on safety is relatively moderate. The SFR white paper on safety was completed, while first version of the white papers for GFR and LFR are being prepared for the Euratom concepts. Some progress was also achieved with the development of the VHTR white paper on safety. It is RSWG's intention to further support the designers in completing this task.

During 2013, the RSWG also monitored and interpreted the lessons learnt from TEPCO's Fukushima Daiichi Nuclear Power Plants in order to evaluate those lessons for their applicability and implications for Generation IV safety work. This work will continue in the coming years. A likely outcome of this work, for example, is the increased emphasis on consideration of external events, some of them characterised by high uncertainty on magnitude and frequency of occurrence. This will be related to the Gen IV safety and reliability goal number 3: "Generation IV nuclear energy systems will eliminate the need for offsite emergency response."

The RSWG experienced important changes to the group's composition during 2011-2013. Retirements of the group's original co-chairs presented challenges in terms of the continuity, and new co-chairs started to take the leadership with the opportunities for new directions and fresh thinking. For the coming next decade, the RSWG activities and roles will be updated in accordance with the developments and progress of the six reactor systems, for supporting the elaboration of the safety design of the systems and for providing the risk-informed technology neutral framework and approach for safety assessment. RSWG will continue to promote consistent approach on risk, safety, and regulatory issues between Generation IV systems; elaborate on safety principles, objectives, and attributes based on Gen IV safety goals to guide R&D plans, and provide consultative support on safety to EG, SSCs and other Gen IV entities. In addition, the RSWG maintained its interfaces with the regulators, IAEA, INPRO, MDEP, and the PRPP methodology working group, participating in joint meetings or otherwise pursuing mutually beneficial collaborations with each of these organisations.

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Chapter 5. Task force reports

5.1 Task force on safety design criteria

In 2013, the sodium-cooled fast reactor task force (TF) on safety design criteria (SDC) completed a first report on SDC and started a follow-up activity on the development of safety design guidelines (SDG). The SDC Report, which had been prepared by the TF during the last two years, was approved by the GIF policy group as a Phase 1 Report, and was made available for review to international organisations and national regulatory bodies in GIF SFR member countries. In September 2013, the TF held its first meeting of the second phase activity, which is devoted to the development of SDG by the end of 2016.

Report on safety design criteria

Following proposals to establish "safety design criteria" for Generation IV SFR reactor systems raised at the PG meeting in October 2010, the PG decided to establish a SDC TF in May 2011. The first TF meeting was held in July 2011. The objectives of the SDC are to provide reference criteria for the safety design of the structures, systems and components of SFR systems, where the criteria are systematically and comprehensively consistent with the safety and reliability goals defined in the *GIF Technology Roadmap* as well as with the basic safety approach defined in the GIF's "Basis for the Safety Approach for Design & Assessment"

In the development process of the SDC, the reviews/feedbacks have been conducted within the GIF community. The draft SDC report* was submitted to the IAEA INPRO for review at the third joint GIF-IAEA workshop of "Safety Design Criteria for Sodium-cooled Fast Reactors" in February 2013. Following this review process, the final reviews/comments on the SDC Report from the GIF member states were provided and included, and the report was then approved by the PG as the SDC Phase 1 Report in May 2013. In order to obtain feedback from outside of the GIF community, the SDC Phase 1 Report was made available to international organisations (e.g. IAEA, MDEP, and OECD/NEA/CNRA) and regulatory bodies of the GIF SFR member states. Further feedback and comments on the SDC Phase 1 Report are expected in the coming years.

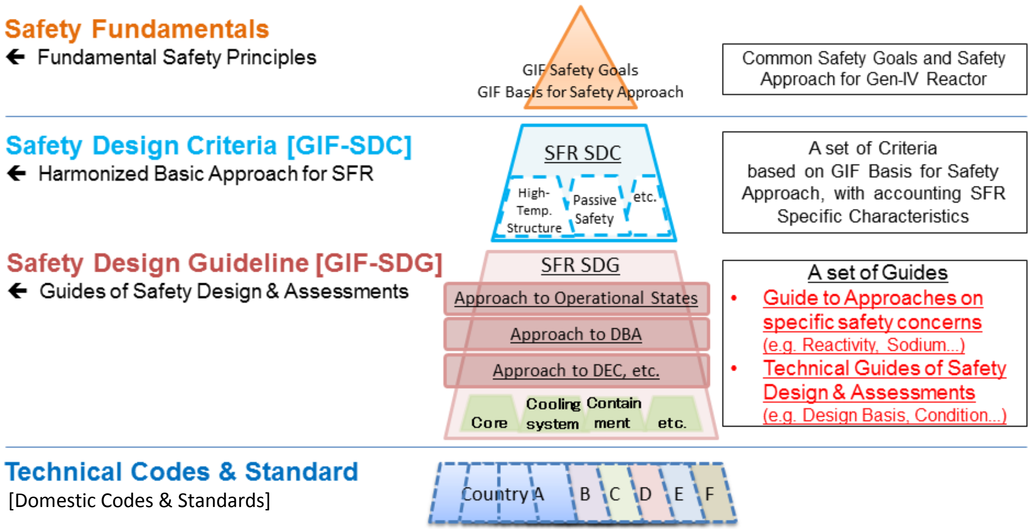
Development of safety design guidelines (SDC TF second phase activity)

The SDC development activity made a remarkable achievement with the publication of the "SDC report" related to the safety design of the Generation IV SFR reactor systems, and thereby raised some additional questions and thus motivations for further technical interpretation and clarification of the SDC. In the interactions with other GIF entities, the recommendations for follow-up actions were: To develop detailed guidelines for application of the SFR SDC", "To pursue quantification/qualification of key aspects", and "To start discussions on technical guidelines, including the regulators and their technical support organisations". In addition, the needs and intentions obtained through the SDC TF activities, especially on developing "Detailed guidelines to support practical application of the SDC" and "Common understandings on technical issues", were also raised. Based on these recommendations/needs/intensions, the Terms-of-Reference (ToR) of the development of "Safety Design Guidelines" as a 2nd phase activity of the SDC TF was prepared by the TF. This ToR was approved by the PG in May 2013. The TF held its 1st meeting of the 2nd Phase activity in September 2013.

In the hierarchy of the safety standards shown in Figure 5.1 below, the safety and reliability goals and the basis for the safety approach of Generation IV nuclear systems have been established as the highest level safety fundamentals, whereas the SDC has been established as the second highest level. However, there is still a gap between the SDC and the country-specific codes and standards at the base level of the safety hierarchy, and the SDG is expected to provide practical guidance on how to apply the SDC in order to resolve the technical concerns when designing the structures, systems and components of GIF SFR systems. It can be said that the SDC is a set of criteria which are based on the GIF safety approach, accounting for specific characteristics of SFRs, whereas the SDG is a set of guidelines on the approaches of specific safety concerns (e.g. reactivity, sodium…) and on the technical aspects of the safety design and assessments (e.g. design basis, design conditions…).

There are two expected outputs from the 2nd Phase of the SDC TF. The first output is a report on “Guidelines on Safety Approach and Design Conditions of Gen IV SFR systems,” which will be completed by the end of 2014. It is for guiding safety approaches according to the SDC and it will be used as a supplemental technical document of the SDC for clarification of technical questions. It will be based on the safety approach and technical issues listed in the SDC Report, and the primary contents are set on “prevention and mitigation of severe accidents (reactivity issues)” and “accident conditions to be practically eliminated (issues related to loss of heat removal)”. The second output is a report on “Guidelines on the Key Structures, Systems and Components” including “Specification of constrains of the design and assessment”. Here, the primary goals are related to fundamental safety functions, such as “shut down”, “decay heat removal”, and “containment”. The SDG development will be started with “functional requirements of structures, systems and components adapted to the SDG on safety approach,” then evolve to a “set of design conditions”, such as postulated events, design parameters and design limits.

Figure 5.1: Hierarchy of safety standards (including GIF SDC and SDG)



5.2 Task force on advanced simulation

Origin of the task force

A specific Task Force had been created by the policy group in 2009 to examine current initiatives and the interest and perspectives for expanded collaboration among GIF partners in the field of Advanced Simulation and associated Verification and Validation. The Task Force issued a report in May 2012, endorsed by the Policy Group, with among others the following recommendation: “The Task Force recommends holding a workshop with developers and designers to define the need for collaboration within GIF on High Performance Computing and Uncertainties Quantification, if any.”

Workshop organisation

A special GIF workshop on “Advanced simulation in support to GIF reactor design studies – Contribution of High Performance Computing and Uncertainties Quantification” was held on 27 October 2013 in Paris as a side meeting of the conference on “Supercomputing in Nuclear Applications and Monte-Carlo (SNA + MC) 2013”.

The objective of the workshop was first to exchange information between GIF systems and components designers and experts in advanced techniques for numerical simulation; then to evaluate the interest of introducing these techniques in the Generation IV International Forum’s projects; and finally to hold a conclusive debate.

Workshop presentations

Eight topics were presented related to the use of advanced techniques in reactor design studies, by representatives of France, Republic of Korea, and Russia. The rising importance of Computational Fluid Dynamics to help optimising an existing design was stressed. In parallel, examples were shown where high performance computing had been used to estimate uncertainties in a given design performance study or to define an operating area of a component where numerous parameters can vary. Presentations can be found in the conference proceedings.

Discussion panel

At the end, a discussion panel addressed the use of supercomputing in design studies of Generation IV reactors. Panellists were from different areas, working on performance studies, simulation codes, safety studies, programme management, and from different countries (France, Republic of Korea, Russia, and the United States). They showed how difficult it is today to promote high performance computation and coupling of disciplines in current design studies. These advanced techniques are more often used for performance evaluation of component or equipment than for drawing a conceptual design. Nevertheless, it can be stated that computational fluid dynamic is the only advanced discipline that is used as a routine to optimise a component. Finally uncertainties evaluation shows a preference for human expertise than multiple calculations, with the goal of convincing a regulator.

5.3 Strategic planning activity (2012-2013)

In 2012, the GIF engaged in a strategic planning activity under the then Chair of the Policy Group, Mr. Sagayama. The objectives of this activity were to assess progress made in the first decade since the initial Technology Roadmap was published, to draw lessons from the way projects were set up and run with a view to improve the level of collaboration within the forum, and to identify benefits of reaching out to other international organisations and academia with the objective of making the work of GIF better known and benefiting from relevant activities performed outside of GIF. A specific Task Force was set up, split in three sub-task groups, ST1 (technology roadmap update), ST2 (strengthening R&D collaborations) and ST3 (strengthening ties with other organisations), as illustrated in Figure 5.2.

5.3.1 Update of the Technology Roadmap (ST1)

More than ten years after the publication of its original Technology Roadmap, the GIF felt that it was necessary to update this document, to take note of technical progress made during the last decade, review the level of interest and engagement in moving from R&D activities related to viability and performance assessment of the six GIF systems to their demonstration phases. Most importantly, several years after the Fukushima Daiichi accident, the update of the Technology Roadmap provided the right opportunity for the GIF to reaffirm the importance of the safety goals that the forum has set itself, to ensure that Generation IV systems are designed with the highest levels of safety.

Figure 5.2: GIF Strategic Planning Task Force

The *Technology Roadmap Update*, published in January 2014, provides high level recommendations and milestones focusing specifically on updating the vision of system missions, R&D needs and plans for prototypes. The document examines whether there are new or different technical questions that will need to be answered in the future, whether there are new concepts that may be the focus of new collaborative research, and whether and how to expand on Gen IV goals given the developments made over the last few years, including lessons learnt from the Fukushima accident. The Technology Roadmap Update provides a set of important and challenging goals, activities and projects that need to be accomplished in the next decade through the GIF.

Figure 5.3: Development timelines for the six GIF systems in the updated roadmap

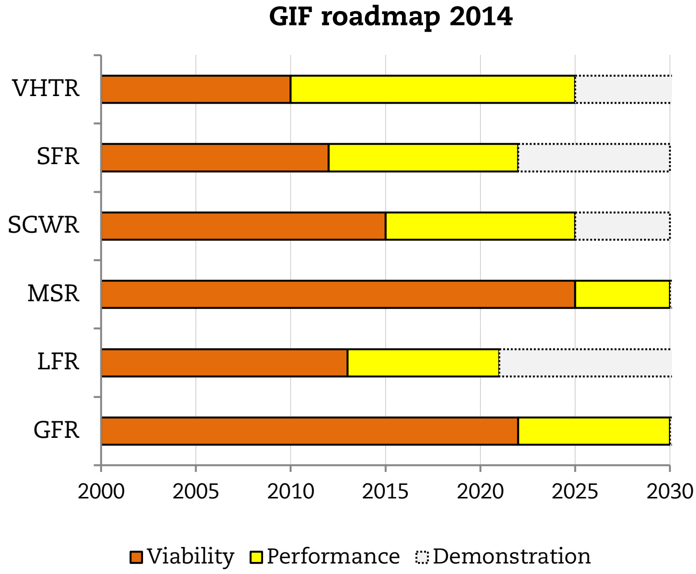


Table 5.1: Key objectives for the next ten years (from the *Technology Roadmap Update*)

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

Table 5.1: Key objectives for the next ten years *(cont'd)*

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

GIF METHODOLOGY WORKING GROUPS	
Economic Modeling Working Group (EMWG)	• Over the next two to three years, the EMWG will release a new version of the G4ECONS cost estimating code with advanced capabilities. • The *Cost Estimating Guidelines* will be reviewed after the *User Guide* updates: – Over the next 10 years, the EMWG will continue to monitor the progress of Generation IV systems economic analyses and further improve the methodology consistent with these designs.
Proliferation Resistance and Physical Protection Working Group (PRPPWG)	• As new and innovative designs for nuclear energy systems are developed through GIF (and other possible fora), the PR&PP methodology approach will be essential to incorporate good design principles for proliferation resistance and physical protection into these new designs. • Enable safeguards by design: Robust safeguards are essential to the PR&PP characteristics of all of the emerging GIF designs. • Assist GIF system developers in introducing PR&PP concepts into their design work.
Risk and Safety Working Group (RSWG)	• In 2008, the RSWG published the *Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems* – a consensus regarding some of the safety-related attributes and characteristics that should be reflected in Generation IV systems: – Future work: Provision for application of the integrated safety assessment methodology (ISAM) in the development of Generation IV systems. – A number of detailed analyses and "lessons learnt" investigations will be performed, especially as related to the Fukushima Daiichi accident.

The update builds on the original Technology Roadmap and assesses current technology advances in fuels, materials, modelling, fuel cycle strategies, etc., confirming that the GIF is pursuing the optimum set of advanced reactor concepts with the six systems originally selected. This update is also timely in addressing global post-Fukushima concerns for installed reactors, new designs, and advanced reactor concepts particularly in terms of severe accident prevention and mitigation.

To produce this document, the sub-task group ST1 was set up in 2012, under the chairmanship of Ch. Behar, Vice Chair of the Policy Group, and with a group of representatives from China, the Russian Federation, Switzerland and the United States. Chairs of the SSCs, the Technical Director and the experts group were also closely associated with the work of updating the technology roadmap. The roadmap update was first discussed at the GIF Symposium in San Diego in November 2012, then at the GIF EG and PG meetings in Beijing in May 2013.

Among the issues that were debated during the drafting of the technology roadmap update, one can cite the appropriate level of details for the "timelines" of the six Gen IV Systems (Figure 5.3), evolutions of national R&D policies since 2002 and how this had an impact on the development of the six systems, and the need to update and complement the list of key R&D objectives. After the Beijing meeting, teleconferences were organised by ST1 to take into account comments from the EG, the PG and the chairs of the SSCs. During that time, it was concluded that timelines should be extended to 2030 to clarify the status of each system regarding the viability, performance and demonstration phases. It was also recalled that these timelines are indicative, as they depend on the level of financial support for RD&D activities in each participating country, and on the licensing of demonstration reactors in the respective host countries. The key objectives for the six systems are recalled in Table 5.1.

5.3.2 Strengthening R&D collaboration (ST2)

Collaboration within the GIF framework has resulted in the completion of hundreds of research deliverables and milestones in such areas as irradiation and fuel development, materials, chemistry, safety and operations. While the GIF framework is an enabling tool for multilateral collaboration, external factors such as a severe global economic downturn, adverse policy decisions resulting from a major nuclear accident, and the current pace of Generation-III systems deployment, as well as internal factors and processes that require time, experience and initiative to mature, have hindered GIF from achieving its full potential as a collaboration framework. ST2 was charged with identifying internal issues with R&D collaborations within GIF and developing promising means of building stronger and broader collaboration.

ST2 used a four-step approach in developing recommendations for improving R&D collaboration:

- Develop a comprehensive understanding of impediments to more effective GIF R&D collaboration (survey and GIF Symposium).
- Analyse and prioritise the issues identified as barriers to effective R&D collaboration (binning).
- Develop an initial set of actionable recommendations based on survey input, binning results, thorough knowledge of how the Forum works today and credible potential outcomes if the recommendations are implemented.
- Develop implementation plans, including responsible parties, for prioritised recommendations to be approved by the GIF Policy Group.

The first step was to conduct a survey of all Policy Group (PG), Expert Group (EG), System Steering Committee (SSC), Methodology Working Groups (MWG), Project Management Board (PMB) members and key researchers and scientists. In general, respondents viewed GIF favourably – 82% of respondents believe that GIF is effective overall and 60% of respondents characterised R&D collaboration within GIF as excellent or good. Most participants saw room for improvement and identified areas where improvements could be made. Several best practices and areas that are working well were also identified by the survey.

The results of the survey were then binned into four categories: major issue, less difficult to fix; major issue, more difficult to fix; minor issue, less difficult to fix; and minor issue, more difficult to fix. The significance was assessed by determining whether the issue could negatively impact the Forum's *overall likelihood of success* during the next decade (major issue) or does it just make some aspects of collaboration less efficient (minor issue). The degree of difficulty was assessed by identifying whether the issue could be addressed *within the GIF framework* (low degree of difficulty) or whether action by *national governments or other external entities* was required (high degree of difficulty).

The outcome of the binning process was:

- minor issue, less difficult to fix – 61%;
- major issue, less difficult to fix – 29%;

- major issue, more difficult to fix – 8%;
- minor issue, more difficult to fix – 2%.

The results of the survey and binning process were used to identify four main issues that were predominantly identified by the GIF community as areas needing improvement, taking into account strong correlations among many comments and a sound understanding of GIF operations. The selected issues were chosen based on their overall importance to the GIF community, their ability to influence international collaborations in the near and long term and that their associated recommendations could be addressed through the existing GIF framework, by GIF members and within a reasonable time frame.

The first issue identified was project management which was categorised as a major issue and less difficult to fix. This issue focused on the need for specifying clear technical objectives and scope as well as periodically updating project plans to reflect changes in national priorities, funding, signatories, technology advancements or design parameters. Identifying best practices and providing examples and templates to SSCs and PMBs was recommended to improve this area. It was also suggested that project plans be assessed against established criteria and updated as needed and that a required project plan review cycle be put in place.

The next identified issue was sharing of capabilities and resources which was categorised as a major issue and more difficult to fix. Evaluating interest in creating crosscutting projects, investigating whether a framework is needed for sharing GIF experimental facilities and developing a process for exchanging researchers and students among GIF members, were all suggested as methods to improve the sharing of capabilities and resources.

Communication was identified as a minor issue which was less difficult to fix. Methods of improving both internal communication and external communication with key stakeholders were recommended. These recommendations included better utilisation of NEA as a resource, improving the GIF external website, communicating achievements to external stakeholders and improving communication between PG, EG, SSCs and PMBs. Improving communications may also help address more difficult issues such as funding and visibility for GIF.

The final identified issue was Senior Industry Advisory Panel (SIAP) engagement which was categorised as a minor issue and less difficult to fix. Industry participation is key to ensuring cost effective optimisation of GIF R&D. Enhancing the use of the EG to support SIAP and establishing a mechanism to enhance interactions between SSCs, PMBs and SIAP were suggested as methods of improving SIAP engagement.

The EG has been tasked with evaluating the identified issues and recommendations and determining the best implementation strategies. Implementation of identified recommendations will improve GIF collaborations for the coming decade and beyond.

5.3.3 Strengthening ties with other international organisations (ST3)

The sub-team ST3, chaired by Yutaka Sagayama, former GIF Chair, was formed by the representatives from Canada, South-Africa and EU and NEA. The sub-team reviewed the current co-operation activities with the following organisations and proposed possible action items for the enhancement of GIF's co-operation with them.

- International Atomic Energy Agency (IAEA);
- OECD Nuclear Energy Agency (OECD/NEA);
- Multinational Design Evaluation Programme (MDEP);
- International Framework for Nuclear Energy Cooperation (IFNEC);
- Academic societies, universities and industry.

Collaboration with other international organisations is useful to achieve GIF's objectives to promote R&D on Gen IV systems efficiently and effectively and also helpful to achieve GIF's goals of Sustainability, Economics, Safety & Reliability and Proliferation Resistance and Physical Protection (PRPP). Appropriate co-operative activities with other international organisations that have different characteristics will create synergy and provide strong support to achieve our goals.

International Atomic Energy Agency (IAEA)

GIF has collaborations with the IAEA. Annual meetings were established between GIF and IAEA/INPRO. GIF has also interactions with the Department of Nuclear Energy, the Department of Nuclear Safety and Security, and the Department of Safeguards of IAEA, as well as Technical Working Groups of the Department of Nuclear Energy. There are a significant number of possible areas of broadened and strengthened co-operation between GIF and the IAEA. Given the large number of potential areas of co-operation, it is necessary to have an approach to prioritise. It should also be noted that there are some potential challenges related to membership and intellectual property rights. The expectation for the future is that strengthened co-operation should focus on information exchange, methodology development, and development of design criteria, particularly safety design criteria (SDC). Establishing guidelines or guidance for PRPP and In-Service Inspection (ISI) will be in the scope of new GIF collaborative activities.

OECD Nuclear Energy Agency (OECD/NEA)

The NEA addresses scientific and safety issues for both current and advanced concepts of nuclear energy systems and helps to maintain the necessary R&D infrastructure through international co-operation. It is therefore a win-win strategy for both organisations to co-operate more closely. A more systematic information of the GIF bodies about the relevant NEA activities and future programmes as well as a greater involvement of the GIF members in the decision making process regarding the future NEA programmes of work should benefit both organisations. The simplest and most efficient way could be to invite GIF's SSCs and PMBs once a year to put an item "GIF activities relevant to NEA" on their meeting agendas. Similarly, NEA should be given a slot once a year in PG meetings to present a broader view on its activities relevant to GIF.

Multinational Design Evaluation Programme (MDEP)

The MDEP is an important forum for discussing new reactor safety issues and exploring harmonisation and convergence opportunities for new reactor regulatory practices. GIF Projects will be able to take benefit from the MDEP experience in making comparisons of the regulatory practices in the member countries, identifying differences, and developing common positions and methodologies. GIF should show the SDC to MDEP to get their comments to it. The work methodologies of MDEP should be known and followed up by GIF as much as possible, and transmitted to each GIF System and Project. This will allow GIF members to get direct contacts with regulators. Further, mutual participation to regular meetings, GIF Policy Group (PG) meetings and MDEP biannual conferences, of their representatives with a presentation of their work status and methodologies development, as well as collaboration in working group level are also proposed.

International Framework for Nuclear Energy Co-operation (IFNEC)

IFNEC consists of 31 partner countries, 30 observer countries, and 3 observer organisations including GIF. Introduction of the GIF activity and outline of Gen IV systems gives a better understanding of IFNEC participants on the target and outline of Gen IV system from its development stage. It may be a strategic arrangement for their future introduction of Gen IV system and lead its global expansion. Further introduction activities on GIF technical collaboration and technical communication seem effective, such as giving technical advice or suggestions in its infrastructure development working group on Small Modular Reactor (SMR), inviting IFNEC participants to GIF symposiums and distributing GIF annual reports and education material of Gen IV systems.

New collaborations with academic societies, universities and industry

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. Senior Industry Advisory Panel (SIAP) provides advice to the PG from the perspective of industry. The collaborations with academic societies and universities involved with GIF system R&D will enable exchange of information and avoid the duplication of efforts. These collaborative activities will allow GIF

Member States to share knowledge and information on the respective researches on the GIF systems. In order to reflect the opinions of the industries and to deepen their understanding about the GIF activity, the collaboration with them from an early stage is effective. The actions proposed are considering collaborations at policy level with industry including collaborations with the nuclear industry associations/societies in GIF participating countries and collaborations with universities and academic societies, R&D with universities, and workshops with academic societies.

References

GIF/RSWG/2007/002 (2008), "Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems", GIF Risk and Safety Working Group.

Third Joint GIF-IAEA Workshop on "Safety Design Criteria for Sodium-Cooled Fast Reactors" available at www.iaea.org/NuclearPower/Meetings/2013/2013-02-26-02-27-TM-SFR.html, 26-27 February 2013.

Chapter 6. Senior Industry Advisory Panel (SIAP)

The senior industry advisory panel (SIAP) provides advice to the GIF policy group on GIF nuclear energy system development from the perspective of industry, on issues related to technology development, demonstration, and deployment, and commercialisation of advanced nuclear energy systems. SIAP meets at least once per year to consider systems and/or crosscutting issues identified by the policy group, to provide its recommendations relative to development, deployment, future nuclear fuel cycles, and international frameworks for safety standards and regulations. At its meeting in May 2013, the Policy Group charged SIAP to consider and advise on the development of SFR safety design criteria and provide industry perspectives on the viability of Gen IV systems, with specific consideration given to particular topics.

SIAP provided a general observation that when developing innovative system concepts, technical viability is not sufficient as a development and demonstration objective. Innovation needs to have a legitimate ultimate use in a market context. Further, competencies (vendor, operator, regulator) should be identified for each system, including whole fuel cycle infrastructure and industrial applications.

In response to its review of the draft Safety Design Criteria (SDC) for the Sodium Fast Reactor, SIAP noted that in order to facilitate further review of the document, a briefing or executive summary should be prepared to a that presents the drivers for and history of preparation of the SDC, as well as a description of the process to facilitate effective review among stakeholder organisations. SIAP further suggested clarification of how the SDC will enhance current guidance and resolve any conflicts or inconsistencies with existing accepted guidelines. SIAP noted that safety design guidelines should reflect design approach, starting with market or customer needs, and not attempt to address safety in isolation of the design drivers. A suggested approach would be to establish gross plant requirements and then seek to eliminate classes of accidents with design decisions.

On the topic of off-site consequence potential of severe accidents, SIAP noted that the class of reactor systems, whether Gen III or Gen IV, is not important; expectations with respect to offsite consequences will be the same. Gen IV system designers should be familiar with these expectations and should engage in a dialogue with regulators to stay current with regulations and standards. SIAP recommended that the design process consider the design within the context of its application (e.g. process heat, etc.) and introduction of collocated hazards, such as chemicals or other energetic features. SIAP further observed that public acceptance of Gen IV systems will likely require that there be no offsite consequences of a severe accident.

SIAP was asked to comment on conditions under which closed fuel cycle system concepts could compete with light water reactors in the market. SIAP observed that the market currently does not support closing the fuel cycle and that government policies and support would be needed to create the necessary market incentives. Sustainability of nuclear energy will eventually require a closed cycle but the pace of implementation will depend on local market conditions. Currently fast reactors are considered as complementary to LWRs – they serve different purposes (such as breeding, transmutation, cogeneration) and in context of comprehensive fuel cycle. SIAP noted that the choice of an actinides management strategy will be driven by business and political factors (spent fuel disposition, uranium price, etc.) and in turn drive the choice of a reactor (or accelerator). SIAP did not judge the LWR transmutation concept to be a realistic solution.

SIAP noted that non-electrical applications of nuclear represent an important growth opportunity, although the specific reactor type or conversion technology should be driven by market conditions and needs. SIAP noted, in particular, that sustainability should be an important consideration for development of the VHTR fuel cycle. Any particular feature, such as minimization of cooling water for waste heat rejection, should be evaluated in the context of market needs, not as an inherent virtue. SIAP cautioned that potential users of collocated industrial plant applications do not want to be nuclear operators. There should be minimal coupling between the nuclear plant and process heat-driven plant to avoid no disruption of one side if the other is damaged. Process heat variability and availability are factors and will depend on how well is the heat source is characterized and controlled.

SIAP concluded by suggesting GIF actions that could make the group more effective. Members agreed that receiving questions earlier and receiving relevant information briefings in advance of the meeting would provide needed context and improve ability to provide quality responses. The GIF secretariat is developing and implementing specific actions in response to these recommendations.

Chapter 7. Other international initiatives

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

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) of the IAEA was established in 2000 to help ensure that nuclear energy is available to contribute to meeting the energy needs of the 21st century in a sustainable manner. To achieve this, INPRO brings together nuclear technology holders and users to consider jointly international and national actions that would result in required innovations in nuclear reactors, fuel cycles or institutional approaches.

INPRO activities are undertaken in close co-operation with member states in the following main areas: National long-range nuclear energy strategies; Global nuclear energy scenarios on sustainable nuclear energy; Innovations in nuclear technology; and the Dialogue Forum.

Since 2009, GIF and INPRO exchange information on their relative programmes and organise co-ordinated activities, mainly in crosscutting area like safety assessment. In 2013, the annual GIF-INPRO interface meeting was held at IAEA headquarters, February 28 to March 1. Synergies were found in developing and using methodologies of reactor evaluation:

- Proliferation Resistance & Physical Protection: a position paper had been prepared and no further action is needed.
- Safety: there is a need to agree on a framework for co-operation in the assessment of advanced concepts.
- Economics: it has been proposed INPRO attends the Economic Modeling Working Group meetings.
- Education, Training and Human Resources: Exchange of status, progress, and crosscutting information.
- GIF is now invited to the Dialogue Forum between nuclear technology holders and users for possible participation.

At the same, a special workshop dedicated to the safety of Generation IV SFRs allowed discussions on safety design criteria and initiated GIF-INPRO collaboration in this field. Two main lines are foreseen: the development of safety design guidance, and involving innovative SFR design organisations to present engineering solutions able to meet Generation IV safety design criteria.

7.2 International Framework for Nuclear Energy Cooperation (IFNEC)

The International Framework for Nuclear Energy Cooperation (IFNEC) was created in 2010 from the previous Global Nuclear Energy Partnership. Two groups help supporting the IFNEC vision: the Reliable Nuclear Fuel Services Working Group and the Infrastructure Development Working Group. IFNEC gathers 32 participating countries, 31 observer countries, and 3 observer organisations, including GIF.

The IFNEC Mission Statement emphasizes the need for international co-operation to ensure the peaceful, safe, secure and efficient use of nuclear energy. Such co-operation applies to both nuclear newcomer countries and nuclear expansion countries, with consideration given to the whole fuel cycle. Front-end fuel cycle services are routinely provided through the commercial market, whilst back-end services are less well developed either by governments or commercial vendors. Services prior to final disposal are provided by a small number of companies but ultimate disposition is not currently provided on a commercial basis. In 2013, IFNEC worked on Comprehensive Fuel Services that encompass all components of the fuel cycle.

Comprehensive Fuel Services are optional, commercially-based, global, fuel cycle supplies and services that provide assurances of fuel supply and a responsible used nuclear fuel and ultimate waste management scheme; this includes enrichment and recycling activities. As such, this work is complementary of the research work devoted to reactor physics and technology made in the GIF.

The GIF Policy Director represented the GIF Chairman in October 2013 to the IFNEC Steering Group and fourth Executive Committee meetings in Abu Dhabi, United Arab Emirates. He explained that the thirteen members of the Generation IV International Forum are working together to lay the groundwork for the fourth generation of nuclear energy systems, that support excellence in safety and security, economics, sustainability, and resistance to proliferation. He presented the strategic planning efforts that include updating the technology roadmap, strengthening R&D collaboration, and strengthening ties with other international organisations.

It is important to note that some of the initiatives of IFNEC and some goals of GIF systems overlap and as such there may be an opportunity for some sharing of information. Specifically Economics and Non-proliferation may allow for beneficial exchange of information. Through the meetings of the Infrastructure Development Working Group, the development status of the six GIF systems is presented to the IFNEC community, and in return information concerning the progress of waste and spent fuel management infrastructures is made available for the GIF.

7.3 Multinational Design Evaluation Programme (MDEP)

The MDEP continues to be an important forum for discussing new reactor safety issues and exploring harmonisation and convergence opportunities for new reactor regulatory practices. MDEP members are the regulators from Canada, China, Finland, France, India, Japan, the Republic of Korea, the Russian Federation, South Africa, Sweden, the United Kingdom and the United States. India’s nuclear regulatory body AERB joined the MDEP on 4 April 2012. The United Arab Emirates regulatory authority has been an associate member since September 2012.

Dr. Allison Macfarlane formerly took over the Chairmanship of MDEP from André-Claude Lacoste, former Chairman of the French Nuclear Safety Authority ASN in January 2013. Ms. Macfarlane was sworn as the Chairman of the US NRC on 9 July 2012.

The OECD Nuclear Energy Agency acts as the Technical Secretariat for the MDEP. The International Atomic Energy Agency participates in the generic activities of the programme to ensure consistency with international requirements and practices. The MDEP focus on safety has become increasingly important in light of the Fukushima Daiichi accident.

In accordance with its terms of references, the MDEP carries out its work through design-specific and issue-specific working groups as follows:

- Design specific activities:
  - There are currently three working groups focusing on Gen III/III+ designs: the EPR working group, the AP1000 working group and the APR1400 working group.
- Issue-specific activities:
  - There are three working groups, working on vendor inspection co-operation, codes and standards and digital instrumentation and controls.

MDEP recognises that other organisations are implementing programmes to facilitate international co-operation on new reactors, but to maintain its specificity, MDEP focuses on short-term activities related to specific design reviews conducted by member countries as well as efforts to harmonise specific regulatory practices and standards.

Appendix 1. GIF technology goals and systems

A.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). 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 utilisation for worldwide energy production.*
Sustainability-2	*Generation IV nuclear energy systems will minimise and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for 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 co-operative 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 co-operation is considered essential for a timely progress in the development of Generation IV systems. This co-operation 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.2 Technology Roadmap Update

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 the technology roadmap update: www.gen-4.org/gif/upload/docs/application/pdf/2014-03/gif-tru2014.pdf

Appendix 2. List of abbreviations and acronyms

Generation IV International Forum	
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)
FQT	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)
MOU	Memorandum of understanding
MSR	Molten salt reactor
MWG	Methodology Working Group
PA	Project arrangement
PG	Policy Group
PMB	Project Management Board
PP	Physical protection or project plan
PPMB	Provisional Project Management Board
PR	Proliferation resistance
PR&PP	Proliferation resistance and physical protection
PRPPWG	Proliferation Resistance and Physical Protection Working Group
PSSC	Provisional System Steering Committee
RSWG	Risk and Safety Working Group
SA	System arrangements
SCWR	Supercritical-water-cooled reactor
SDC	Safety design criteria
SFR	Sodium-cooled fast reactor
SIA	System integration and assessment (SFR Project)
SIAP	Senior Industry Advisory Panel
SO	Safety and operation (SFR signed project)
SRP	System research plan
SSC	System Steering Committee
SWP	System white papers
TF	Task force
TH&S	Thermal-hydraulics and safety (SCWR signed project)
VHTR	Very-high-temperature reactor

Technical terms

AECS	Advanced energy conversion system
AGR	Advanced gas-cooled reactor (United States)
AHTR	Advanced high-temperature reactor
ALFRED	Advanced lead fast reactor European demonstrator
ASTRID	Advanced sodium technological reactor for industrial demonstration
ATR	Advanced test reactor (at INL)
AVR	*Arbeitsgemeinschaft Versuchsreaktor*
CCG	Creep crack growth
CEFR	China experimental fast reactor
CFD	Computational fluid dynamics
COL	Combined construction and operating licence
CRP	Co-ordinated research project
DHR	Decay heat removal
DNS	Direct numerical simulation
DO	Dissolved oxygen
DWT-SG	Double wall tube steam generator
EE	Explicit elicitation
ELFR	European lead fast reactor
ESFR	Example sodium fast reactor
ETPP	European test pilot plant
EVOL	Evaluation and viability of liquid fuel fast reactor system (Euratom FP7 Project)
FHR	Fluoride-salt-cooled high-temperature reactor
FOAK	First of a kind
GTHTR300C	Gas turbine high temperature reactor 300 for cogeneration
GT-MHR	Gas turbine-modular helium reactor
HEC	High efficiency channels
HPLWR	High performance light water reactor
HTGR	High-temperature gas-cooled reactor
HTR-PM	High-temperature gas-cooled reactor power generating module
HTR-10	High-temperature gas-cooled test reactor with a 10 MW_{th} capacity
HTSE	High temperature steam electrolysis
HTTR	High temperature test reactor
IASCC	Irradiation assisted stress corrosion cracking
IHX	Intermediate heat exchanger
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
IRRS	Integrated Regulatory Review Service
ISTC	International Science & Technology Centre
IVTM	In-vessel transfer machine (Monju)
JSFR	Japanese sodium-cooled fast reactor
KALIMER	Korea advanced liquid metal reactor
LOCA	Loss-of-coolant accident
LWR	Light water reactor
M&M	Measures and metrics
MA	Minor actinides
MCST	Maximum fuel cladding surface temperature
MSFR	Molten salt fast reactor
NGNP	New generation nuclear plant
NHDD	Nuclear hydrogen development and demonstration
NPP	Nuclear power plant
NSRR	Nuclear safety research reactor (Japan)
ODS	Oxide dispersion-strengthened

Technical terms *(cont'd)*

PBMR	Pebble bed modular reactor
PDC	Plant dynamics code
PHWR	Pressurised heavy water reactor
PIE	Post irradiation examinations
PWR	Pressurised water reactor
PYCASSO	Pyrocarbon irradiation for creep and shrinkage/swelling on objects
R&D	Research and development
RF-ECT	Remote field eddy current testing
RIA	Reactivity-initiated accident
RPV	Reactor pressure vessel
SCC	Stress corrosion cracking
SCW	Supercritical water
SCWL	Supercritical water loop (in Řež)
SMART	System-integrated modular advanced reactor
SMFR	Small modular fast reactor
SMR	Small modular reactor
SOEC	Solid oxide electrolyser cell
SS	Stainless steel
SSTAR	Small, sealed, transportable, autonomous reactor
STELLA	Sodium integral effect test loop for safety simulation and assessment
SWR	Sodium water reaction
THTR	Thorium high temperature reactor
TRISO	Tristructural isotopic (nuclear fuel)
TRU	Transuranic
YSZ	Yttrium-stabilised zirconia

Organisations, programmes and projects

ANRE	Agency for Natural Resources and Energy (Japan)
ANS	American Nuclear Society
ARC	DOE Office of Advanced Reactor Concepts (United States)
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)
CNSC	Canadian Nuclear Safety Commission
DoE	Department of Energy (Republic of South Africa)
DOE	Department of Energy (United States)
EC	European Commission
ENSI	Swiss Federal Nuclear Safety Inspectorate
EU	European Union
FP7	7th Framework Programme
IAEA	International Atomic Energy Agency
ICN	Institute of Nuclear Research (Romania)
IFNEC	International Framework for Nuclear Energy Cooperation
INL	Idaho National Laboratory (United States)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre (Euratom)
KAERI	Korea Atomic Energy Research Institute
KIT	Karlsruhe Institute of Technology (Germany)

Organisations, programmes and projects *(cont'd)*

MDEP	Multinational Design Evaluation Programme
MEST	Ministry of Education, Science and Technology (Republic of Korea)
MOST	Ministry of Science and Technology (China)
MS	Member states
NEA	Nuclear Energy Agency (OECD)
NEAC	Nuclear Energy Advisory Committee (United States)
NETC	Nuclear Energy Technical Committee (Republic of South Africa)
NNEECC	National Nuclear Energy Executive Coordination Committee (Republic of South Africa)
NRC	Nuclear Regulatory Commission (United States)
NRCan	Department of Natural Resources (Canada)
NRF	National Research Foundation (Republic of Korea)
NRI	Nuclear Research Institute (Czech Republic)
NSSC	Nuclear Safety and Security Commission (Republic of Korea)
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory (United States)
PBMR Pty	Pebble Bed Modular Reactor (Pty) Limited (Republic of South Africa)
PSI	Paul Scherrer Institute (Switzerland)
SNL	Sandia National Laboratories (United States)

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This seventh edition of the *Generation IV International Forum (GIF) Annual Report* highlights the main achievements of the Forum in 2013, and in particular the progress made in the collaborative R&D activities of the ten existing project arrangements for the gas-cooled fast reactor, the sodium-cooled fast reactor, the supercritical-water-cooled reactor and the very-high-temperature reactor. Progress made under the Memoranda of Understanding (MOU) for the lead-cooled fast reactor and the molten salt reactor, including Russia's signing of the latter MOU in November 2013, is also reported. The Phase 1 report on safety design criteria for the sodium-cooled fast reactor was published in May 2013, marking an important milestone in the development of safety standards for generation IV reactors. Finally, the *Technology Roadmap Update for Generation IV Nuclear Energy Systems* was completed in 2013, providing both an assessment of progress made in the development of Generation IV systems since the publication of the original *Technology Roadmap* in 2002 and an overview of the GIF's key objectives for the next ten years.

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