ANNUAL REPORT 2014

GEN IV International Forum℠

Foreword from the GIF Chair

It is once again my privilege to present the Generation IV International Forum (GIF) Annual Report, our flagship publication that offers an update on the achievements of collaboration under the GIF Framework. In 2014, progress was made on five of the six nuclear systems, on the implementation of evaluation methodologies, and on a new course of outreach to the international regulatory community that will eventually license generation IV reactors. The 2014 *Technology Roadmap Update* was published and significant progress was made on implementing our latest strategic plan. In addition, exciting design and construction projects in GIF member countries were closely followed, many of which are documented in this report.

The 2014 GIF Annual Report focuses on progress made during the calendar year. For more background, other publications can be consulted, such as the GIF-organised special issue of Journal of Progress in Nuclear Energy, (www.sciencedirect.com/science/journal/01491970/77) published in November 2014. While the gas-cooled fast reactor was not represented in the journal publication, the other five systems were described in sufficient detail to trace back to the earliest days of development for each technology. The third GIF Symposium, will be held in co-ordination with the 23rd International Conference on Nuclear Engineering on 19 May 2015 at Makuhari Messe, Chiba, Japan, and will include a broad summary of all GIF technical activities.

The Generation IV International Forum, now in its second decade, was formed by a multi-national agreement among countries that recognised that the long-term future of nuclear energy depended on developing the next generation of reactors that would be even safer and more efficient while continuing to meet economic and non-proliferation goals. Currently, the active members in GIF are Canada, the European Atomic Energy Community (Euratom), France, Japan, Korea, the People's Republic of China, the Russian Federation, the Republic of South Africa, Switzerland and the United States. Argentina, Brazil and the United Kingdom are inactive members, but remain cognisant of the forum's activities. GIF members signed the extended Charter in 2011 and are currently signing the extended Framework Agreement, which entered into force on the 28 February 2015.

GIF maintains a long-standing collaborative relationship with the International Atomic Energy Agency (IAEA) with emphasis on 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. GIF and INPRO held their eighth interface meeting in March 2014 to discuss areas of mutual interest related to technology status, including the convergence of assessment methodologies in the area of economics, safety or proliferation resistance and physical protection.

Over the last few years, a GIF task force has developed Safety Design Criteria (SDC) for the sodium fast reactor, which is likely to be one of the first generation IV systems to be demonstrated. The SFR SDC report was distributed for external review to national regulators and international organisations, including the Multinational Design Evaluation Programme (MDEP), the Nuclear Energy Agency (NEA), and the IAEA. During 2013, the report was presented at a joint SFR safety workshop organised by IAEA/INPRO and GIF in Vienna, Austria. In June 2014, a follow-up workshop attended by reactor designers, regulators and safety experts delved more deeply into this subject, exploring implementation guidelines for the safety criteria. On 1 December 2014 in Paris, France, the GIF presented the SFR SDC to a meeting of the NEA Committee on

Nuclear Regulatory Activities (CNRA). The outcome was the initiation of an ad hoc group that will co-ordinate international regulatory discussion on licensing of generation IV reactors.

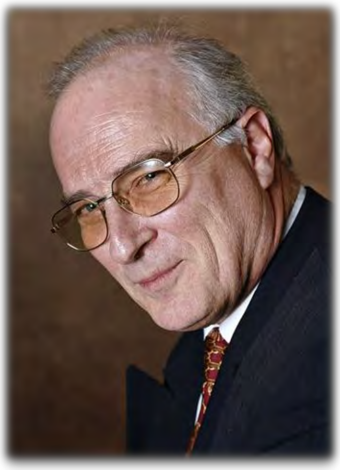
GIF has four overarching goals, three of which are the subject of methodology working groups on safety, non-proliferation and economics. In 2014, preparations began to organise a task force to look into sustainability aspects. As of this writing, the Sustainability Task Force (STF) is being organised and has begun to develop a work plan. The STF will initially focus on reviewing internationally recognised definitions and metrics for nuclear sustainability, particularly those developed by INPRO.

2015 marks the tenth anniversary of the signing of the Framework Agreement that allowed collaborative research and development to be organised under the GIF banner. On 26 February 2015, the Framework Agreement was extended for another ten years. The next step will be to extend the System Arrangements to allow research on the different systems to continue well into the next decade.

Sincerely,

Dr. John E. Kelly
GIF Policy Group Chairman

A tribute to Jacques Bouchard



It is with great sadness that the Generation IV International Forum learnt of the passing of Mr Jacques Bouchard on 30 January 2015 at the age of 76.

Jacques Bouchard had a long career with the French Atomic Energy Commission (CEA, now the Alternative Energies and Atomic Energy Commission), which he joined in 1964. In 1982, Mr Bouchard became the Head of the Department of Fast Neutron Reactors in Cadarache, in 1990 the Director of the Nuclear Reactor Division and then from 2000 to 2004, the Director of the newly established Nuclear Energy Division, which brought together the nuclear fuel cycle and reactor divisions of the CEA.

With his enthusiastic vision of the future of nuclear energy, Jacques Bouchard was also deeply convinced of the value of international collaboration. Along with William D. Magwood, IV, he co-founded the Generation IV International Forum, and he also served as the second GIF Chair from 2006 to 2009. During this period, GIF membership increased from the six member countries that had signed or acceded to the Framework Agreement in 2005 (Canada, France, Japan, Korea, Switzerland and the United States) to the current ten members, with the accession of Euratom in 2006, China in 2007, South Africa in 2008 and Russia in 2009.

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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, 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 2014**

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	
Korea (KR)	Ministry of Science, ICT and Future Planning (MSIP) and National Research Foundation (NRF)	08/2005			04/2006	11/2006		
People's Republic of China (CN)	China Atomic Energy Authority (CAEA) and Ministry of Science and Technology (MOST)	12/2007		05/2014	03/2009	10/2008		
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.

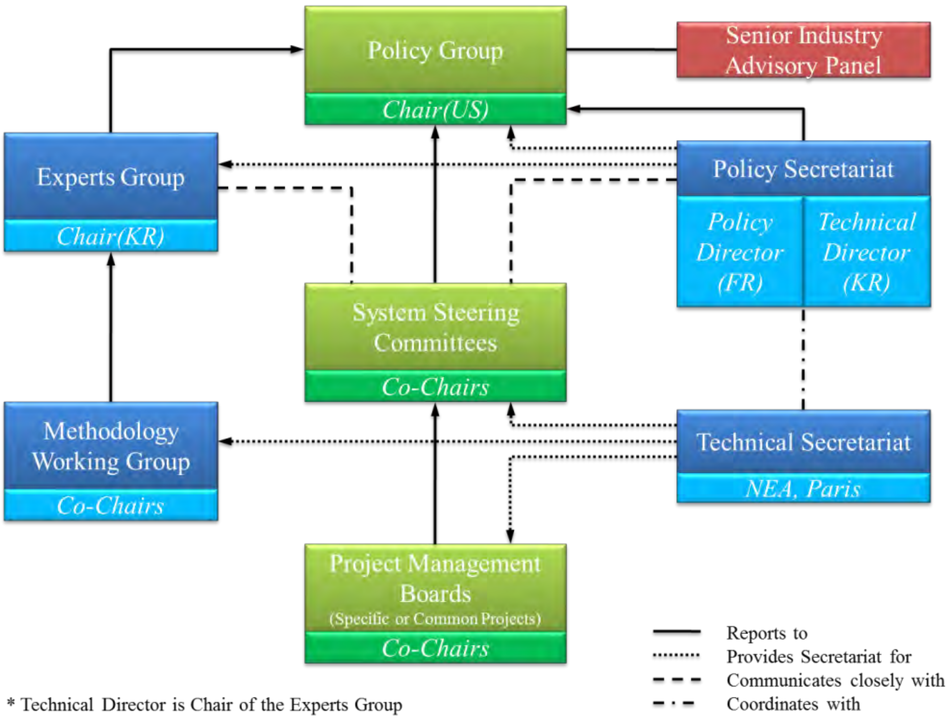
Among the signatories to the charter, ten members (Canada, France, Japan, China, Korea, the Russian Federation, South Africa, Switzerland, the United States and Euratom) have signed or acceded to the framework agreement (FA) as shown in Table 1.1. Parties to the FA formally agree to participate in the 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 agent 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. In May 2014, China signed the System Arrangement for the SCWR, which has now five members (Canada, Euratom, Japan, Russia and China). The participation of GIF members in SAs and MOU is also shown in Table 1.1.

1.2 GIF organisation

The GIF charter provides a general framework for GIF activities and outlines its organisational structure. Figure 1.1 is 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 2014**



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 twice a year. In 2014, the two PG meetings were both held in Paris, the first one was hosted by CEA (France) in May, the second one was hosted by the Nuclear Energy Agency (NEA) at the OECD Conference Centre (Figure 1.2) in December.

Figure 1.2: **Policy group in Paris at the OECD Conference Centre (December 2014)**

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 also usually meets twice a year. The meetings are held back-to-back with the 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 oversee the project activities described in a detailed multi-annual Project Plan (PP) that aims 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 and the drafting of a PP. R&D carried out under an MOU (case of 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) – the Economic Modelling Working Group (EMWG), the Proliferation Resistance and Physical Protection Working Group (PRPPWG), and the Risk And Safety Working Group (RSWG) – 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. The MWGs 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 can create 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 sustainability.

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 the PG on policy matters whereas the technical director serves as chair of the EG and assists the PG on technical matters. The technical secretariat, provided by the NEA, supports the SSCs, PMBs, MWGs and TFs and maintains the public and password-protected websites. 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 to support 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 consisting of specific tasks to be performed by the signatories.

In terms of PAs, a new PA for the SFR System Integration and Assessment (SIA) became effective in October 2014, with partners CEA, CIAE, DOE, JAEA, JRC, KAERI, and ROSATOM. China joined the VHTR Fuel and Fuel Cycle PA in January 2014, and is in the process of joining the VHTR Materials and the VHTR H_2 production PAs. China is also engaged in discussion with the PMB members of both the SCWR PA on Thermal-hydraulics and Safety and the SCWR PA on Materials and Chemistry regarding its contribution and with a view to join these two projects.

Table 1.2 shows the list of signed arrangements and provisional co-operation within GIF as of 31 December 2014.

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 collaboration 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 2014**

	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	S	X			O	X
FFC PA	30-Jan-08		X	X	X	X	X				X
MAT PA	30-Apr-10		X	X	X	S	X			X	X
CMVB PA	Provisional		P		P	P	P			O	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	22-Oct-14		X	X	X	X	X		X		X
SCWR SA		X	X		X	X			X		
M&C PA	6-Dec-10	X	X		X	O			O		
TH&S PA	5-Oct-09	X	X		X	O			O		
SIA PA	Provisional	P	P		P	P			P		
FQT PA	Provisional	P	P		P	P			P		
GFR SA			X	X	X					X	
CD&S PA	17-Dec-09		X	X						X	
FCM PA	Provisional		P	P	P					P	
LFR MOU			X		X	O	O		X		O
MSR MOU			X	X	O	O	O		X		O

X = SIGNATORY **P** = PROVISIONAL PARTICIPANT **O** = OBSERVER **S** = SIGNATURE PROCESS ONGOING

PROJECT ACRONYMS

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

Chapter 2. Highlights from the year and country reports

2.1 General overview

In 2014, nuclear energy remains one of the most effective low carbon sources of electricity and an essential part energy policy planning in many countries. Yet, even as Generation III reactors are becoming the technology of choice for new nuclear build, the interest in Generation IV reactors remains high to meet expected requirements in terms of safety, sustainability, economics and proliferation resistance. The newly released IEA/NEA Nuclear Technology Roadmap[3], which gives an overview of nuclear development up to 2050, including policy measures to support the place of nuclear energy in a low carbon scenario, recommends continuous support for R&D for Generation IV energy systems, as well as support for the construction of prototypes to open the way for longer-term commercial deployment.

The GIF has continued to work actively on the safety of Generation IV systems, in particular on the safety of the SFR, with the help of a dedicated task force in addition to the Risk and Safety Working Group (RSWG). The GIF also engaged discussions with regulators at national level as well as with international organisations. Several interactions with regulators took place in 2014, including regulators of countries pursuing the development of SFRs, the Multinational Design Evaluation Programme (MDEP), the NEA Committee on Nuclear Regulatory Activities (CNRA), and the Committee on the Safety of Nuclear Installations (CSNI). Building on the technical work provided by the task force, interactions with regulators will continue in 2015.

Regarding the sustainability aspect, GIF is reconsidering the need for a specific Methodology Working Group in the area of sustainability. An interim task force was launched to prepare the basis for such a working group. With the support of the Experts Group, the work scope of the task force was established and specific actions were developed. The goal is to develop a methodology to assess the Generation IV reactor systems and to compare the GIF approach with other work in this area, such as that from IAEA or the US Department of Energy (DOE).

Funding for some of the Generation IV reactor systems has decreased, thereby slowing down progress. This is the case in particular of the gas-cooled fast reactor system. An action is ongoing among the signatories of that SA to discuss ways to continue R&D, possibly by seeking synergies with other systems.

At the same time, China signed the SCWR SA in May 2014 and a new project, the SFR System Integration and Assessment project arrangement, became effective in October 2014, showing that GIF remains an attractive forum for collaborative R&D.

2.2 Highlights from the experts group

In 2014, two EG meetings were held in Paris, one hosted by CEA in May, and the other hosted by the NEA in December. One of the main topics of discussion was the "Sustainability Methodology" for which a working group has not yet been established, although one of the technology goals for Generation IV systems is sustainability. The EG concluded in May that clarification was needed

3. IEA/NEA (2015), *Technology Roadmap: Nuclear Energy*, www.oecd-nea.org/pub/techroadmap/techroad map-2015.pdf.

on the definition of sustainability and on the scope of work that the GIF could address. Based on the EG's recommendations, the PG approved the creation of an interim task force to develop a draft action plan as a starting point. At the meeting in December, the EG provided a draft action plan for the interim task force with specific actions in five categories and the task force began to work on it with the PG's approval. The status of actions will be reported on at the next PG meeting in May 2015.

One of the EG's main responsibilities is to monitor the activities of the methodology working groups and the R&D on the different systems. Each methodology working group system has reported on the status of progress during the EG meetings. This year, the EG approved the updated terms of references submitted by each MWG to harmonise the timeline and work scope. The EG discussed the future path for the GFR for which there is very little funding available. The EG proposed to explore possible synergies with other systems, such as the VHTR; a joint meeting between GFR and VHTR System Steering Committees is planned in 2015.

Shared interests between GIF and INPRO were reviewed and discussed at the 8th GIF-INPRO interface meeting which was co-ordinated by the EG and the IAEA. The objectives of the meeting included an exchange of information on the progress, status and future plans of activities in the development of Generation IV reactors, reviewing joint actions, and identifying new areas for potential co-operation.

As a follow-up to the development of Safety Design Criteria (SDC) for the SFR, the Task Force is developing Safety Design Guidelines for which the EG is continuously providing high-level review.

Finally, each of the actions initiated under the recent strategic planning activity, namely project planning, sharing capabilities and resources, communications, and engagement with the SIAP, were reported at the EG meetings. Significant progress has been made in a number of those activities.

2.3 Country reports

Canada

Energy policy

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

In Canada, nuclear energy falls within the jurisdiction of the federal government. 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.

Nuclear fleet

In 2013, nuclear energy provided close to 15% of Canada's total electricity needs (over 55% in the Province of Ontario) and will continue to play an important role in supplying Canada with power in the future. As of today, Canada has a fleet of 19 power reactors in commercial operation. One reactor is located in the Province of New Brunswick, and the others 18 power reactors are located in the Province of Ontario.

The Government of Ontario plans to refurbish all four reactor units at the Darlington power plant, and six units at the Bruce power plant. Refurbishment will add about 25-30 years to the operational life of each unit.

Uranium production

Canadian uranium production totalled 9 331 tU in 2013, 16% of the total world production. All current Canadian uranium production in Canada is from mines located in the northern part of the Province of Saskatchewan.

Nuclear liability

In January 2014, the Government of Canada began the legislative change process to increase operator liability for nuclear damage from USD 75 million to USD 1 billion. The legislation will align Canada's nuclear liability regime with international conventions. This legislative change will permit Canada to implement the Convention on Supplementary Compensation for Nuclear Damage, by the International Atomic Energy Agency (IAEA), that it signed in December 2013. The legislation is expected to be finalised by Parliament in the near future.

Waste management

There are two deep geological repositories (DGRs) being considered in Canada. Ontario Power Generation (OPG) is proposing to prepare, construct and operate a DGR on the Bruce Nuclear Site within the Municipality of Kincardine, in the province of Ontario. The DGR would manage low-level and intermediate-level radioactive waste produced during operation of the Bruce, Pickering and Darlington nuclear power plants in Ontario, which are owned by OPG. The proposal is currently undergoing a federal environmental review assessment.

The Nuclear Waste Management Organisation (NWMO), established in 2002 to assume responsibility for long-term management of Canada's used nuclear fuel, is seeking an informed and willing community with a suitable site to eventually prepare, construct, and operate a DGR for the long-term management of nuclear fuel waste. The NWMO is currently evaluating 13 participating communities. One community is in the Province of Saskatchewan, and the others are in the Province of Ontario.

Emergency planning

In May 2014, Canada conducted a national emergency response field exercise, based on a simulation of severe nuclear emergency. The exercise, led by OPG, involved over 50 organisations and over 2 000 participants, including 19 federal Government of Canada departments, and IAEA observers. The initiative exercised the preparedness of utilities, governments, non-government agencies, and local communities to respond to nuclear accidents. The Department of Health Canada is now developing an action plan for continuous learning and improvement, featuring an ambitions five-year exercise schedule.

Supercritical-water-cooled reactor progress

In 2014, Canada's national GIF programme continued to focus on the development of the Canadian SCWR concept. Part of the Canadian GIF programme applied methodologies developed by the GIF cross-cutting working groups in the areas of safety, economics, and proliferation resistance. A review of the Canadian SCWR concept has been scheduled in 2015 to engage the Canadian nuclear industry.

China

Nuclear energy policy

China adheres to the policy of developing nuclear power in a safe and efficient manner. Nuclear energy will play a more important role in national energy supply in the future. Following the Nuclear Safety and Radioactive Pollution Prevention and Control Programme for the 12th Five-Year Plan Period and Long-term Goals by 2020, the Nuclear Power Safety Programme (2011-2020) and the medium- and long-term Nuclear Power Development Programme (2011-2020) approved by the State Council in October 2012. The State Council published the Energy Development Strategy Action Plan (2014-2020) on 19 November 2014, which aims to reduce China's dependence on coal, and promote the use of clean energy. The action plan requires that new nuclear power projects should be launched at appropriate times under the precondition that the world's highest safety standards be adopted; meanwhile, feasibility studies for the construction of inland plants should be carried out. According to the Action Plan, the total installed nuclear generating capacity in operation will reach 58 GW by 2020 and a further 30 GW or more new capacity will be under construction by that time.

President Xi-Jinping, stated in his speech at the Nuclear Security Summit held in the Hague, the Netherlands, in March 2014, that "we should place equal emphasis on development and security, and develop nuclear energy on the premise of security. The peaceful use of nuclear energy is important to ensuring energy security and tackling climate change. We must strictly abide by the principle of making safety the top priority if we are to keep the flame of hope for nuclear energy development burning forever".

Nuclear safety is essential in the development of nuclear energy. "Safety first, quality first" has consistently been the fundamental policy of nuclear industry in China. The National Nuclear Safety Administration (NNSA) issued general technical requirements for corrective actions of NPPs in the wake of the Fukushima Daiichi accident, requesting corrective actions to be made to further improve the level of safety of NPPs in eight areas that include flood control, emergency water supply, portable power supply, hydrogen monitoring and control, radiation monitoring and emergency response. The corresponding safety upgrade actions are performed at the relevant nuclear power facilities based on areas for improvement identified during comprehensive safety inspections.

To further improve nuclear safety management, the drafting of the Nuclear Safety Law is under way.

Operation and construction of nuclear power plants

The nuclear power units in operation have kept a good record of safety, and the projects under construction are progressing according to schedule. By the end of November 2014, there were 21 nuclear power units in commercial operation in China mainland, with the installed capacity of 19.21 GW (including Fuqing Unit 1). Twenty seven nuclear power units were under construction (including Fangjiashan Unit 1, which was connected to the grid on 4 November 2014), with the total installed capacity of 29.66 GW. Construction continues on four AP1000 reactors at two projects in Sanmen and Haiyang, which represent a new generation of nuclear power plants based on passive safety concepts. Two EPR reactors are also being constructed on schedule at the Taishan site.

VHTR R&D

R&D on high-temperature gas-cooled reactor (HTR) has made encouraging progress. In December 2012, the construction of the high-temperature reactor HTR-PM demonstration plant started in Shandong Province. According to the current schedule, HTR-PM will be connected to the grid by the end of 2017. The peak of equipment installation will occur in 2015. HTR-PM adopts a steam cycle power conversion system, with the steam turbine system driven by two reactor modules. HTR-PM will demonstrate the technological maturity and near-term market potential of VHTR.

In October 2013, the China Atomic Energy Authority (CAEA) agreed to the amendment to the project arrangement on fuel and fuel cycle (FFC) for the international R&D of VHTR nuclear energy system and gave approval to the Institute of Nuclear and New Energy Technology (INET) of Tsinghua University to join the project. So far China has joined or is in the process of joining all PAs in the VHTR system, and is providing research proposals and submitting deliverables. Substantive studies on coated particle fuel, high-temperature steam electrolysis, material and software development etc. have been performed.

Themed as "Modular High Temperature Reactor: A Coming Reality" the 2014 International Conference on High Temperature Reactor Technology (HTR-2014) was organised by the Institute of Nuclear Energy Tsinghua University in October 2014 in Shidao Bay, Weihai, China. The conference venue was in close proximity to the construction site of the 2-module nuclear demonstration plant, HTR-PM. More than 400 experts from nearly 20 countries, regions and international organisations gathered and exchanged the latest information on industrial, economic, energy policy and research topics related to high-temperature reactor technology. Over 180 presentations were made at the conference. The well-attended HTR-2014 conference reflects the growing international interest in the development of safe, clean and sustainable nuclear energy.

Sodium-cooled fast reactor (SFR) R&D

Research and development on sodium-cooled fast reactor has made progress. The China experimental fast reactor (CEFR) was restarted on 14 March 2014 and completed load rejection test at 75% power level on 29 November 2014. CEFR reached 100% power level on 15 December 2014. Since then, it has continuously been operating for 72 hours. It successfully generated electricity for about 1.80 million kWh by 18 December 2014. During this period of time, a series of experiments on reactor physics, safety, control system, thermal hydraulics, etc. were performed. According to the China fast reactor development plan, the concept design of the demonstration project for the China fast reactor (CFR600) has been initiated. China is taking part in the R&D project on system integration and assessment after joining the safety and operation project for the international R&D of SFR nuclear energy system.

Super-critical water-cooled reactor (SCWR) R&D

China acceded to the GIF System Arrangement for the International Research and Development of the Super-critical Water-Cooled Reactor Nuclear Energy System in May 2014. The Nuclear Power Institute of China (NPIC) and Shanghai Jiaotong University are authorised to join the TH&S PA and M&C PA respectively as representatives of China's SCWR research consortium. According to the requests of the SCWR System Steering Committee, China has provided project plan proposals for the TH&S and M&C projects.

Research and development on SCWR has been carried out. Several R&D activities have been started by different universities and institutes supported by several national and enterprise projects. The research interests include SCWR design, thermal-hydraulics behaviour, materials and water chemistry, related research on fuel qualification tests, such as the heat transfer to water at super-critical pressures in tubes, annuli and bundles, and 2×2 Rod bundles, the research on turbulence models for flow and heat transfer under super-critical water in single channel and fuel assemblies, the experiment of flow instability in parallel channels under SCW conditions, natural circulation experiments in a simple rectangular loop under SCW conditions, stress corrosion cracking, development of novel materials for fuel cladding, the water chemistry behaviour under SCW conditions etc. Several computer codes have been developed which are used for safety analysis and steady-state nuclear T/H coupling analysis as well as flow instability analysis and a lot of CFD computations have been carried out in universities and institutes.

The conceptual design and relevant R&D studies of the first phase of the development of the Chinese SCWR, called CSR1000, have been completed. The assessment for the "R&D on SCWR technology at phase-1 stage" funded and organised by CAEA with the aim of developing an industrial level SCWR design, was conducted by an expert group in November 2013. The China-Europe co-operation project on fuel qualification and testing activities, the SCRIPT project, is underway. China is going to start the project-R&D on SCWR technology (phase II), aimed at finishing the design of the experimental reactor of CSR1000 and tackle problems in key technologies such as the thermal-hydraulic characteristics, system safety behaviour, material optimisation and design of fuel element irradiation test devices, etc.

Several test facilities operating with super-critical water have also been set up in China to satisfy the requirements of SCWR R&D for thermal-hydraulic, materials and water chemistry. The maximum parameters of thermal-hydraulic test facilities is up to 30 MPa, 550°C, 32 m^3/h, the maximum parameter of materials and water chemistry test facilities is up to 30 MPa, 650°C, 2.5 L/h. A great deal of experimental data has been obtained from these systems and equipment.

Lead-cooled fast reactor (LFR) R&D

The Chinese Academy of Sciences (CAS) has launched the ADS Project, and plans to construct a demonstration ADS transmutation system by the 2030s in three stages. China LEAd-based Reactor (CLEAR) is selected as the reference reactor for ADS project and for LFR technology development. The Institute of Nuclear Energy Safety Technology (INEST) of CAS has completed the detailed conceptual design for a 10 MWt lead-based Research Reactor called CLEAR-1. The preliminary engineering design is underway. A large Pb/Bi experiment loop (KYLIN Series) has already been set up. In 2014, INEST, as an observer, participated in a series of activities of the GIF LFR PSSC including preparation of documents and guidelines such as the System Research Plan for the Lead-cooled Fast Reactor, Lead-cooled Fast Reactor (LFR) Risk and Safety Assessment White Paper as well as the drafting of Safety Design Criteria for Generation IV Lead-cooled Fast Reactor System. Meanwhile, INEST hosted the 16th GIF LFR PSSC meeting in 10-11 December 2014 in Hefei city, China.

Molten salt reactor (MSR) R&D

In 2011, the Chinese Academy of Sciences initiated the "Thorium Molten Salt Reactor (TMSR) Nuclear Energy System" project. The aims of TMSR are to develop Th-based energy systems, including non-electric application of nuclear energy based on TMSR systems with liquid fuels (LF) and TMSR systems with solid fuels (SF) in the next 20-30 years. In 2014, the Shanghai Institute of Applied Physics of CAS completed the concept design of the 10 MW TMSR solid fuel prototype (SF-1), pre-concept design and its international peer review of the 2 MW TMSR liquid fuel prototype LF1. As an observer, the Shanghai Institute of Applied Physics participated in a series of activities of the GIF MSR and hosted the pSSC meeting at the end of April in Shanghai, China.

Euratom

EU 2014 nuclear safety directive

In order to keep nuclear installations safe and enhance European leadership in nuclear safety worldwide, the EU amended the 2009 Nuclear Safety Directive on 8 July 2014. The amendment is based on the lessons learnt from the Fukushima Daiichi nuclear accident, EU nuclear stress tests, and the safety requirements of the Western European Nuclear Regulators Association (WENRA) and the International Atomic Energy Agency (IAEA).

The new Directive:

- strengthens the power and independence of national regulatory authorities;
- introduces a high-level EU-wide safety objective to prevent accidents and avoid radioactive releases while promoting an effective nuclear safety culture;
- sets up a European system of peer reviews by competent regulatory authorities on specific safety issues every six years;
- increases transparency on nuclear safety matters by informing and involving the public;
- promotes an effective nuclear safety culture.

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

Euratom Horizon 2020 framework programme

The Interdisciplinary Study on "Benefits and limitations of nuclear fission for a low-carbon economy" (February 2013) and the FISA and Euradwaste symposia (October 2013) have identified major themes for the Horizon 2020 research programme. These are: nuclear safety and safety culture; socio-economics in the overall energy mix; civil society and education and training with main orientations being challenge oriented and emphasis on impact.

A new seven-year EU research programme called Horizon 2020, starting in 2014, has been agreed and adopted by the EU Parliament and the EU Council. The Horizon 2020 Euratom

Programme for nuclear research and training activities (which is a five-year programme), supports the EU research in nuclear fission and fusion including Generation IV research. As far as research on Generation IV is concerned, the priority is on safety and security issues.

The first cluster on the 2014-15 call for proposals, "Support Safe Operation of Nuclear Systems", included themes linked to Generation IV such as "Improved safety design and operation of fission reactors" and "new innovative approaches to reactor safety". The outcome of the evaluation of the above 2014-2015 call will be published in March 2015.

SNETP/ESNII

The EU has committed for the year 2020 to: reduce by 20% its greenhouse gas 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 (SNETP) promotes research, development and demonstration of nuclear fission technologies.

The European Sustainable Nuclear Industrial Initiative, ESNII, is one pillar of the SNETP, a platform dedicated to nuclear energy technologies. ESNII is devoted to fast neutron reactors with closed fuel cycles, which represent a sustainable version of nuclear energy systems thanks to a better use of the uranium resource through plutonium breeding and recycling. Presently, ESNII includes the study of three types of technologies that could lead to future industrial deployment: SFR, LFR, and GFR, thanks to different domestic projects of construction of demonstration plants promoted by some European countries: ASTRID an SFR in France, ALFRED a LFR in Romania, and ALLEGRO a GFR in central Europe (Czech Republic, Slovak Republic, Hungary, Poland).

ALFRED

A consortium has been formally set up in December 2013 for the construction of a demonstration lead-cooled fast reactor in Romania. The conceptual design of the ALFRED reactor and the integrated project were led by Ansaldo Nucleare (IT). A memorandum of technical co-operation has been signed between Italy's National Agency for New Technologies, Energy and the Environment (ENEA) and Ansaldo Nucleare, as well as Romania's Nuclear Research Institute (Institutul de Cercetari Nucleare, RATEN-ICN), to implement the construction of ALFRED. A new partner, the CV-REZ research centre located close to Prague (Czech Republic) joined in December 2014. The group is to be known as the "Fostering Alfred Construction" (Falcon) consortium. The total cost of the project is estimated at some EUR 1 billion. Alfred is seen as a prelude to an industrial demonstration unit of about 600 MWe. The lead-cooled reactor will employ mixed-oxide (MOX) fuel and will operate at temperatures of around 500°C. It features passive safety systems. The demonstration Alfred unit will be built at ICN's facility in Mioveni, near Pitesti in southern Romania, where a fuel manufacturing plant is in operation for the country's two operating Candu reactors.

The overall long-term plan of Alfred includes two phases. The first step (2015-2020) focused on ALFRED experimental infrastructures, experiments and design with a strong effort in technology development; a second step (2020-25) for the construction of ALFRED (the unit could start operating in 2026).

ALLEGRO

The ALLEGRO project started in 2010 by the signature of an MOU by UJV (Czech Republic), VUJE (Slovak Republic) and AEKI/EK (Hungary), with an essential scientific support of CEA (France). Poland’s NCBJ joined in 2012. The project started with a preparatory phase (2010-13) aiming at the preparation of documents needed for the decision of the three governments, the licensing and construction and the operational phases will start later (the hosting of the facility will most likely be in the Slovak Republic). The new legal entity “V4G4 Centre of Excellence” (Visegrad 4 countries for Generation 4 reactors) was introduced to the public at the Hungarian Academy of

Sciences on 18 July 2013. The new strategy for the ALLEGRO reactor was set in April 2014 (reduce ALLEGRO power from 75 MWth to circa 10 MWth and find an optimum core configuration; increase main blowers inertia; as a consequence of potential fuel supply difficulties, use UO_2 pellets in AIM1 cladding instead of MOX pellets).

Within the ALLEGRO project the tasks and responsibilities are shared as follows: Hungary monitors and analyses the closed fuel cycle and fuel issues – safety-related activities will be financed from a new National Nuclear Energy Research Programme; the Czech party performs the technological utilisation of high-temperature gas testing; the Slovakian party deals with planning and security testing and the preparatory works for designing and establishing a non-nuclear mock-up of ALLEGRO to be financed in 2014-2015. The Polish party will be responsible for the material testing. The Czech Sustainable Energy Project (SUStainable ENergy (SUSEN), funded by the European Regional Development Fund/ERDF/) is implemented as a regional R&D centre located at the CVR, and is involved in ALLEGRO.

EC funding for Euratom indirect actions

Since the 5th Framework programme (FP5) in 1998, the global EC funding for Euratom indirect actions (R&D projects performed by consortia of European organisations) has been almost constant in constant Euro value. In FP5, a topic "Safety and efficiency of future systems" was clearly identified; in FP6 there was a sub-topic "Innovative concepts" in the domain "Other activities of nuclear technology and safety"; finally in FP7, one can find the topic "Potential of advanced nuclear systems" which became more precise for the last calls in 2011 and 2012 in the extension part of FP7 Euratom, with sub-topics such as "Generation IV nuclear systems and ESNII", "Crosscutting activities and ESNII", and "Advanced reactor systems". There is a tendency to better organise the European efforts, and not only to aggregate disjoint domestic programmes. Euratom indirect actions related to GIF collaborative programmes have increased over the years (7% under FP5, 20% under FP6 and 26% under FP7).

The following summarises the budgets linked to the ESNII Implementation Plan 2013-15.

ESNII-1 (SFR): ASTRID project (Advanced Sodium Technical Reactor for Industrial Demonstration); Budget: 2010-2012 – EUR 198.1 m; 2013-2014 – EUR 210.4 m; 2015-2017 – EUR 243.1 m; Total: EUR 651.6 m

ESNII-2 (Fast spectrum research reactor): MYRRHA Project (Multipurpose Hybrid Research Reactor for High-tech Applications); Budget: 2013 – EUR 30 m; 2014 – EUR 40 m (including EC contribution); 2015 – EUR 61 m (estimate). Total: EUR 131 m

ESNII-3 (LFR): ALFRED Project (Advanced Lead Fast Reactor European Demonstrator). Budget for Programme LFR, ALFRED, ELECTRA: 2013-2015 – EUR 41.9 m (including EC contribution of EUR 2.7 m)

ESNII-4 (GFR) ALLEGRO Project. Budget: 2013 – EUR 5 m; 2014: EUR 8.5 m; (including EC RTD contribution) 2015 – EUR 11 m (estimate). Total: EUR 24.5 m

France

Energy transition act

The energy bill was approved on 14 October by the French national assembly. It is scheduled to be examined by the Senate in the first quarter of 2015, and then will be formally adopted.

In its actual version, the main objectives are:

- Reduction of the use of fossil resources by 30% by 2030, of greenhouse gas emissions by 40% also by 2030, and the halving of the overall energy consumption in 2050 compared to 2012 level.
- Capping the installed nuclear capacity to the current level (63 GWe), and decreasing the share of nuclear electricity from 75% to 50% around 2025.

- Decision of shutting down nuclear power plants left to the operator (EDF), in accordance with the decisions of the Safety Authority.
- Follow-up of the Energy Policy, based on a 5-year actualised plan.

Evolution of the governance of major French industrial and institutional actors

- Regarding the operator EDF, M. Jean-Bernard Levy, former Chairman and CEO of Thales has been nominated as the new CEO to replace M. Proglio.
- Regarding AREVA, the company is in the process of changing its governance, moving from a structure with two boards, (supervisory board and executive one) to a simpler one, with a single board of directors. M. Philippe Varin, former Peugeot CEO, is foreseen to take the helm of this board, the official nomination being scheduled in the beginning of January.
- Regarding CEA, it has been announced that M. Bigot, General Administrator, will not remain for a third term, and a new General Administrator will be appointed soon.
- Regarding ANDRA, Pierre-Marie Abadie, former Director of Energy at the Ministry of Ecology, Sustainable Development and Energy, has been nominated Chief Executive Officer in replacement of Marie-Claude Dupuis.

CIGEO: High-level waste deep repository

As the result of a public consultation process, ANDRA (National Radioactive Waste Disposal Organisation) has engaged several main improvements for the CIGEO project:

- Provide a pilot plant to test, under real conditions at the CIGEO site, all of the disposal functions: technical measures to control operating risks, capacity to remove packages being disposed of, disposal monitoring sensors, and techniques for sealing cavities and galleries, among other things.
- Detail the conditions for implementing reversibility.

In addition, it was decided to continuously update the roadmap for the entirety of CIGEO development and operation with input from stakeholders and government approval.

In this framework, it is planned next year that ANDRA will submit to the government the project roadmap for operation and disposal, as well as a first set of options for security and retrievability. Afterwards, in 2017 the application to the regulatory authority to construct CIGEO is planned for an expected construction in 2020. The pilot phase of disposal will start in 2025. This facility will be in operation in 2030.

Creation of an R&D institute between CEA, AREVA and EDF on LWR and associated fields

To renew and boost their R&D long lasting partnership, CEA, Areva and EDF have founded a dedicated R&D three-party institute. In this new framework, every R&D project is managed by a three-party team which reports to the Executive Committee. The Institute covers PWR technology and simulation matters including fuels, transportation and interim storage. The Institute comprises a Scientific Council chaired by the atomic energy High Commissioner.

International co-operation in the field of fast neutron reactor ASTRID

The French Prime Minister confirmed during the World Nuclear Exhibition (WNE) in October 2014 in Le Bourget that fast neutron reactors (FNRs) represent the future of nuclear energy.

As regards to the ASTRID project, which is linked to the CIGEO project (in the framework of the 2006 Waste management law), an agreement between France and Japan has been signed during the State visit to France of Prime Minister Abe. This agreement refers to the project engineering phase scheduled to be completed by the end of 2019, prior to the decision to be taken to construct the reactor. It consolidates the already existing international partnership of the project, after the signatures of two other agreements, with Great Britain earlier this year, and another one with the United States last year.

EPR project at Hinkley Point

The European Commission concluded on 8 October that the modified UK financial measures for the Hinkley Point C project are compatible with EU rules on state aid. The investment decision by EDF is the next step to officially launch the project.

Japan

Current status of nuclear policy

Concrete measures to realise the nuclear energy policy which is indicated in the new Strategic Energy Plan approved by the Cabinet on 11 April 2014 have been discussed in the Nuclear Energy Subcommittee of the Advisory Committee for Natural Resources and Energy since June 2014.

Intermediate report (tentative) will be summarised in December 2014, which will underline the necessity of continued development of various technologies such as fast reactors to utilise plutonium properly and the necessity for the government to discuss a suitable way of continuing fast reactor development etc.

Situation at the Fukushima Daiichi nuclear power plant

Currently Units 1 to 3 are in a stable state of cold shutdown.

Planned work for removing fuel from the spent fuel pool of Unit 4 will be completed around the end of this year.

Toward the start of removal of the fuel and fuel debris of Units 1 to 3, preparatory work to remove the cover of Unit 1 is being carried out. Dismantling of the cover is planned for completion around the end of this fiscal year (March 2015) and subsequent rubble-removal work will begin in JFY 2016.

The International Collaborative Research Centre on Decommissioning will be established next April. This centre has four functions: international collaborative research, research support with promotion system, human resources development and information transmission. In the next year initial operation will start using existing JAEA infrastructures and new infrastructures will be constructed in Fukushima, opening in 2016.

Safety inspection of nuclear power plants and nuclear fuel cycle facilities

The Nuclear Regulation Authority (NRA) is currently performing safety reviews of 18 units at 12 nuclear power stations based on the new safety standards.

- The NRA approved on 16 July this year the draft document of safety review of two nuclear power plants at Sendai Nuclear Power Station of Kyushu Electric Co. Ltd, which expresses that the safety measures taken by Kyushu Electric Co. Ltd are deemed to conform to the new regulatory requirements, and formally approved the safety review on 10 September 2014. Currently, the plan for construction works and operational safety programmes are under deliberation in the NRA.
- As for the deliberation on the seismic safety of nuclear facilities, the NRA judged that Units 1 and 2 of Japan Atomic Power Company's (JAPC) Tsuruga NPP and Unit 1 of Tohoku Electric Power Co., Inc.'s Higashidori NPP are on active faults. Although the utilities were not convinced of the judgment and re-examination was carried out, the judgment is not expected to be altered. It is said, however, that JAPC still plans to apply for the safety review based on the new regulatory requirements aiming at restarting operation of its units, which will bring the discussion in the official deliberation process.

Nuclear fuel cycle facilities

- In the course of the reviews on compliance with the new regulatory requirements, Japan Nuclear Fuel Limited (JNFL) submitted to the NRA a revised version of "Reprocessing Business Licensing Application" and the change of utilisation schedule of reprocessing facilities. By changing the assumptions about the review timeframe and the period of time for countermeasure construction, JNFL revised the completion timing of the Reprocessing Plant from October 2014 to March 2016.
- JNFL submitted a revised version of "MOX Fuel Fabrication Business Licensing Application to the NRA. It revised the completion timing of the MOX Fuel Fabrication Plant from March 2016 to October 2017.

Lower house election

The Lower House of the Diet was dissolved on 21 November 2014 and a general election was held on 14 December, with the campaign period starting on 2 December.

- The Liberal Democratic Party (LDP), the governing party of Prime Minister Abe, won a landslide victory winning 290 out of the 475 seats. This result is expected to spur the restart of nuclear energy programmes, since the LDP has recognised the role of nuclear energy and advocated the continuation of its use to a certain extent, provided safety standards are met.

Update on the Japan Atomic Energy Agency

With regard to Monju, based on the requests from the NRA, JAEA prepared a report on "Safety Approach for Monju" in terms of (1) approach on enhancement of measures against design basis accident (DBA) and (2) approach on prevention and mitigation against severe accident (SA), and submitted it to the NRA in July 2014. Since the new regulatory requirements are said to be revised based on public comments etc., JAEA took the initiative to summarise the above report with the expectation that NRA will approve such a safety approach.

As for Joyo, JAEA successfully retrieved the damaged upper core structure (UCS) and the material testing rig with temperature control (MARICO-2), and installed a new UCS in the rotating plug on 21 November 2014, which resulted in restoring the periphery around the reactor vessel to normal. Re-installation of the devices which were retrieved from the rotating plug will be conducted over the coming half year.

Application of the decommissioning plan of Tokai reprocessing plant during the next midterm target period (starting from JFY 2015) is under consideration.

JAEA submitted "Permission for change in reactor installation license" and "Approval of operational safety programs" of the high-temperature engineering test reactor (HTTR) to the NRA on 26 November 2014.

Korea

Overview

Korea has harnessed nuclear energy as an engine for national development for the past 40 years. Recognising that nuclear safety is a top priority, Korea will continue to use nuclear energy as a practical solution to pressing issues such as rising energy demand and climate change. Under the 2nd national energy master plan, the share of installed nuclear power capacity in the total energy mix is planned to reach 29% by 2035. According to this plan, 11 nuclear power units will be built by 2024. In particular, Shin Kori units 3 and 4 are expected to begin commercial operation in 2015 and 2016, respectively.

To ensure the sustainable development of nuclear energy, Korea is making every possible effort to resolve global issues and is conducting R&D on technology innovations such as strengthening nuclear safety, enhancing non-proliferation and nuclear security, developing research reactors, and promoting the use of nuclear energy.

Improving nuclear safety

Korea has established and operated a Top Regulators Meeting on Nuclear Safety (TRM) with China and Japan to enhance nuclear regulations and the exchange of information among the three counties in Northeast Asia. In particular, the three countries agreed last year to establish a new TRM+ mechanism under the TRM framework with the participation of other countries and relevant international organisations. The 2nd TRM+ was successfully held in Korea in November 2014.

Non-proliferation and enhanced nuclear security

President Park delivered a keynote speech at the 2014 Hague Nuclear Security Summit. In her speech, the President outlined a four-point proposal to bolster the international nuclear security regime, and Korea is currently implementing related follow-up measures. In addition, the International Nuclear Non-proliferation and Security Academy (INSA) opened in February this year, and has provided high quality education and training programmes on safeguards and nuclear security to IAEA Member States.

Achievements of nuclear R&D

Having obtained the world's first Standard Design Approval (SDA) for SMART (System-integrated Modular Advanced Reactor) in 2012, Korea is carrying forward innovative technology development in its endeavour to improve the reactor's safety and economics. At the same time, to improve the export competitiveness of research reactors and ensure a stable supply of medical radioisotopes, Korea is moving ahead with the development and demonstration of key technologies through a new research reactor project, which is slated for completion by 2018.

Korea and the Netherlands officially signed a contract for the Netherlands research reactor improvement project (OYSTER) at the summit meeting held in Korea last November. This project aims to build a cold neutron research facility at the research reactor HOR operated by Delft University of Technology. The commissioning test is to be completed by 2018.

Efforts to promote peaceful uses of nuclear energy

The Korean government established a Nuclear Energy-based Creative Economy Action Plan (2013-2017) to expand the research infrastructures of radiation convergence technologies, disease diagnosis and treatment technologies, and radiation and radioisotopes applications. Furthermore, Korea has embarked on the development of a radiation medicine database this year, and plans to collaborate with the IAEA's radiotherapy and nuclear medicine database.

Korea has participated in the IFNEC initiative since its launch, and believes its activities are of great aid to promoting the peaceful uses of nuclear energy and ensuring non-proliferation. In an endeavour to provide an example to newcomer countries participating in IFNEC, Korea is sharing its valuable experience in human resource development, nuclear infrastructure, and small and medium sized reactors. Both the Executive Committee and Steering Group Meeting of IFNEC were held in Korea in October 2014, and provided a good opportunity for member states to share their opinions on pending issues.

R&D efforts on Gen IV nuclear system

Currently, an advanced nuclear energy system that couples pyro-processing and Generation IV SFRs is being considered as an efficient way to manage and utilise spent fuel. Korea has also been promoting VHTR R&D projects and is actively participating in the Generation IV International Forum.

Sodium-cooled fast reactor (SFR)

Korea has been developing a prototype Generation IV SFR (PGSFR) design according to an official long-term plan, and is going to submit a specific safety analysis report to the regulatory body by 2017, and will obtain its design approval by 2020. As a preparatory step, KAERI is going to submit a preliminary safety information document (PSID) to the regulatory body by 2015 to have an independent and authorised peer review on the safety of the prototype SFR. Korea has been actively participating in the GIF SFR activities, in particular, the GIF SFR System Integration and Assessment (SIA) Project which started in October 2014. The large sodium loop for component tests, STELLA-1, started operation in 2014. Through this test loop, the performance tests of the major components of a decay heat removal system, such as a decay heat exchanger (DHX), an air heat exchanger (AHE), and a mechanical sodium pump were completed. The test results will be used for the validation and verification (V&V) of the computer codes, and will also be used to design an integral test loop that simulates the T/H characteristics of the prototype SFR primary and intermediate heat transport systems.

Very-high-temperature gas-cooled reactor (VHTR)

The VHTR is primarily dedicated to heat generation for hydrogen production in Korea. The long-term VHTR development plan consists in two major projects: a nuclear hydrogen key technology development project, and a nuclear hydrogen development and demonstration project. The key technology development project focuses on the development and validation of key and challenging technologies required to realise a nuclear hydrogen system. The key technology development project, which consists in the development of computational tools, high temperature experimental technology, TRISO fuel fabrication, and a hydrogen production process, is the basis of the Gen IV VHTR collaboration R&D.

In 2014, a lab-scale pressurised sulfur-iodine hydrogen production process was demonstrated and a helium test loop was operated at an elevated temperature to generate data for validation of the computational tools. Irradiation testing of the TRISO fuel manufactured by KAERI has been completed at the HANARO research reactor. In addition, an economic evaluation was conducted through a VHTR conceptual design study, and a tentative plan for VHTR demonstration projects was set up appropriately.

Russian Federation

Nuclear power in Russia

Thirty-three nuclear power units are in operation in Russia, with 25.242 GWe total electric power capacity. At the moment, in the Russian Federation there are nine power units under construction. The third Rostov NPP unit has been fully loaded with the fuel assemblies and is in the stage of "first criticality".

In 2013, total electricity production by NPPs in Russia was 172 billion kW/h, that is approximately 16% of Russia’s total electricity production. In 2014 the Russian NPPs’ load factor was more than 80%.

After the Fukushima Daiichi accident, the analysis and enhancement of all the operating, constructed and designed Russian NPPs safety is underway with regard to a similar type of event.

Activities on innovative reactor technologies

Russia is involved in the development of five of the six advanced reactor technologies considered in the GIF. Conceptual research is being provided in three areas: supercritical water reactor, molten salt reactor, and fast gas reactor. The main activities are focused on sodium-cooled fast reactors and fast reactor with lead metal coolant. The work on the items listed is carried out within the framework of the Federal Target Program (FTP) “Nuclear power technologies of a new generation for period of 2010-2015 and with outlook to 2020”.

The work is comprehensive, it is not just limited to the development of particular projects related to reactor facilities, but also covers the issues related to the closure of the nuclear fuel cycle.

Sodium-cooled fast reactor

There are three units with fast neutron reactors in Russia:

- Industrial power unit BN-600 (more than 34 years of operation).

- Research reactor BOR-60 (more than 44 years of operation.
- Industrial power unit BN-800 (commissioning operations).

According to the lifetime extension approved by the regulator, BN-600 will operate until the end of March 2020; the lifetime of BOR-60 has also been extended to the end of 2020.

Regarding the BN-800 unit which is under construction at the Beloyarsk NPP site, the first criticality was achieved in 27 June 2014. Currently the reactor is under commissioning and is prepared for start-up in 2015.

In the frame of the FTP, the development of the project of high-power fast neutron reactor BN-1200 which meets the requirements of the 4th generation reactor power systems is underway as well as multifunctional fast sodium-cooled research reactor MBIR which will replace the BOR-60 reactor. An international research centre based on the MBIR reactor is planned to be established.

The FTP foresees completion of the BN-1200 project in 2015 and to start operation of the MBIR reactor in 2019 at the RIAR site (Dimitrovgrad). The construction of an advanced Beloyarsk NPP unit with BN-1200 is under consideration.

Fast reactors with heavy liquid metal coolant (HLMC)

In the frame of FTP the project of the lead-cooled BREST-OD-300 is under development, to be completed by the end of 2015, as well as the construction of demonstration plants aimed at the verification of feasibility of the above mentioned reactor technologies with HLMC.

The decision was made to construct a demonstration facility with BREST-OD-300 reactor and with the on-site nuclear fuel cycle at the site of the Siberian Chemical Combine in Tomsk. The preparatory work was launched.

Besides, the FTP roadmap envisages:

- A large-scale refurbishment of the nuclear industry experimental base, including construction of the MBIR reactor and upgrading the fast critical facilities (BFS).
- Creation of the industrial basis for the fuel production for advanced reactor facilities and its reprocessing to close the nuclear fuel cycle, in particular, production of MOX-fuel for the new generation fast reactors, including BN-800 reactor; work is underway to develop a pilot industrial complex for the dense nitride fuel production and demo semi-industrial pyrochemical complex.
- Development of computation codes for justification of the parameters and safety level of the advanced new generation nuclear energy systems.

Activities within GIF

On 22 October 2014 the State Corporation “Rosatom” signed the GIF Project Arrangement on system integration and assessment of sodium-cooled fast reactor.

Preparatory work is underway to sign the GIF SFR Project Arrangements on advanced fuel and equipment projects and the energy conversion unit. Proposals to join the GIF SCWR Project Arrangement on thermal-hydraulics and safety are also under preparation.

South Africa

In February 2014, South Africa completed an International Atomic Energy Agency Emergency Preparedness Review (EPREV) mission in order to assess and further improve its readiness for radiological emergencies.

A peer review conducted in 2014 by the World Association of Nuclear Operators at the Koeberg nuclear power plant showed significant improvement since the previous review in 2011. The plant also broke all its performance records during the year and the procurement process for the replacement of the six steam generators was concluded.

Newly re-elected President Jacob Zuma emphasised the importance of the 9 600 MW nuclear power expansion programme in his State of the Nation Address of June 2014. It was recognised that further technical work would need to be done on funding, safety, exploitation and the local manufacture of components for the nuclear programme.

Government has entered into several negotiations with vendor countries and has also signed Inter-Governmental Framework Agreements on Nuclear Energy (bi-lateral) with the Russian Federation, France, and China during the second half of the year.

During the last quarter of 2014, the South African government successfully conducted the nuclear vendor parade workshops with delegations from the Russian Federation, China, France, Korea and the United States. The conclusion of the vendor parade marks a significant milestone in the government pre-procurement phase for the roll out of the nuclear new build programme. Going forward government will design and launch a procurement process.

Switzerland

The Swiss government has taken the decision to phase out nuclear energy, and the currently operating four nuclear power plants will not be replaced. The duration of the remaining operation time of the four reactors should be determined by safety considerations according to the Swiss licensing regime. This was confirmed by the decisions taken by the National Council, one of the two chambers of the Federal Assembly end of 2014 (The Council of States will decide in Spring 2015).

One utility (BKW) has announced that it will shut down the Mühleberg BWR-4 by 2019, after 47 years of successful nuclear operation. This decision was influenced by many factors. Economic consideration in relation to the required post-Fukushima related retrofitting played an important role. In general, the generation of base load electricity is facing considerable economic challenges in Switzerland. The transformation of the Swiss energy system becomes therefore even more demanding. Yet, important infrastructure projects are ongoing: The replacements of the reactor vessel heads of the two-unit Beznau PWR as well as the construction of a large hydro-pump storage plant are well underway.

The key mission of the PSI Nuclear Energy and Safety Department as unique nuclear competence centre in Switzerland is to maintain nuclear competence for the foreseeable future. The focus remains on the safety of LWRs. The scientific support for the storage of deep geological waste repositories represents another significant key element. Strong dedication to nuclear education (with three university professors and many senior scientists as lecturers) will help ensure an adequate flow of competent researchers into the nuclear field.

Involvement with Gen IV reactor concepts, particularly through research related to high-temperature materials and MSR and by bilateral co-operation on SFR work, offers attractive opportunities for innovative research, of importance especially to keep young researchers in the field. At the same time, it allows Switzerland to closely monitor the international progress of reactor technology towards more sustainable nuclear energy.

United States

New developments

Nuclear energy continues to be a vital part of the United States "all-of-the-above" energy strategy for a sustainable, clean energy future. As Secretary of Energy, Dr. Ernest Moniz stated at the IAEA General Conference in September 2014, "President Obama has made it clear that nuclear energy is an important part of our all-of-the-above energy strategy."

The United States has experienced a reduction in the amount of energy produced from nuclear power with several plants closings, but is very optimistic with the construction of 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 projected for 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. In fact Watts Bar Unit 2 is 90% complete and when on line, will be the first US reactor to be completed this century since Watts Bar Unit 1 began operating in 1996.

On 16 September 2014, the US Nuclear Regulatory Commission (NRC) approved the final rule certifying the GE-Hitachi Nuclear Energy's (GEH) economic simplified boiling water reactor (ESBWR), providing additional momentum to the deployment of a new generation of passively safe nuclear power plants. The rule went into effect on 14 November 2014. Certification of this design allows an applicant for a nuclear power plant licence to reference this design without submitting all the associated safety information; instead focusing only on the site-specific safety issues for the proposed plant. The ESBWR is a 1 594 MWe natural circulation reactor, which includes passive safety features that would cool the reactor after an accident without the need for human intervention.

The NRC is currently reviewing two Combined Licence applications referencing the ESBWR: Detroit Edison's Fermi Unit 3 in Michigan and Dominion's North Anna Unit 3 in Virginia. GEH was a participant in the DOE Nuclear Power 2010 (NP 2010) programme, which cost-shared the development of the Design Certification application (DCA) and first-of-a-kind engineering (FOAKE) for the ESBWR. The NP 2010 programme also cost-shared the development of the DCA and FOAKE for the Westinghouse AP1000, which is currently under construction in Georgia and South Carolina, as noted above.

The United States continues to strongly believe that leveraging our financial and technical resources through engagement in multilateral forums, as well as through bilateral co-operation, is important.

DOE's nuclear reactor technology programmes are managed with the Office of Nuclear Energy (NE). The advanced non-light water reactor programme, managed under NE's 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 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 generic advanced reactor technologies, which include high-temperature metals, instrumentation and controls, and energy conversion systems.

The United States "all-of-the-above" energy strategy includes DOE's Small Modular Reactor (SMR) Licensing Technical Support (LTS) programme. This is a six-year, USD 452 million programme managed by NE's Office of Light Water Reactor Technologies supports 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 the mPower America team, consisting of Babcock & Wilcox (B&W), Bechtel International and the Tennessee Valley Authority (TVA), for cost-shared investment to support the design development, certification and licensing activities of B&W's mPower reactor to be sited at TVA's Clinch River site in Tennessee. In December 2013, DOE announced the selection of NuScale Power LLC to receive a financial assistance award to support efforts to design, certify and commercialise NuScale's SMR design. The DOE stands firmly behind small modular reactors as a way meet the nation's growing energy demands – including replacing retiring power plants – while providing reliable, affordable low-carbon power. Overall, the SMR LTS Programme supports the licensing of innovative designs that improve SMR safety, operations and economics. We expect these SMRs to have lower core damage frequencies, longer post-accident coping periods, enhanced resistance to natural phenomena, and potentially smaller emergency preparedness zones than currently licensed reactors.

An important effort pertains to the development of advanced non-light water reactor licensing strategies. NE has been working with the NRC over the past several years on specific technical issues related to the licensing approach for high-temperature gas-cooled reactors (HTGRs). The NRC recently issued staff positions on key issues related to HTGRs which show progress toward reduction of regulatory uncertainty for possible licensing of future HTGRs.

Expansion of this effort to address advanced reactor concepts more broadly has been initiated to develop Advanced Reactor Design Criteria that may be used as a first step in developing a framework for a licensing approach for advanced non-light water reactors. NE is in the process of drafting Advanced Reactor Design Criteria (applicable to most advanced concepts) and design criteria sets tailored specifically to sodium fast reactors and modular high-temperature gas reactors. These three design criteria sets were provided to the NRC in December 2014.

In support of the nuclear energy industry's long-term viability, NE is also working to train the next generation of nuclear engineers and scientists. NE funds a number of research activities at US universities. Through our Nuclear Energy University Programs (NEUP), NE is currently funding multiple Integrated Research Projects (IRPs). Two of our IRPs address different aspects of the technology required for the development and deployment of fluoride salt-cooled high-temperature reactors (FHRs), including pre-conceptual designs, approaches for tritium management, coolant chemistry and corrosion control, structural materials qualification, advanced heat exchanger designs, development and validation of thermal hydraulics and neutronics codes, and investigation of licensing issues. NE is also sponsoring an IRP to develop benchmarked methods for predicting the behaviour of reactor materials at the very high neutron irradiation doses needed for fast reactor applications by using microstructures and properties developed under accelerated irradiation from multiple ion beams.

NE is also continuing efforts, begun in 2012, to seek interaction with industry for the development of its R&D programme. In April 2014, NE issued a request for information (RFI) to solicit information on advanced concepts from industry. Based on the information received from industry that addressed the R&D needs of emerging concepts, NE issued a Funding Opportunity Announcement and in late October awarded several grants to industry to perform cost shared R&D on advanced reactor technologies.

Finally, NE is in the process of obtaining formal approval of its recently updated R&D roadmap. The 2014 version of the NE Roadmap proposes to continue addressing extending the life of current reactors, developing technologies to support the deployment of advanced reactors and developing sustainable nuclear fuel cycles. The 2014 version adds new emphasis on developing and maintaining an integrated national research, development and demonstration framework and maintaining US leadership at the international level by engaging nations that pursue peaceful uses of nuclear energy.

Turning to the status of the US response to the lessons learnt from the Fukushima Daiichi accident, the NRC is currently reviewing US plant operators' integrated plans for complying with the commission's March 2012 orders mandating enhanced mitigation strategies for responding to extreme weather events resulting in the loss of offsite power and enhanced spent fuel pool instrumentation. Implementation of these plans will be carried out over the next two years. boiling water reactor (BWR) operators are expected to submit updates to their integrated plans to address the requirements of the March 2012 order on BWR containment vents, as modified by the commission in June 2013. The NRC's rulemaking activities related to Fukushima lessons learnt also continue to advance:

- The Station Blackout Mitigation Strategies Rule, which will make Mitigation Strategies Order a regulation, is expected to be finalised in December 2016.
- The Onsite Emergency Response Capabilities Rule, which will integrate plant emergency procedures, is expected to be finalised the same month.
- Meanwhile, the Filtering and Confinement Strategies Rule, which will consider additional protections to limit potential release of radioactive material, is expected to be finalised in 2017.

Meanwhile, US nuclear operating utilities have continued to implement their "FLEX" approach for responding to extreme external events leading to station blackout and involving multiple units. As part of a USD 400 million investment from the nuclear energy industry, two regional response centres for storing emergency equipment to support long-term cooling and stabilisation were established. In May 2014, the first regional response centres opened in Phoenix, Arizona, followed by the second in June 2014 in Memphis, Tennessee. Plant operators have also completed seismic and some flooding re-evaluations for each of the nuclear stations and are taking appropriate interim actions, if necessary, pending more in depth analyses of plant risks associated with the revised natural hazards.

The DOE Waste Isolation Pilot Plant (WIPP) is the nation's first repository for permanent disposal of defence-generated transuranic (TRU) radioactive waste resulting from research and production of nuclear weapons. Located in southeastern New Mexico, 26 miles east of Carlsbad, WIPP's facilities include disposal rooms excavated in an ancient, stable salt formation 2 150 feet (655 metres) underground. Waste disposal began at WIPP on 26 March 1999.

WIPP has been shut down since 5 February 2014, when a vehicle used to transport salt caught fire in the underground facility. Underground workers were safely evacuated. The Department deployed an independent Accident Investigation Board (AIB) to determine the cause of the fire. The AIB released their findings on 13 March, which primarily identified poor maintenance as the root cause of the fire. A Corrective Action Plan is being finalised.

On 14 February, a continuous air monitor detected a radiological release in the underground facility. The ventilation system shifted to filtration mode, redirecting air exiting the repository through high-efficiency particulate air filters and minimising radiation releases to the environment. Slightly elevated levels of airborne radioactive concentrations were detected outside the WIPP facility after the release occurred. However, the radiation levels were well below a level hazardous to the public or environment.

A second independent AIB was deployed to determine the cause of the radiation release and the Phase I report was released on 24 April. The Phase I investigation focused on the reaction to the radioactive material release, including related exposure to above-ground workers and response actions. The Phase 1 report stated the direct cause of the release was the breach of at least one TRU waste container in the underground facility, which resulted in airborne radioactivity escaping into the environment downstream of the high-efficiency particulate air filters. A Corrective Action Plan is in development. Phase 2 of the radiological release investigation is ongoing. After the source of the radiological event is fully evaluated, the AIB will release a supplemental report focused on the direct cause, as well as contributing causes, of the release in the underground facility.

The department established a Technical Assessment Team to determine the mechanisms and chemical reactions that resulted in the drum failure and release of the radioactive material through analyses and assessments. The TAT draws upon the technical and scientific capabilities of the department's national laboratories (Savannah River National Laboratory, Pacific Northwest National Laboratory, Los Alamos National Laboratory, Sandia National Laboratory, Oak Ridge National Laboratory, and Lawrence Livermore National Laboratory).

The Department of Energy is committed to reopening the WIPP repository as soon as possible, while assuring the safety of the public, the workers, and the environment. However, the length of time required to recover from these two incidents cannot be fully known until the cause of the events are identified and understood, and until correction actions to mitigate reoccurrence of the incidents are completed.

On 26 August 2014, the NRC approved a final rule on the effects of continued storage of spent nuclear fuel (the so-called "waste confidence" issue), thus paving the way for the lifting of the moratorium on final actions for nuclear power plant licences and renewals. Historically, "waste confidence" has been NRC's generic determination regarding the environmental impacts of storing spent nuclear fuel beyond its licensed life for operation of a nuclear power plant. That generic analysis had been incorporated into the commission's National Environmental Policy Act (NEPA) reviews for new reactor licences, licence renewals, and Independent Spent Fuel Storage Installation licences through the 2010 Waste Confidence Rule. In response to near-unanimous public comment to more accurately reflect the nature and content of the rule, the new final rule and Generic Environmental Impact Statement (GEIS) has been renamed from “waste confidence” to “continued storage of spent nuclear fuel.”

The commission’s action marks the end of a two-year effort to satisfy a remand from the US Court of Appeals for the District of Columbia Circuit. In June 2012, the court found that some aspects of the 2010 rulemaking did not satisfy the NRC’s NEPA obligations and struck down the rulemaking. It directed the agency to consider the possibility that a geologic repository for permanent disposal of spent fuel might never be built and do further analysis of spent fuel pool leaks and fires. In response to the court’s decision, the commission decided in August 2012 to stop all licensing activities that rely on the Waste Confidence rule. The commission instructed the staff to develop a new rule and issue it and the supporting GEIS no later than fall 2014.

The new continued storage rule adopts the findings of the GEIS regarding the environmental impacts of storing spent fuel at any reactor site after the reactor’s licensed period of operations. As a result, those generic impacts do not need to be re-analysed in the environmental reviews for individual licences. The GEIS analyses the environmental impact of storing spent fuel beyond the licensed operating life of reactors over three timeframes: for 60 years (short-term), 100 years after the short-term scenario (long-term) and indefinitely.

The GEIS analyses impacts across land use, air and water quality, and historic and cultural resources throughout each timeframe. It also contains the NRC’s analysis of spent fuel pool leaks and fires in response to the Appeals Court remand. The rule does not authorise, license, or otherwise permit nuclear power plant licensees to store spent fuel for any length of time, as that is covered by the applicable licence.

In a separate Order issued at the same time, the commission approved lifting the suspensions and provided direction on the resolution of related contentions in 21 adjudications before the commission and the Atomic Safety Page and Licensing Boards. The Order authorises the NRC staff to issue final licensing decisions as appropriate once the final rule becomes effective.

Brazil

Nuclear power in Brazil

Brazil is constitutionally committed to the peaceful use of nuclear energy. Nuclear power has a share of about 3% in the country’s electricity installed capacity. The country has currently two pressurised water reactor (PWR) nuclear power plants (NPPs) in operation, totalling about 2 000 MWe of nuclear installed capacity. A third 1 350 MWe PWR NPP is under construction and is expected to enter into operation in 2018. Nuclear power generated around 4% of the electricity consumed in the country in 2014.

The 2030 National Energy Plan includes the construction of four additional NPPs up to 2030, two plants in the northeast and another two in the southeast regions of the country. The Fukushima Daiichi nuclear accident in March 2011, among other factors, put this expansion plan on hold.

R&D activities on innovative technologies

Brazil joined the Generation IV International Forum from the beginning in 2001 and contributed to the development of the Generation IV Technology Roadmap 2002. Since the completion of the technology roadmap process, however, Brazil has not joined any R&D project for advancing the six Generation IV nuclear energy systems selected. Despite being a non-active member, Brazil signed the revision of the GIF Charter in 2011 in the expectation that the country might collaborate in the future to advance some of the Generation IV systems.

In parallel to its participation in the GIF, in 2002 Brazil also joined the International Project on Innovative Nuclear Reactors and Fuel Cycle (INPRO) co-ordinated by the IAEA. In this initiative, Brazil performed an assessment study of two small medium-sized reactors for deployment in the country using INPRO assessment methodology, and participated in INPRO Collaborative Projects. One of them is devoted to investigate the technological challenges related to the use of liquid metal and molten salt coolants for heat removal from reactor cores operating at high temperatures, and the other is related to the environmental impacts caused by the deployment of innovative nuclear energy systems. Over the years, Brazil participated in INPRO dialogue forums on topics such as nuclear energy innovations, global nuclear energy sustainability, licensing and safety issues for small and medium size reactors (SMRs), and the sustainability of nuclear energy systems based on evolutionary reactors, the latter held in 2013.

Starting in 2008, the Ministry of Science, Technology and Innovation organised Brazilian scientific research into large networks named National Institutes of Science and Technology, aiming to promote national and international scientific co-operation on several thematic areas of knowledge. One of the approved networks is the National Institute of Science and Technology for Innovative Nuclear Reactors, which recently proposed a five-year research programme focusing on technologies related to hybrid subcritical systems, very-high-temperature nuclear reactors and advanced light water reactors.

United Kingdom

Energy policy and status of nuclear energy in the UK

The UK has benefitted from nearly 60 years of successful and, above all, safe low carbon power generation from nuclear energy. Nuclear power plant currently contributes about 20% of the UK's electricity production and the UK government sees nuclear energy as continuing to be a key part of the country's low-carbon energy mix.

A programme of new build is currently underway, with the initial aim of delivering up to 16 GW of new PWR generating capacity by the late 2020s. This would supplement and eventually replace current nuclear generation capacity as the present fleet of reactors is retired.

As the deployment of nuclear electricity generation in the UK is market led, the eventual amount of nuclear power plant on its grid will depend on industry's ambition, the success of the initial new build programme, subsequent reduction in cost through experience, growth in investor confidence, and realising economies of scale. It is thought that this could rise to be as much as 40%-50% of the total UK generating capacity by the middle of this century. In scenarios with the highest levels of electricity production, this could be equivalent to 75 GW of nuclear generation.

Nuclear industry strategy, research and innovation co-ordination

In March 2013 the UK government published a Nuclear Industrial Strategy[4], setting out a consistent long-term approach to the deployment of resources to grow commercial opportunities, stimulate economic growth and create jobs. The strategy reiterated the government's view that nuclear power is essential to meeting the objective of delivering a secure, sustainable and low carbon energy future. Innovation and R&D are recognised as being central enablers to realising this goal. Government also committed to keep under review the level of public nuclear R&D expenditure, including that relating to future (including Gen IV) reactors and their associated fuel cycles.

During 2012-14 an initial government funded R&D programme on advanced reactors and fuel cycles was undertaken by a consortium led by the National Nuclear Laboratory, universities and industry. This has enabled the development of an informed input to the R&D needed to underpin the recommendations in Nuclear Industry Strategy.

In January 2014 the UK government established the Nuclear Innovation and Research Advisory Board (NIRAB). NIRAB is charged with advising ministers and government departments on the government funded innovation and R&D that will be required to underpin government energy and industrial policies. The initial approach taken by NIRAB has been to focus on

4. Nuclear Industrial Strategy – the UK's Nuclear Future, HM government, March 2013.

identifying gaps in both funding and research that need to be filled if the UK is to keep open its options to increase the contribution of nuclear power. The key priority that has been identified to date is funding for R&D associated with advanced reactors, including Generation IV systems, and their associated fuel cycles.

The key recommendation is that, in order to meet the objectives set out in the Nuclear Industrial Strategy, the UK government needs to fund a balanced portfolio of R&D programmes and infrastructure investment across the areas of fuel manufacture, reactor technology, fuel recycle, waste management and cross cutting themes, such as modelling and simulation.

NIRAB and the Nuclear Innovation and Research Office (NIRO), which acts as its secretariat, have been working with government departments to make the case to secure the funding required to initiate such a programme.

It is recognised that international collaboration will be an essential component of any UK research programme.

Investment in research infrastructure

In line with its ambitions to continue to deploy nuclear energy and develop its technical expertise, the UK has undertaken a suite of new initiatives to expand nuclear energy research infrastructure, which include the following actions.

Jules Horowitz reactor

In 2013, the UK joined the consortium developing the Jules Horowitz Reactor in France, with UK interests in the project being represented by its National Nuclear Laboratory.

National nuclear user facility

In 2013, the UK government also established a National Nuclear User Facility (NNUF). The aim of this is to provide the UK nuclear R&D community with better research facilities. The initial investment, which focuses on laboratory equipment, was in three complementary facilities at: the Central Laboratory of the National Nuclear Laboratory (NNL), the Culham Centre for Fusion Energy (CCFE), and the Dalton Cumbrian Facility (part of The University of Manchester). This investment has enabled the NNUF to provide equipment for experiments on materials that university laboratories cannot accommodate.

Nuclear fuel centre of excellence

In October 2014, the UK launched its Nuclear Fuel Centre of Excellence (NFCE). This is a unique facility, established with government support, in order to provide academic research capability in a technology that is key to securing the future energy security of the UK. The NFCE will provide the equipment and expertise to develop advanced nuclear fuels with enhanced safety and economic benefits for new reactor systems, as well as playing a leading role in the optimisation of current fuel designs. It will also help in developing and transferring skills from academia to the commercial nuclear sector.

Nuclear fuel R&D will be carried out across four NFCE locations, the University of Manchester's Dalton Cumbrian Facility, the National Nuclear Laboratory's (NNL) Central Laboratory in Cumbria, and the laboratories based at the University of Manchester and NNL's Preston Laboratory. The facilities will be available for access by all UK academic and industry researchers.

Chapter 3. System reports

This chapter gives a detailed overview of the achievements made in 2014 in the R&D activities carried out under the four system arrangements (VHTR, SFR, SCWR, GFR) and under the two MOUs (LFR and MSR).

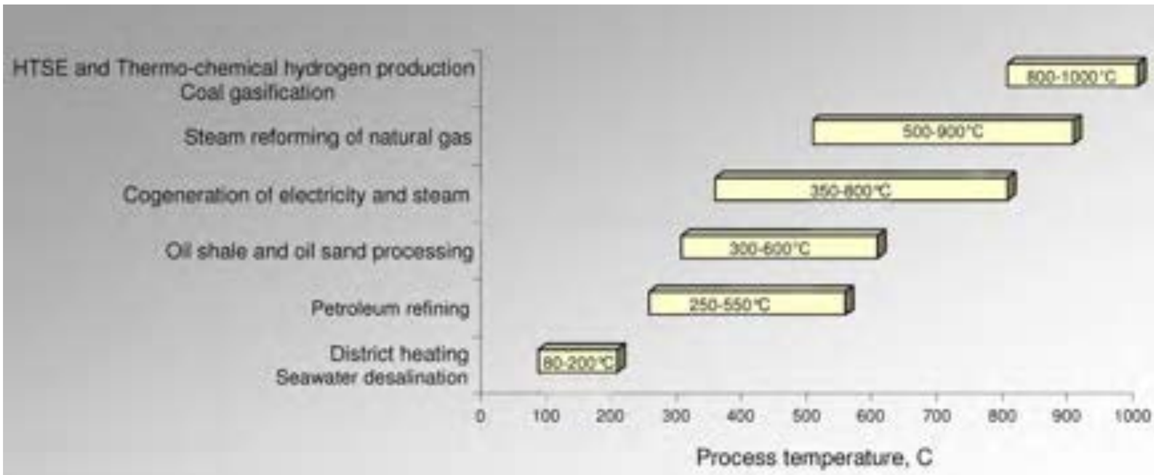
3.1 Very-high-temperature reactor (VHTR)

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, self-acting decay heat removal capability, and resulting in inherent safety and process heat application capability.

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

Figure 3.1: **Industrial applications vs. temperatures**



Courtesy: Phil Hildebrandt, Battelle Energy Alliance, Global Petroleum Conference, 11 June 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 under normal operation and accident conditions.

Former HTR reactors, such as AVR and HTTR, were already operated at temperature up to 950°C. VHTR can now 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 available. However, direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles are perceived as longer-term options.

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

The technology is being advanced through near and medium-term projects, such as HTR-PM, NGNP, GT-MHR, NHDD, and GTHTR300C, led by several plant vendors and national laboratories respectively in China, the United States, Korea and Japan. The construction of HTR-PM demonstration plant (two pebble bed reactor modules with one super heat steam turbine generating 200 MWe) 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 be connected to the grid in 2017, which will represent a major step toward a Generation IV demonstration plant.

Status of co-operation

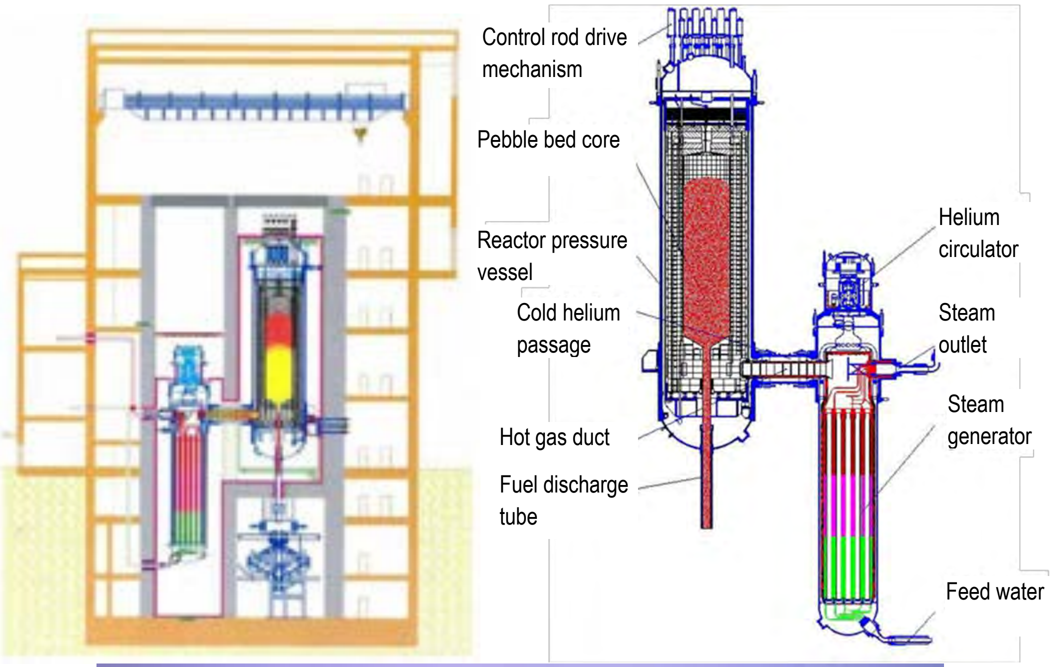
The VHTR system arrangement was signed in November 2006 by Canada, Euratom, France, Japan, Korea, Switzerland and the United States. In October 2008, China formally signed the VHTR SA during the policy group meeting held in Beijing. South Africa, which has expressed high interest in the VHTR, formally 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, Korea and the United States. The project arrangement has been extended to include input from China and was amended in 2013. It went into effect in January 2014.

The materials PA, which addresses graphite, metals, ceramics and composites, was signed by implementing agents from Canada, France, Japan, 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 from the material PA at the end of 2012. The details to amend the PA to reflect China's (INET) joining, and to extend the duration of the incorporated

Program Plan until 2015 have been finalised and approved by the VHTR SSC in 2014. The amended PA is expected to be signed by the signatories in early 2015.

Figure 3.2: **HTR-PM reactor building/primary circuit/construction (photo)**



The hydrogen production PA became effective on 19 March 2008 with implementing agents from Canada, France, Japan, Korea, the United States and Euratom. In 2010, China expressed its wish to join this PMB. As a result, an amended Project Plan incorporating Chinese contributions and other countries' updated contributions was prepared under the consensus of 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 2015.

The computational methods validation and benchmarks (CMVB) PA remains provisional. The 11th Provisional Project Management Board (PMB) Meeting was held on 27 October 2014 in China, with participants from China, Korea, Switzerland, the United States and Euratom. A new chair and co-chair were elected. The previous project plan was discussed to determine which tasks should be continued, modified, or eliminated. Leads were assigned in each major task area to gather input from members and draft new scope. The new Project Plan will be reconstructed and finalised in 2015.

Two other projects on components and high-performance turbo-machinery and on system integration and assessment (SIA) are still being discussed by the VHTR SSC but the associated research plans and project arrangements have not yet been developed.

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, higher fuel performance, coupling with process heat applications, co-generation, with potential conflicts between those challenging R&D goals.

The VHTR system research plan describes the R&D 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 is provisional, 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. R&D aims to increase the understanding of standard design (UO_2 kernels with SiC/PyC coating) and examine the use of uranium-oxicarbide UCO kernels and ZrC coatings for enhanced 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. In addition to other high-temperature heat exchangers, additional attention is being paid to the metal performance in steam generators, which reflects the current interest in high-temperature steam-based process applications. 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 or coupling processes (such as 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 GFR, so that common R&D could be envisioned for specific requirements, when identified.

System integration and assessment (SIA) 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 to 950°C) are being pursued to meet the needs of current 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 to 950°C) will enter the demonstration phase around 2017, based on HTR-PM experience in China which is scheduled to operate in 2017. Higher temperature version 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 and fuel cycle (FFC) project

The Very-High-Temperature Reactor (VHTR) Fuel and Fuel Cycle (FFC) project is intended to provide demonstrated solutions for the VHTR fuel (design, fabrication, and qualification) and for its back-end management, including novel fuel cycle options.

Tristructural isotropic (TRISO) coated particles, which are the basic fuel concept for the VHTR, need to be qualified for relevant service conditions. Furthermore, its standard design – uranium dioxide (UO_2) kernel surrounded by successive layers of porous graphite, dense pyrocarbon (PyC), silicon carbide (SiC), then PyC – could evolve along with the improvement of its performance through the use of a uranium oxycarbide (UCO) kernel or a zirconium carbide (ZrC) coating for enhanced burnup capability, minimised fission product release, and increased resistance to core heat up accidents (above 1 600°C). Fuel characterisation work, post irradiation examinations (PIE), safety testing, fission product release evaluation, as well as the measurement of chemical and thermomechanical material properties in representative conditions will feed a fuel material data base. Further development of physical models enables assessment of in pile fuel behaviour under normal and off normal conditions.

Fuel cycle back-end encompasses spent fuel treatment and disposal, as well as used graphite management. An optimised approach for dealing with the graphite needs to be defined. Although a once through cycle is envisioned initially, the potential for deep burn of plutonium and minor actinides in a VHTR, as well as the use of thorium-based fuels, will be accounted for as an evolution towards a closed cycle. The task structure is shown in Figure 3.3.

Status of ongoing FFC activities

During 2014, significant work was accomplished in the areas of irradiation and PIE, characterisation, safety testing, and back-end fuel cycle issues.

The Amended project arrangement has been signed and is effective since 12 January 2014.

Figure 3.3: **Task structure**

WP1 Irradiations and PIE	
Task 1.1	Irradiation design and operation
Task 1.2	Hosted joint irradiations
Task 1.3	PIE protocol and procedures
Task 1.4	Irradiation and PIE results
WP2 Fuel Attributes and Material Properties	
Task 2.1	Measurements of critical material properties
Task 2.2	Fuel material property database
Task 2.3	Characterisation techniques
Task 2.4	Fuel performance modelling
WP3 Safety	
Task 3.1	Pulse irradiation testing
Task 3.2	Heating test capabilities
Task 3.3	Heating tests
Task 3.4	Source term experiments
WP4 Enhanced and Advanced Fuel	
Task 4.1	Process development
WP5 Waste Management	
Task 5.1	Head-end process
Task 5.2	Graphite management
Task 5.3	Disposal behaviour and waste package
WP6 Other Fuel Cycle Options	
Task 6.1	Transmutation
Task 6.2	Thorium cycle

Irradiation and PIE

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. Post-irradiation examination is underway.

In the United States, PIE of AGR 1 is complete. 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 Korea, an irradiation of TRISO fuel began in the high-flux advanced neutron application reactor (HANARO) in July 2013. It was completed in March 2014. PIE is expected in 2015. Target burnup is 40 000 MWd/MtU.

The project parties have presented information on these irradiation tests at the High-Temperature Reactor 2014 conference (October 2014 in China) along with detailed reports in 2014.

Fuel attributes and material properties

The VHTR FFC Project Management Board (PMB) held the third SiC workshop on Jeju Island in Korea in September 2014. The workshop allowed researchers working in the areas of TRISO fuel and SiC composites for nuclear applications to interact and discuss the status of their research. Overviews of the TRISO and fully ceramic microencapsulated fuel programmes in the US, SiC composite work for nuclear application in Korea and the approach to statistical failure analysis of a brittle material like SiC were given in the opening plenary. Follow-on sessions provided more detailed results on using advanced microscopy to elucidate fission product behaviour in the irradiated SiC of AGR-1 particles, the irradiation creep behaviour of SiC composites, moisture/oxidation testing of SiC TRISO, and ZrC fabrication. Recommendations for follow-on workshops were elicited from participants and include:

- keep the workshop focused on TRISO fuel;
- continue to invite speakers from the broader SiC materials community;
- include facility tours when possible;
- keep the workshop to 1.5 to 2 days depending on the number of papers.

With the retirement of Y-W Lee in Korea, Dr. Tyler Gerczak (United States) volunteered to be the new organiser for future workshops. The next meeting is schedule for Paris in Spring 2017 to avoid a conflict with future HTR meetings.

In the EU, the pyro carbon irradiation for creep and swelling/shrinkage of objects (PYCASSO) I and PYCASSO II are irradiations of surrogate particles from France, Japan and Korea. X-ray tomography and nano-indentation of PYCASSO I samples are complete.

China has performed extensive characterisation of an oxidised SiC layer on TRISO fuel. Between 800 and 1 600°C. Work this year has focused on microstructural characterisation and understanding of the oxidation mechanisms. The testing was also expanded to include water vapor in the air.

Plans for continuing the successful round robin on fuel characterisation conducted under the International Atomic Energy Agency (IAEA) auspices from 2006-2010 was discussed among PMB members. The need for a leach burn- leach round robin was approved. At the September 2014 PMB meeting, a leach burn leach round robin discussion was led by the United States (Hunn). The goal is to see if everyone can measure the same level of defects from a batch of TRISO particles spiked with defective particles. All agreed with the approach presented by the United States to have four batches of particles (150 000 each) with 0, 1, 2 and 4 defects. Participants will not know how many defects are in each sample so the test is "blind". China has the Natural Uranium coated TRISO particles. ORNL gave presentations on how the defective particles will be created. Thus, the defective particles will be sent to China with detailed instructions on how to create each sample of TRISO particles. The samples will be sent to the United States, China and Korea. Korea will be responsible for collecting and assembling all the data from participants and writing the final report. A meeting is planned in association with the 2017 FFC PMB to present and discuss results.

A meeting was held in conjunction with the 2014 PMB meeting to discuss details of an accident benchmark for TRISO fuel performance codes as a follow-up activity to similar work conducted under the IAEA Coordinated Research Project (CRP). The United States (B.Collin) gave a historical review of the prior benchmark that was performed under the IAEA CRP6 and made recommendations for this new benchmark. He distributed a report that provided all of the input data needed for the benchmark and a schedule was established for the work. The United States will compile all participants' results in one report. A status update/workshop is planned in association with the 2016 FFC PMB meeting and a final report is planned at the end of the current five-year Project Plan in 2017. Korea and Japan confirmed their plan to participate. China is now unsure because their code is not yet ready. Europe had expressed interest at the previous PMB meeting but they did not attend this meeting so confirmation is being sought via e-mail.

Safety testing

In the EU, safety testing of HFR EU1 pebbles is completed at 1 700°C and 1 800°C. In Korea and China, the conceptual design of accident heating furnaces is underway but has been delayed because of technical and resource issues in each country.

The AGR 3/4 irradiation was initiated in December 2012 and completed in April 2014. In this experiment, 12 separate capsules containing designed-to-fail fuel are being irradiated over a spectrum of burnup, temperature, and fast fluence 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 are being gathered. Correlations of fission gas release to temperature and half-life have been established and a paper was presented at HTR-2014. PIE is anticipated to begin in 2015.

Both Korea and Japan are performing out-of-pile oxidation experiments with several graphite materials and SiC TRISO coated (surrogate) fuel particles under the air ingress accident condition for high-temperature gas-cooled reactor. Korea has studied the oxidation rate on fuel matrix material over a range of temperatures. 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.

Enhanced and advanced fuel

In the area of advanced fuel, both 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. Korea has focused on different methods of carbon dispersal. China is interested in developing UCO ZrC TRISO and has been evaluating ZrC coating layers.

Waste management and other fuel cycle options

This area covers three issues:

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

In 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 PMB in 2014. No activity has been started yet within the FFC project regarding the assessment of the VHTR thorium fuel cycle.

Project management

The VHTR FFC developed a five-year project plan (2012–2017). Based on successful collaboration in the first five years, the focus of the next five years will be in the following areas:

- Irradiation and PIE: focusing on PIE of irradiations from the first five-year plan and new irradiations in Korea and the United States.
- Fuel and material properties: focusing on additional SiC characterisation, a new leach-burn-leach round robin, and a new code benchmarking on accident performance of TRISO fuel.
- Safety testing: focusing on heating tests, source term testing, and air and moisture ingress experiments.

Conclusions

With the completion of the first five-year plan of collaborative work, the FFC project of the VHTR is producing many positive results. The success has led to an ambitious second five-year plan (2012–2017).

Materials

Although the term of the original Materials Project Plan (PP) was completed in 2012, the Materials Project Arrangement (PA) continued through 2014 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, when it was signed by the final Signatory of the PA. Contributions for the extension of the PP through 2015 were developed by the remaining six Signatories (European Union, France, Japan, Korea, Switzerland, and United States), as well as China that will be joining the PA. The extended and augmented contributions were compiled into a revised PP and unanimously recommended by the PMB for approval by the VHTR System Steering Committee, which was received on 18 February 2014.

As part of the development of the revised PP, a thorough review was made of all the high-level deliverables (HLDs). Where appropriate, HLDs were consolidated, added, deleted, or clarified to enhance accountability. All HLDs scheduled for completion prior to the end of 2014 were completed. Additionally, by the end of 2014, 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 is 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 and have now begun for graphite.

In 2014, research activities continued focused on near- and medium-term projects needs (i.e. graphite and high-temperature metallic alloys) with limited activities on longer-term activities related to ceramics and composites.

Characterisation of selected baseline data and its inherent scatter of candidate grades of graphite was performed by mulitiple members. Thermal conductivity, pore distribution (volume fraction and geometry), and fracture behaviour were examined for numerous grades. Graphite irradiations continued to provide data on property changes, especially at low doses and for irradiation-creep behaviour, while related work on oxidation examined both short-term air and steam ingress, as well as the effects of their chronic exposure on graphite, and potential alleviating effects of boron additions on oxidation behaviour. Data to support graphite model development was generated in the areas of microstructural evolution, irradiation damage mechanisms, and creep. Support was provided for both ASTM and ASME development of the codes and standards required for use of nuclear graphite. Multiaxial fracture testing, at both the laboratory and component scale, as well as analysis of graphite was performed.

Examination of high-temperature alloys (800H and 617) provided very useful information for their use in heat exchanger and steam generator applications. Alloy 800 studies included a detailed evaluation of the existing historical data base and an extension of it through creep, creep-fatigue and relaxation to testing to 850°C, as well as corrosion tests in VHTR helium. Reviews of the operational history of the use of alloy 800H in steam generator and heat exchanger applications was performed and extended through fabrication studies, actual heat exchanger mockup preparation, and subsequent testing.

Significant studies on the thermo-physical, mechancal, and fracture properties of alloy 617 were performed as part of the development of the information required to include it in the ASME Code as an additional material for use in construction of high-temperature reactor components. This included studies of creep and creep-fatigue on both basemetal and weldments, plus crack-growth behaviour studies.

Other metallic materials were also examined as part of the PA. Irradiation and irradiation creep was studied on 9Cr-1Mo ferritic-martensitic steels and oxide-dispersion-strengthed steels, plus creep behaviour was examined in 2.25Cr-1Mo steel for steam generator applications.

In the near/medium term, metallic alloys are considered as the main option for control rods in VHTR projects, which target temperatures below about 850°C. However, future projects are considering the use of ceramics and ceramic composites where radiation doses, environmental challenges, or temperatures (up to or beyond 1 000°C) will exceed capabilities of metallic materials. This is especially true for control rods, reactor internals, thermal insulation materials, and for gas-cooled fast reactor fuel cladding. Limited work continued to examine the thermomechanical properties of SiC and SiC-SiC composites and oxidation in C-C composites, to develop testing standards and design codes for composite materials, and to examine irradiation effects and fabrication methods on ceramic composites for these types of applications.

Hydrogen production

The Hydrogen Production Project Arrangement has been signed by Canada, France, Japan, Korea, the United States and Euratom. For the past four years, China has been a candidate for joining the PMB. Active participation in the PMB HP has evolved considerably over five years. The French representation, missing between 2010 and 2014, is active again and present works focused on high-temperature electrolysis. The US participation has recently been less active. Fortunately the participation of Asian countries (mainly Korea and China as a candidate) and Canada remained very active.

The main activities overseen by the HP PMB deal with the thermochemical cycles (Sulphur Iodine (SI) Cycle, Hybrid Sulfur Cycle, other cycles) and the high-temperature steam electrolysis (HTSE).

Japan, Korea and China are strongly involved on SI developments. Japan’s representative could not join the two technical meetings in 2014 but stayed very active on SI systems coupled to the HTTR reactor.

Korea has engaged an experimental programme on a Sulphur Iodine Integrated system producing 50 NLH2/h. After a serial of separate tests for the Bunsen section unit and partial Integration Test for section 1 and 2 until the end of August 2014, KAERI succeeded in continuous operation of the integrated facility for eight hours at the beginning of September. Through this operation, almost constant hydrogen production was observed at the hydrogen production measuring gauge. KAERI is currently preparing 72 hour continuous operation with the same integrated facility.

In China, a bench-scale integrated SI facility named IS-100 was set up and successfully operated to achieve stable operation of the SI cycle with H2 rate of 60NL/h for 86 hours. During the operation of this facility, major key parameters of three sections (Bunsen, SA, and HI) were monitored and measured.

Concerning the Cu-Cl cycle, Canada works on one of the main key issue consisting in preparing a suitable membrane, designing an optimal cell, and finally integrating the Cu-Cl cycle. A new double membrane electrolyzer design for CuCl/HCl electrolysis, which can mitigate copper species crossover into the cathode for a finite period of time, was proposed from the previous understanding for the transport of Cu species. The experiment of the double membrane cell (DMC) showed that the DMC can maintain copper concentration in Cathode at a small amount level, whereas in case of SMC the copper concentration constantly increased with time. This DMC was shown to have good cell performance till 1600h of duration of life. A conceptual integrated-Cu-Cl cycle design diagram has been achieved.

With regards to the HTSE activities, France, Canada and China have shown new results. The modelling of the integration study of the HTSE with Canadian reactors was performed in collaboration between CNL (Canadian Nuclear Laboratories) and INL (United States). To do this, experimental procedures have been developed to produce different potential materials to be used as anode and cathode in the electrolytic cell.

In France, CEA has developed a low-weight and low-cost stack design, which was validated at several scales and in different running modes (HTSE, Co-electrolyze CO_2/H_2O, Fuel cell). The world 1st SOEC system based on this stack technology has been built and tested, including the heat recovery exchanger allowing hydrogen production directly from steam at 150°C. This first prototype could produce from 1 to 2.5 Nm3/h hydrogen. Comparison of operating points of alkaline, PEM and HTSE showed that the HTSE can be characterised as having a better efficiency and lower sensitiveness to the price of electricity, but higher cost for initial investment. Globally hydrogen produced is cheaper than with PEM or alkaline electrolysis, especially for high power plants.

Computational methods validation and benchmarks

The list of provisional signatories to the computational methods validation and benchmark (CMVB) PA evolved, as a reflection of the national programmes of the participants in the VHTR System Agreement. Provisional members are now China, Euratom, Korea, Japan and the United States. The VHTR SSC member from Switzerland expressed the wish to become an observer. No meetings of the CMVB PPMB took place since 2010, but the research activities in the member countries continued, with for instance the setup of new test facilities (HTTF, NSTF, MIR) and benchmarking activities in the United States, the setup and operation of 16 new test facilities and code development activities in China, code development and validation activities in Korea, and code development and validation activities in Europe. The members of the PPMB were re-confirmed in 2014. On 27 October 2014, the 11th provisional PMB meeting was held in Weihai, China, just before the HTR-2014 conference. In the meeting, the participants agreed to restart the CMVB activity, firstly by defining a research plan in the upcoming year, based on current research activities in member countries, then by fixing the next PMB meeting in April 2015 in China.

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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 including recently the Phenix end-of-life tests, and will be continued with the Astrid project in France, the restart of Joyo and Monju in Japan, the lifetime extension of BN-600 and the start-up of the BN-800 in Russia, and 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.

Figure 3.4: **JSFR (loop-configuration SFR)**

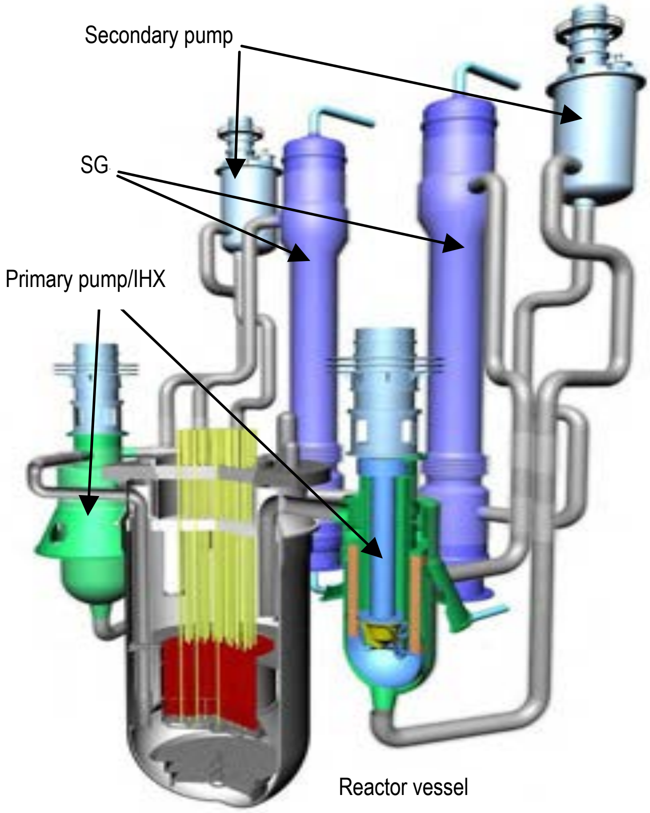


Figure 3.5: **ESFR (pool-configuration SFR)**

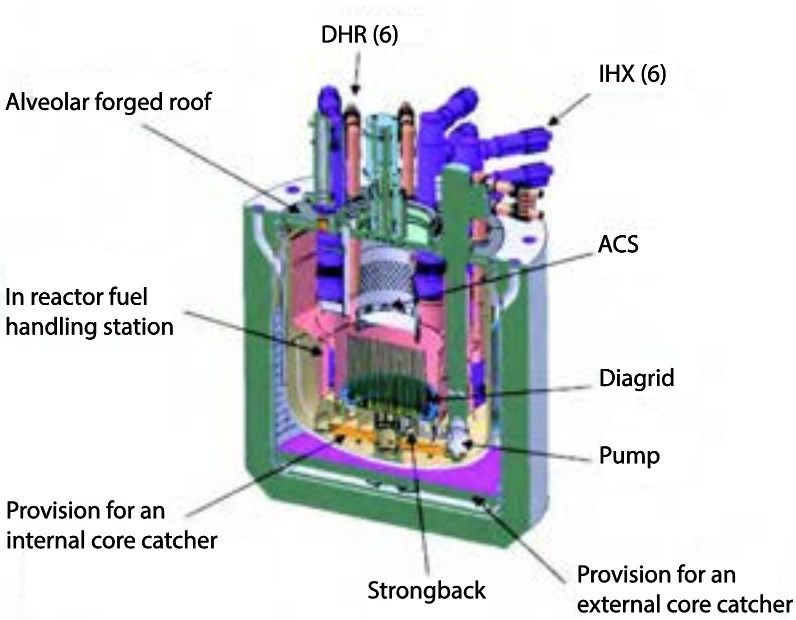


Figure 3.6: **KALIMER (pool-configuration SFR)**

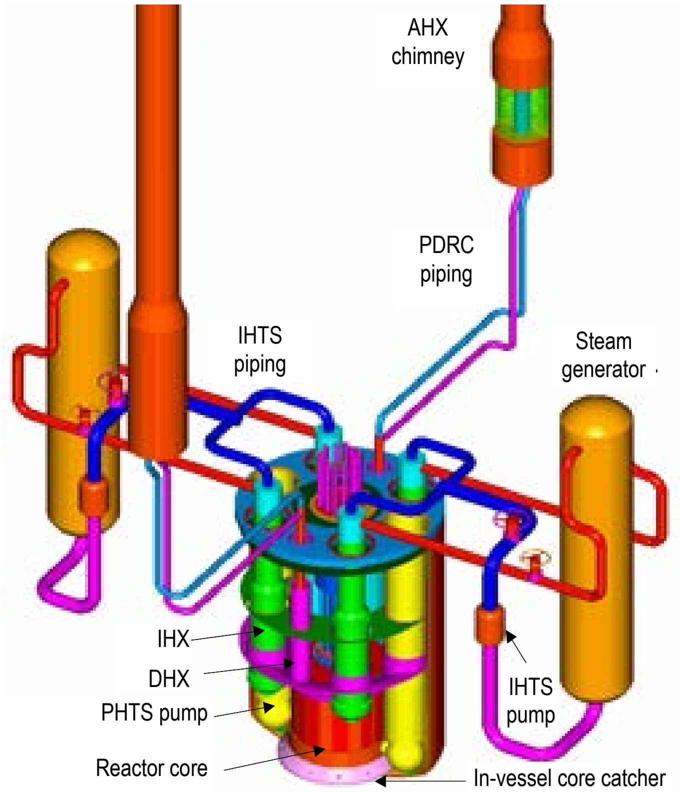
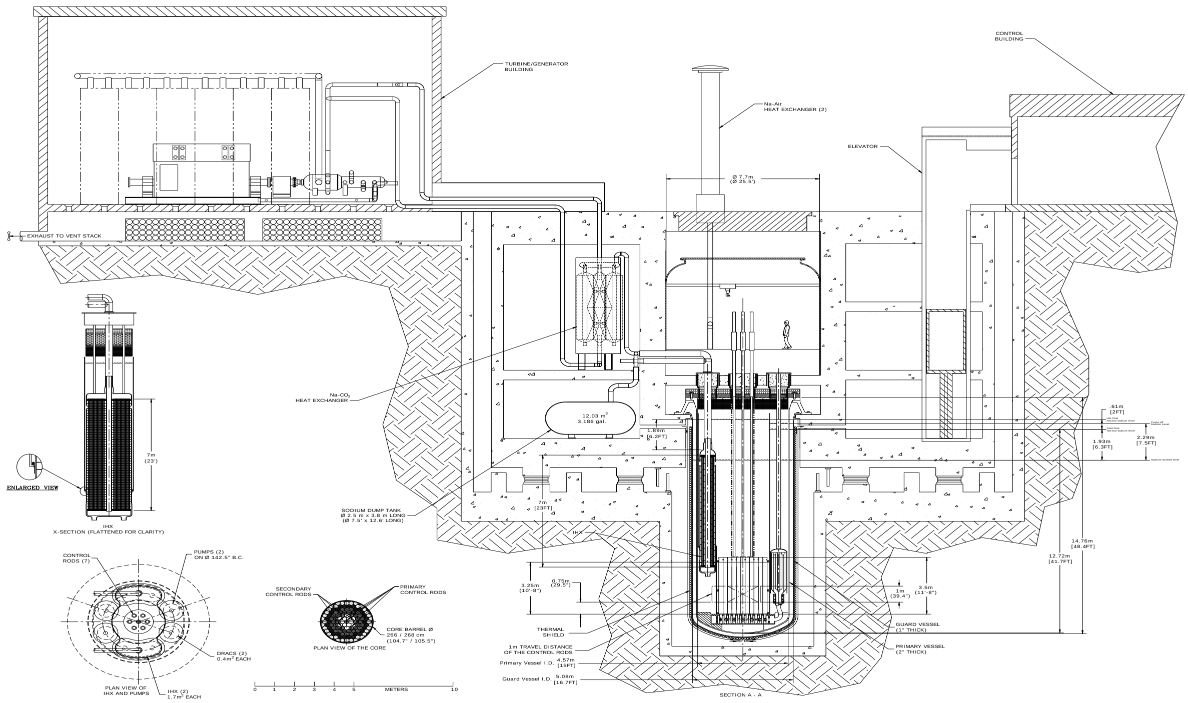


Figure 3.7: **SMFR (small modular SFR configuration)**



Status of co-operation

The system arrangement (SA) for the international R&D of the SFR nuclear energy system became effective in 2006 and the present signatories are:

- Commissariat à l'énergie atomique et aux énergies alternatives, France.
- Department of Energy, United States.
- Joint Research Centre, Euratom.
- Japan Atomic Energy Agency, Japan.
- Ministry of Education, Science and Technology, Korea.
- China National Nuclear Corporation, China.
- ROSATOM, Russian Federation.

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 and in 2014 was amended to extend the effective period by three years until September 2017. The Advanced Fuel membership Extension to China and Russia process was finalised in 2014 and the Project Arrangement amendment is under signature. The new CD&BOP PA includes a new member, Euratom, has already been drafted and legally checked inside NEA to be moved forward with the signature process. The Project Arrangement for Safety and Operation (SO) was signed in 2009 and amended in 2012 to include the contributions of Euratom, China and the Russian Federation. The Project Arrangement for System Integration and Arrangement (SIA) was signed by all members in October 2014.

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)

Through systematic review of the Technical Projects and relevant contributions on design options and performance, the SIA Project will help define and refine requirements for Generation IV SFR concept R&D. Results from the technical R&D projects will be evaluated and integrated to assure consistency. The Generation IV SFR system options and design tracks will be identified and assessed with respect to Generation-IV goals and objectives.

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 and 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, Korea and United States.
    - **2012:** Research collaboration between China, France, Japan, 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, Korea and United States. (Collaboration with Korea started in 2009).
    - **2012:** Research collaboration between China, France, Japan, 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, Korea and United States.
  - **2012:** Research collaboration between Euratom, China, France, Japan, Korea and United States**.**

- AF Project:
  - **2007-2012:** Viability study of proposed concepts.
  - **2009-2015:** Performance tests for detailed design specification.
  - **2014-2016:** Demonstration of system performance.
  - **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:** Amendment No.2 of Project Arrangement approved. Planning and design of Joyo and Monju fuel irradiation tests.

3.2.3 Main activities and outcomes

System integration and assessment (SIA) project

After several years to create the unique Project Arrangement and a delay to allow all Members to engage in at least one of the SFR Technical Project, the SIA Project Arrangement was signed by all seven Members of the SFR System Arrangement. The official start date for the SFR SIA Project is 22 October 2014; with the first Project Management Board meeting held on 4-5 November 2014.

At the PMB meeting, the SFR system options and design tracks in the System Research Plan were confirmed, and the comprehensive list of SFR R&D needs was updated. The technical Project activities were reviewed with a focus on recent technical accomplishments.

As described in the initial Program Plan, trade study contributions in 2014 included design track studies on transruranic transmutation in KALIMER (ROK) and ESFR design options downselection (Euratom), and general studies on modular design options (CEA), supercritical CO_2 energy conversion comparison (DOE),and thorium fuel conversion (DOE). An economic self-assessment of JSFR was contributed by JAEA.

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 2014 Annual Work Plan have been summarised as follows:

WP SO 1: Methods, models and codes

For the preliminary assessment of Ex-vessel DHR systems, a simplified model was developed to perform fast evaluations of the system performance in support to the design avoiding CDF approaches at the very first stage. The model, coupled with system simulations on long-term transients of about 100 hours, considers conduction, convection and radiation phenomena through the reactor pit. Additional parametric studies to assess the influence of the primary vessel emissivity were carried out and concluded on an insufficient performance unless the emissivity is quite high (in the range 0.7-0.8). The option of filling the main safety vessel cavity with sodium was considered and new models were developed. The study of such option on a severe accident transient without any decay heat removal system operating inside the primary vessel showed acceptable maximum temperatures.

An advanced sodium fast reactor system analysis module is being developed based on advanced numerical methods in a modern finite-element framework. The module is capable of plant-scale simulations for scoping, safety analysis, and licensing support. A key objective is to develop fast-running transient analysis capabilities that significantly improve upon traditional systems analysis codes. Advanced physical models have been developed for single-phase sodium flow. The numerical methods are second-order accurate in both space and time, and the use of higher-order finite elements produces extremely accurate results with minimal spatial discretisation. Use of modern computational frameworks has also enabled the development of coupling between the new system module and high-fidelity methods such as computational fluid dynamics. Coupled system/CFD results have been demonstrated for the Advanced Burner Test Reactor conceptual design. Future plans include coupling the system analysis module into the SAS4A/SASSYS-1 safety analysis code maintained by Argonne National Laboratory.

Phenomena identification and ranking table (PIRT) exercise on source-term phenomena in generic SFR primary and secondary containments during postulated severe accidents was performed. The focus of the exercise was on the identification of source terms potentially being released in accidental sequences, the evaluation of the importance of the phenomena on the evolution and consequences of the accident and the review of the available tools and proposals for new models. Firstly, the PIRT process was applied to source-term phenomena in the primary and secondary containments. Then, the present modelling capabilities of the identified and ranked phenomena have been evaluated. The main outcome of the PIRT exercise is the identification of phenomena that are poorly known but that are thought to play an important role in governing contaminant transfer.

A work on CDA analytical methods development to simulate phenomena in self-levelling behaviour of the debris bed was conducted. The analytical methods are implemented in SFR safety analysis code. An experimental study of self-levelling behaviour, in which the particle bed behaviour driven by bubble inflow from the bottom of bed in gas-solid-liquid three-phase flow was observed, is analysed to validate the new methods. Simulation results well reproduced the transient changes of particle bed, whose elevation angle and form deformation becomes gradually small and obscure, respectively. The assessment results show that these methods provide a basis to develop analytical methods of self-levelling behaviour of debris bed in the safety assessment of SFRs.

Preliminary risk assessment methodology against extreme snow was developed. The snow hazard indexes are the annual maximum snow depth and the annual maximum daily snowfall depth. Snow hazard curves for the two indexes were developed using 50-year weather data at the typical sodium-cooled fast reactor site in Japan. For each snow hazard category, accident sequences were evaluated by producing event trees that consists of several headings representing the loss of the decay heat removal. In this work, the snow risk assessment showed less than 10-6/reactor-year of core damage frequency. The dominant snow hazard category was the combination of 1~2 m/day of snowfall velocity and 0.75~1.0 day of snowfall duration. Sensitivity analyses indicated important human actions, which were the improvement of snow removal velocity and the awareness of snow removal necessity.

SAS4A code model development work was conducted to perform severe accident analysis for PGSFR. The pre-transient metal fuel characterisation model SSCOMPA has been developed and integrated with the SAS4A code. The formation of radially intermediate fuel regions with component change was investigated. The molten fuel cavity geometry at the time of transient model initiation was analysed.

Continuation of the assessments of the ULOF and UTOP accidents for SFR core with nitride and MOX-fuel is carried out by using the COREMELT code developed in the SSC RF-IPPE for analysis of severe accidents in the SFR core. In particular, a comparative study of the ULOF and UTOP accidents for SFR core with nitride and MOX-fuel is performed. The influence of sodium void reactivity effect on SFR safety under the ULOF and UTOP accident conditions is further investigated. Activities on further development of the COREMELT code are continued.

WP SO 2: Experimental programmes and operational experiences

At the CEFR, the loss of off-site power test was performed. The initial steady state condition of the test is 40% nominal power operation with electricity power generation. The test activated the reactor scram, two primary pumps coast down to lower speed and the air dampers of PDHRS fully open to initiate decay heat removal. The analysis of the preliminary test was results analysis was conducted.

Further, CEFR PDHRS Performance Demonstration Test was conducted at 40% nominal power steady state. The test results show that the performance of the two loops of PDHRS is unsymmetrical, and heat transfer power of loop 1 is over two times than loop 2. In addition, the heat transfer tube temperature of DHX tends to be colder when air damper opening is larger. The results suggest for loop 1 30% opening and for loop 2 10%. The total heat transfer power of natural circulation is demonstrated to satisfy the design requirement. The reasons for the imbalance of the performance of the two loops are under investigation.

The electromagnetic computer code 3D-RFECT that is used in ISI of steam generator (SG) tubes for FBR, based on eddy current technique (ECT) is further developed including investigations of specific issues and validation activities. The code simulates the electromagnetic ECT sensors used in detection of defects in SG tubes and it was used to analyse the signal from a full circumferential groove and partial outer tube defect located near SG support plate (SP).

An enhanced multi-frequency ECT technique, named “Window Multi-Frequency”, was developed to reduce signal from and support plate.

Validations of the technique for tubes similar with SG tubes of Monju were conducted using both numerical FEM simulations and experimental measurements in a small tank mock-up. The codes simulation results were validated against experimental measurements for single ECT frequencies. Validations of “Window Multi-Frequency” algorithms to suppress SP were also validated using FEM simulations or experimental from a mock-up tank with SG tubes similar as in Monju FBR.

The construction of experimental facility to evaluate performance of a passive decay heat removal circuit was completed, and experimental performance test on DHRS was conducted. Safety licensing work was finalised and related works such as firefighting equipment expansion, texture ceiling installation and fire lane construction were conducted. And pre-service inspection works such as instrumentation test, leak test, repair and maintenance works were carried out. Preparation works for Na filling (installations of measurement devices in sodium tank lorry, thermal oil boiler) and purification are completed for STELLA performance test. Experimental performance tests for DHX and AHX in STELLA facility were conducted and test results in good agreement with code results were observed.

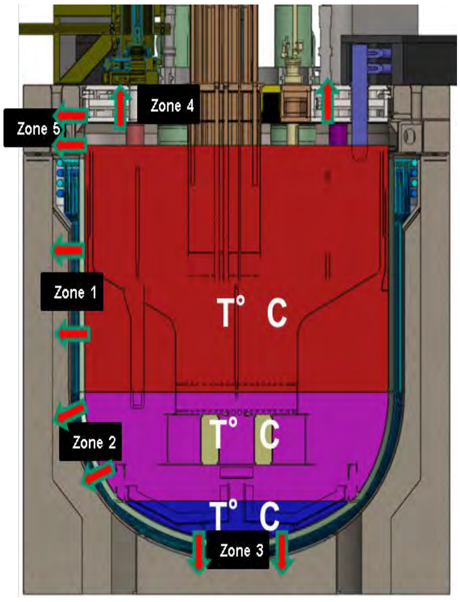
Activities on upgrading the AR-1 experimental facility were continued. The AR-1 facility is designated for investigation of sodium boiling modes. After completion of upgrading of the AR-1 facility preliminary tests are planned to be implemented with modelling sodium boiling in single fuel subassembly with seven fuel pin simulators.

WP SO 3: Studies of innovative design and safety systems

The use of additional shutdown systems based on passive features to cope with unprotected transients of ASTRID was studied. The first stage of the work was devoted to the analysis of the natural behaviour on the low sodium void effect core and the influence of reactivity coefficients during most representative transients. The impact of the implementation on shutdown systems based on hydraulic, Curie point and materials melting (SEPIA device) effects were studied using a CATHARE model of the reactor with a detailed core simulation. For the ULOHS, SEPIA and Curie point devices were compared using very conservative assumptions. The results show similar acceptable consequences in terms of core temperatures for the long term. The same conclusion was reached for the ULOSSP using the hydraulic device.

In order to demonstrate improved economics for sodium fast reactors, the development and demonstration of ultra-high burnup metallic fuel concepts is under investigation. The objective is to develop metallic fuel forms that are capable of safely achieving 40% burnup. The study proposed innovations decreasing fuel smear density to reduce fuel cladding mechanical interactions as solid fission products accumulate at high burnup, adding fuel alloy constituents to inhibit minor actinide and lanthanide migration to reduce fuel cladding chemical interactions, adding inner clad coatings to avoid eutectic formation at elevated temperatures, venting gaseous fission products to the primary coolant to reduce internal fuel pin pressure and reduce overall core and plant dimensions, and evaluation of uranium-molybdenum based fuel alloys. Each of these innovations has impacts on fuel performance and safety characteristics during plant operations and transients. The performance and safety characteristics for ultra-high burnup metallic fuel forms are evaluated.

Figure 3.8: **Ex-vessel DHR system modelling**



A model to simulate the Na-water reaction applied to the cleaning of components containing solid sodium in cells with inert or no atmosphere has been developed. The highly exothermal chemical reaction of sodium when brought in contact with water is an important safety issue for SFR systems, in particular, during decommissioning, when sodium needs to be firstly converted into non-reactive species. The main safety concern is the combustion of hydrogen in the surrounding air which according to the present work also may appear possible without air atmosphere. Since available knowledge does not allow a robust extrapolation of existing data/model to full scale plants, the study provided the details of the phenomenology, especially at the sodium/water interface, isolated the leading mechanisms and proposed a robust and innovative modelling approach. A large body of yet unreleased experimental data extracted from

CEA archives was collected and analysed on the basis of "explosion" physics. Some additional experiments were performed to fill some gaps, especially about the kinetics of the reaction. The results strongly suggest that the fast expansion of gas producing a blast wave in certain conditions is a kind of vapour explosion. It also appears that any potential hydrogen-air explosion could be strongly mitigated by the large quantity of water vapour emanating also from the reaction zone. The limitations of existing modelling approaches are clearly identified and alternatives are proposed.

Figure 3.9: **Upgrading the AR-1 experimental facility**

Figure 3.10: **3D numerical simulations of the in-service inspection of steam generator tubes using Eddy current testing**

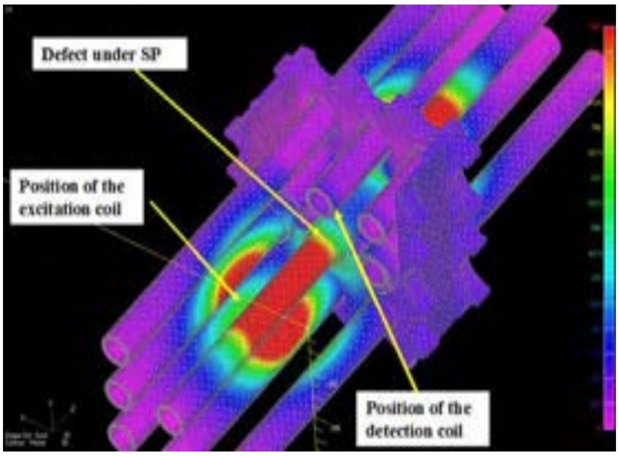


Figure 3.11: **STELLA construction and DHRS experimental test facility**

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 2014, irradiation test preparation and implementation, Post-Irradiation Examinations as well as calculations of fuel behaviour under irradiation, have continued regarding oxide and metallic fuels. In particular, Non Destructive Examinations for Am bearing oxide fuel irradiated up to very high burn-ups in ATR within the AFC-2 campaign have been completed. Preparation work has continued for an irradiation test up to a medium burn-up of U-Zr-type fuels in HANARO. The effect of the oxygen potential during sintering on $(U,Pu)O_2$ microstructure as well as the corrosion resistance of Am-bearing oxide fuels in liquid sodium have been investigated. New developments on fuel fabrication routes have been performed. Regarding cladding development, fabrication and characterisation of Ferritic/Martensitic cladding tubes have continued.

Component design and balance-of-plant project

The CD&BOP project started in October 2007 when the Project Arrangement was signed by the members of CEA/France, DOE/US, JAEA/Japan and KAERI/Korea. The CD&BOP activities include in-service inspection and repair technologies, LBB assessment technology and sodium heated steam generators. Supercritical CO_2 Brayton Cycle has been also studied as an advanced energy conversion system to the conventional steam Rankin cycle system. Details of each study are stated as follows:

Inspection technologies

The capabilities of CIVA simulation modelling tool in support of non-destructive examinations of SFR has been investigated by showing the potentialities on different applications (Figure 3.12).

The development of TUSHT and EMAT transducers has been continued. The objective is to launch the design and manufacturing of focused TUSHT in order to test later (>2014) its ability at under sodium viewing. The development of a new experimental bench able to insure movements of the transducers under the surface of liquid sodium has been launched (planned to be commissioned in 2015). And in parallel, a new multi phased array EMAT has been realised. Also, a new activity on eddy current flow meter has started with a synthesis of the experimental feedback of such measuring device.

The performance tests using an ultrasonic waveguide sensor has been executed as a different approach to under-sodium visualisation technology below:

- design and performance improvement of ultrasonic waveguide sensor;
- performance test of ultrasonic waveguide sensor in sodium.

Repair technologies

Remote repair techniques (Na removal, welding) with LASER techniques has been carried to confirm its performance. The feasibility of scouring techniques applied on metallic wall wetted by a liquid metal was demonstrated in 2014.

LBB assessment technology

The following tasks have been executed:

- tests of creep crack growth and fatigue crack growth for Mod.9Cr-1Mo structures;
- preparation of a summary progress report for 2014.

Supercritical CO_2 Brayton cycle

The results on corrosion study were synthesised.

The CFD simulations focused on physical aspects in a centrifugal compressor associate with a supercritical has been continued, in comparison with those in a classical compressor such as pressure and temperature gradients in specific locations (boundary layer, impeller inlet cross section) to clarify the phenomenology of the compression near the critical point.

The development and application of the Plant Dynamics Code to advanced SFR concepts has also continued.

The analyses of the dynamic operation and performance of SFRs with S-CO_2 Brayton cycle power converters and active control of SFRs has been continued (Figure 3.13). As part of control strategy development, sensor response times are an important feature that must be accounted for in the design. The G-PASS code has been used to investigate the effects of sensor response times on S-CO_2 cycle performance.

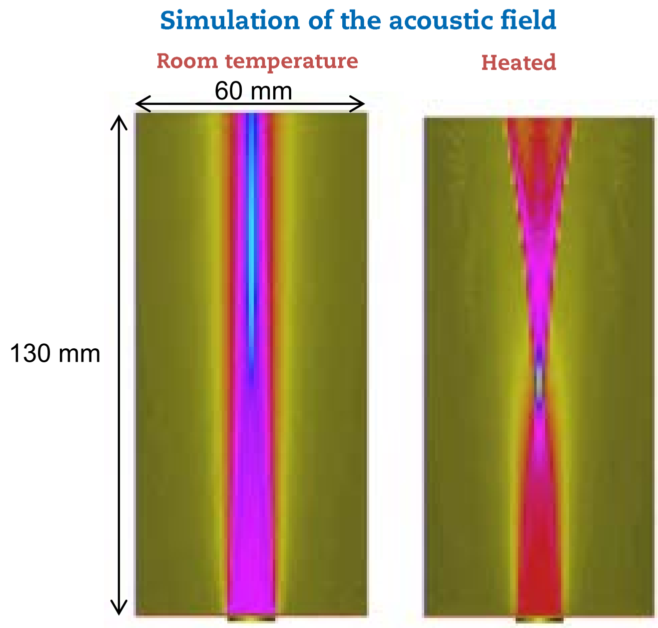
The material analysis has been performed to progress on the common understanding of material corrosion in CO_2 at high temperature and pressure condition representing the SC-CO_2 cycle operation conditions.

The feasibility of an S-CO_2 Brayton cycle power conversion system coupled to a pool-type SFR has been evaluated. In this design study, Sodium-CO_2 interaction has been identified as one of the significant key issues (Figure 3.14).

The knowledge in the field of sodium drainability of various components has been summarised in order to obtain a basis of comparison with the design choice of compact diffusion-bonded heat exchanger (size of channels).

Modelling improvements to the Plant Dynamics Code has been continued including modelling of dynamic effects and comparison with available suitable data from small-scale demonstration. The plugged cold trap circuit in the Plugging Phenomena Loop was replaced and installed an innovative plugging meter. Sodium plugging testing has been resumed with rerunning of the initial plugging test in the upgraded Plugging Phenomena Loop with the goal of nearly complete plugging.

Figure 3.12: **CIVA code simulation**



Steam generators

The sodium/water reaction control technology using nano-size metallic particles in sodium has been studied to increase the safety of steam generators. Since steam generators need to be designed by taking account sodium/water reaction, the suppression of the sodium chemical reactivity results in an innovative concept for highly reliable steam generators. The purpose of this study is to clarify the suppression of sodium/water reaction by nano-particles and to investigate its effect on the design of steam generator (Figure 3.15).

The structural integrity of the tube-sheet of the JSFR double-walled steam generator has been evaluated. Because the tube-sheet is subjected to severe thermal stresses during transient events such as a plant trip. In this study, a cyclic thermal loading test was performed using a semi spherical tube-sheet test model. After the test, the test model was inspected by liquid penetrant testing, scanning electron microscope and hardness testing to understand the tube-sheet failure mechanism. The elastic and inelastic stress FEM analysis was performed to reveal thermal stress distribution in the tube-sheet. The analysis results were confirmed by comparing with the tube-sheet failure of the cyclic thermal loading test.

As for SG tube inspection, the development of a remote field eddy current inspection technique and a magnetic sensor technique for single-walled tubes made of G91 steel for a Rankine-type SG has been conducted.

Figure 3.13: **SFR with S-CO_2 cycle power convertor**

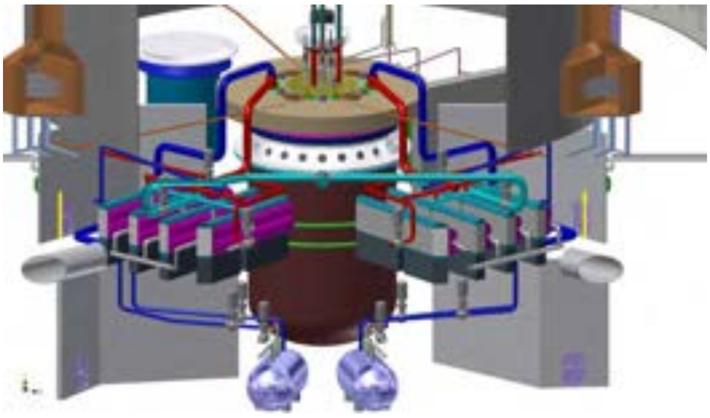


Figure 3.14: **Na-CO_2 interaction test loop**

Figure 3.15: **Na/water reaction technology using nano-particles**

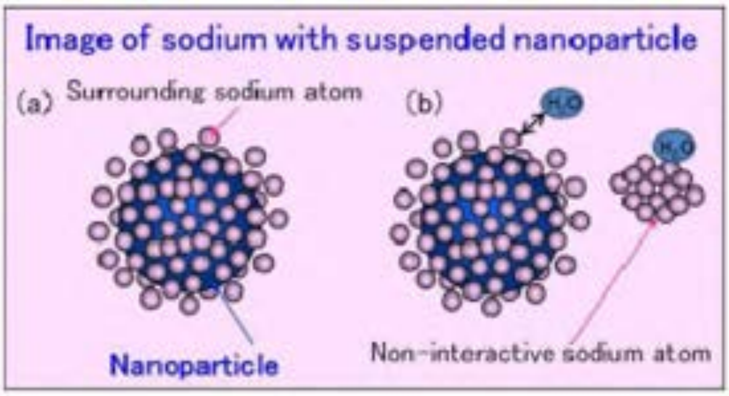
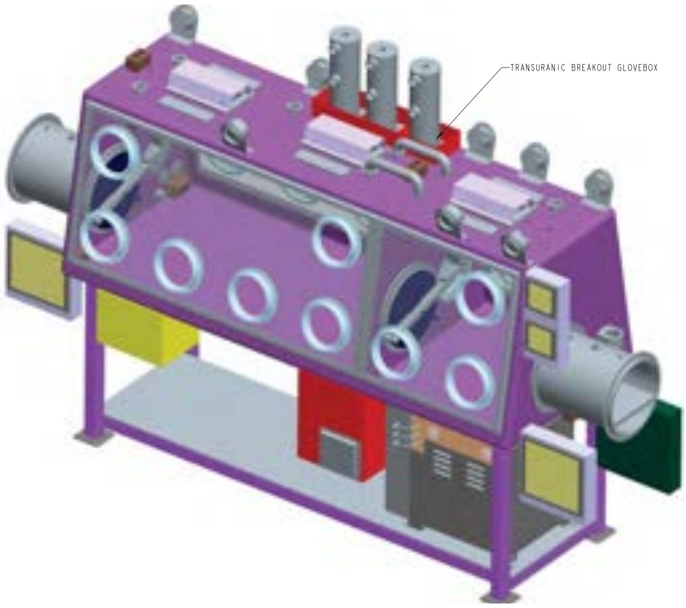


Figure 3.16: **Photograph of CEA fabricated (U,Pu, Am, Np)O_2 pellet (LEFCA facility)**

Figure 3.17: **Schematic of new Idaho National Laboratory transuranic glovebox for Am processing**



Global actinide cycle international demonstration project

The Global actinide cycle international demonstration project aims to show that 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. The irradiation experiment will resume after completion of Joyo repairs.

R&D on fabrication is in progress and the specifications of (U, Pu, Am, Np)OX, have been established at CEA. The overall programme on property measurements was defined and split between several laboratories. Figure 3.16 is a photograph of a uranium-americium oxide pellet fabricated by CEA.

The availability of americium is limited. Figure 3.17 shows a schematic of a new glovebox at Idaho National Laboratory that will be utilised to obtain and process americium supply.

References

Delage F., et al. (2014), "Progress status of the sodium fast reactor advanced fuel project within the Generation IV International Forum", Actinide and Fission Product Partitioning and Transmutation, 13th Information Exchange Meeting, Seoul, Korea, 23-26 September 2014.

3.3 Supercritical-water-cooled reactor (SCWR)

3.3.1 Main characteristics of the system

The 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 a pressure tube concept 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 high 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.

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 lists the members and shows the status of these PMBs. China signed the SCWR System Arrangement in 2014 and expressed its interest to join the Thermal-Hydraulics and Safety Project as well as the Materials and Chemistry Project. Projects plans are being update to include their planned contributions. The Project Arrangement for the Fuel Qualification Testing has been drafted and is waiting for government approval. Nevertheless, both Canada and Euratom are collaborating informally on this project. China has also expressed interest to participate in the second phase of testing.

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 an 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 an 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

Canada has completed the Canadian SCWR concept, which is a light-water cooled, heavy-water moderated, pressure-channel type reactor. The reference reactor-core concept consists of 336 fuel channels providing about 2 500 MWth power (and 1 200 MWe at 48% efficiency) and a small concept with 108 fuel channels generating 300 MWe has also been developed. It is developed to operate at the pressure of 25 MPa and a mean coolant outlet temperature of 625°C to fully utilise the advanced high-pressure turbine developed for the fossil-power plant. Various components in the core (e.g. fuel channel, fuel assembly, calandria vessel, etc.) have been developed. The reactor plant layout, including safety system, refueling system, spent-fuel storage, etc., has been established. A safety analysis of key postulated accident scenarios, such as large-break loss-of-coolant accident and station blackout event, has been completed. The Canadian SCWR concept has been assessed and demonstrated improvement on the GIF technology goals on safety, economics, sustainability and proliferation resistance. A review of the Canadian SCWR concept has been scheduled with Canadian nuclear industry (including utilities, manufacturers and regulator) in February 2015 and with international experts in October 2015.

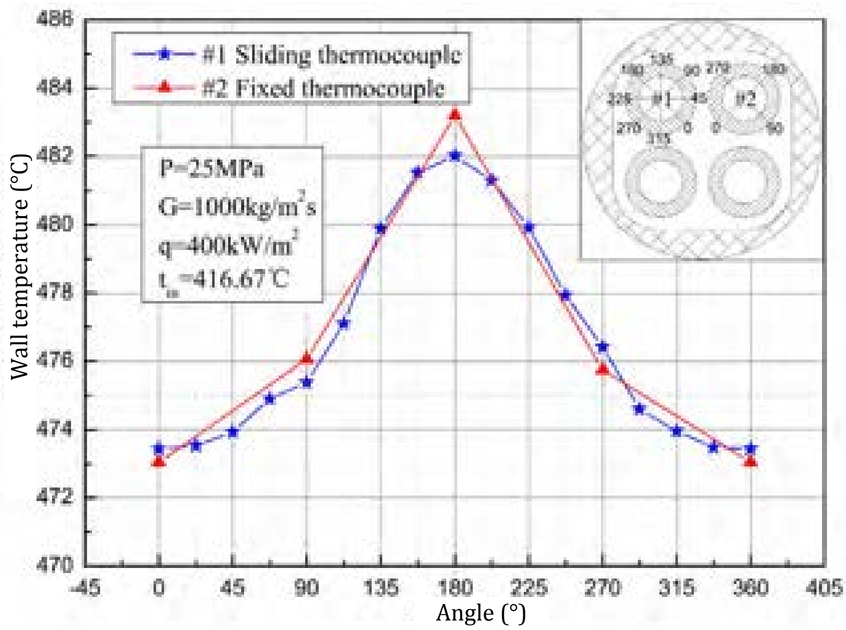
Thermal-hydraulics and safety

The thermal-hydraulics and safety projects in the Canadian National Program for Gen IV Energy Technologies have been established to (i) provide relevant experimental data for verification and validation of prediction methods and analytical toolsets, and (ii) improve the accuracy of prediction methods in support of fuel assembly optimisation and safety analyses. Several experimental projects are currently being carried out to obtain heat-transfer data with annuli, 3-rod assembly, and 4-rod assembly in refrigerant-134a flow, carbon dioxide flow, and water flow, and blow-down and natural-circulation data with tubes in water and carbon dioxide flow, respectively. These experimental data have led to improved understanding of the thermal-hydraulics phenomena and enhanced the prediction accuracy of parameters.

Heat-transfer experiments have been performed with supercritical water through a 4-rod (2×2) bundle to provide circumferential wall-temperature measurements around the heated rods. These experiments consist of two phases: the first phase focuses on the bundle configuration with no spacing device (i.e. bare bundle) and the second phase on the bundle configuration with the wrapped-wire spacers. Figure 3.18 illustrates the circumferential wall-temperature distributions around the heated tubes of the 4-rod bundle without spacers. The presented wall temperatures correspond to outer-surface values calculated from inner-surface measurements obtained at a location 500 mm from the start of the heated length. Wall temperatures at the corner region (around 180°) are higher than those in other regions. The increase in wall temperature at the corner region is attributed to the small gap with low flows and high enthalpies lowering the heat-transfer coefficient. The temperature gradient between the corner and the centre subchannel (where the lowest temperature is observed) regions is about 9°C. This signifies that the wall temperature at the corner region increases more rapidly than that at the centre subchannel region. Overall, the temperature variations from 0°-180° and from 180°-360° are relatively symmetrical. This signifies no tilting or bowing on the heated rod at this location. Measurements are similar between moveable and fixed thermocouples.

Figure 3.19 illustrates the circumferential wall-temperature distributions around the heated tubes of the 4-rod bundle with the wire-wrapped spacers. The overall variations of the wall temperature around the wire-wrapped rods are similar to those around the bare (without the wire) rods. However, the wall temperatures for the wire-wrapped rods are mostly lower than those for the bare rods, especially at the peak-temperature location (i.e. around 180°). Those areas where higher temperatures are observed for the wire-wrapped rod, correspond to the location below the wire, where flow stagnation, could be encountered. Peak temperature was also observed at the vicinity of the narrow-gap area (i.e. 180°).

Figure 3.18: **Circumferential wall-temperature distributions around two heated rods of the 4-rod bundle without spacers**



Heat-transfer experiments were performed with supercritical carbon-dioxide (CO_2) flow through a 3-rod bundle assembly. The bundle was constructed with three 10-mm OD Inconel-600 tubes having a heated length of 1.5 metres. Each rod consisted of an unheated copper section at each end for connecting to the power bars. The spacing between rods was 1.4 mm, resulting in a pitch-to-diameter (p/D) ratio of 1.14. It was maintained by wrapping a hypodermic stainless-steel tubing of 1.3 mm around each rod. To eliminate mal-distribution of flow in various subchannels, three unheated fillers were installed at the subchannels neighbouring to the pressure tube. A moveable thermocouple assembly was installed inside each heated rod. It consisted of a carriage rod, near the upstream end of which two insulated K-type thermocouples were mounted across from each other. A loaded spring pushed each thermocouple against the heated surface to ensure good thermal contact. Thermocouples were rotated within the rod over 360° and traversed along the rod.

Figure 3.20 illustrates the circumferential temperature variations around the three heated rods of the bundle. The circumferential temperatures are non-symmetrical with the peak temperature located at the subchannel between the heated rod and the unheated filler rod (insufficient data to confirm this observation for Rod C due to thermocouple malfunctions). The peak temperature locations for all rods appear tilting to one side and is possibly attributed to the winding direction of the spacer along the heated rod.

Figure 3.19: **Circumferential wall-temperature distributions around two heated rods of the 4-rod bundle with wire-wrapped spacers**

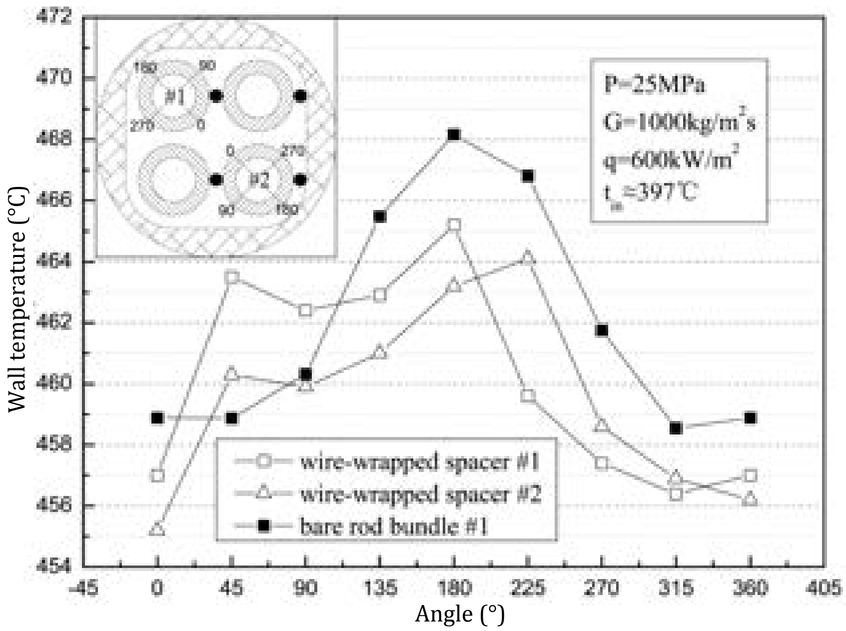


Figure 3.20: **Circumferential temperature maps obtained with the 3-rod bundle cooled with CO_2 flow**

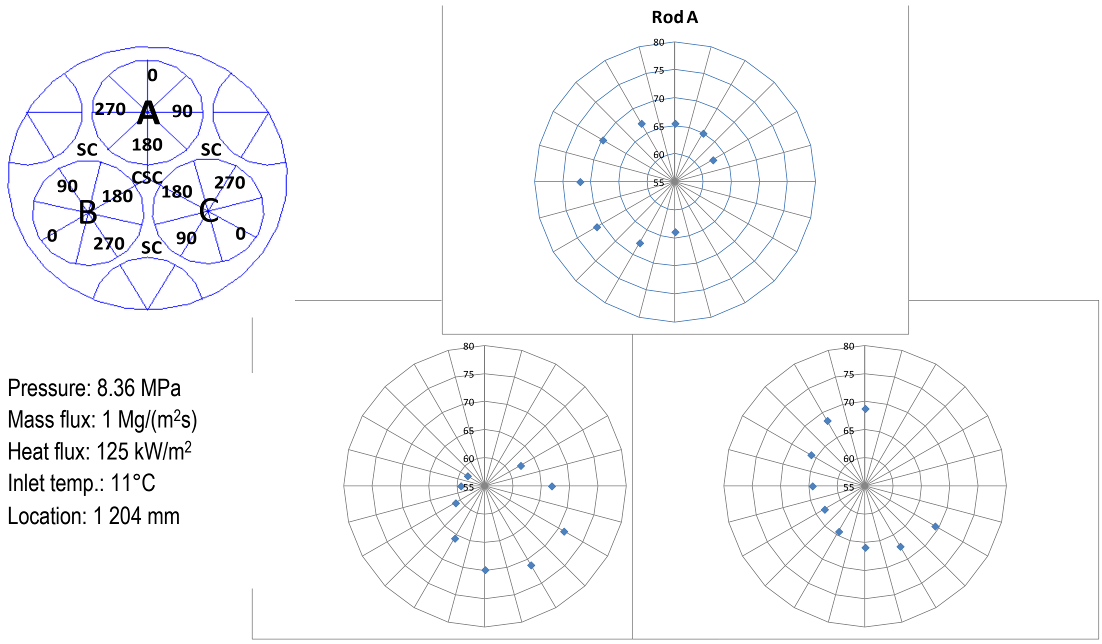


Figure 3.21 illustrates the axial surface temperature measurements at the peak temperature angle of each rod. Some fluctuations in surface temperature have been observed mainly due to the presence of the wire-wrapped spacer. It appears that no significant increases in wall temperature (corresponding to the deteriorated heat transfer phenomenon) were encountered over the heated rod. As indicated above, there are insufficient data to confirm the peak-

temperature angle for Rod C. The peak-temperature angle of 0° was established from the available data only.

The effect of a wire-wrapped spacer on heat transfer was examined using a heated annulus test section cooled with refrigerant-134a flow. The annulus test section consisted of an inner heated Inconel-625 tube of 10-mm outer diameter with a wall thickness of 2.5 mm and an outer unheated Inconel-625 tube of 18-mm inner diameter (see Figure 3.22). It had a heated length of 2.244 metres. A moveable-thermocouples assembly was installed inside the inner heated tube to measure the inner-wall temperature distribution. Four sets of discrete spacers were attached along the heated tube at locations of 85 mm, 856 mm, 1 122 mm (midpoint) and 2 105 mm. These spacers maintained the heated tube at the centre of the pressure boundary, and held down the stainless-steel (SS316) wire wrapped around the heated tube. The outer diameter of the wire was 1.2 mm. Two wire pitches (100 mm and 200 mm) were tested in the experiment.

Figure 3.21: **Axial wall-temperature distributions at the peak temperature angle of each rod in the supercritical CO_2 cooled 3-rod bundle**

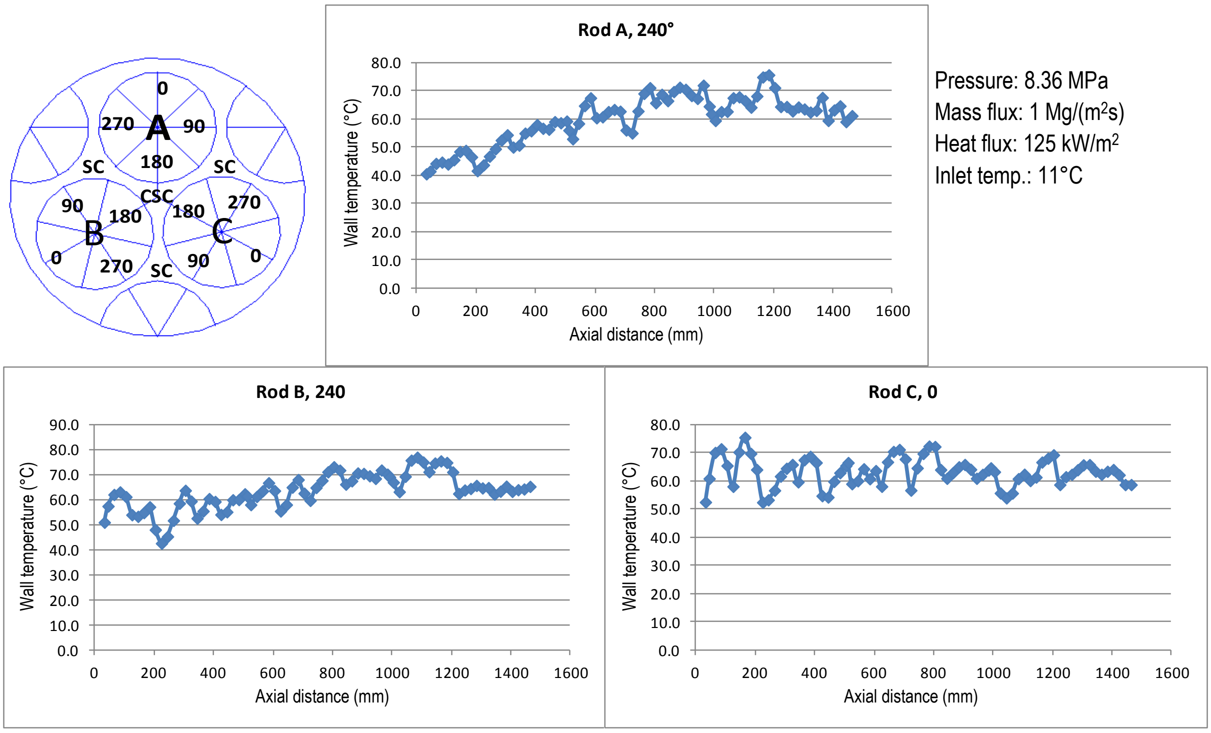


Figure 3.23 compares the experimental Nusselt numbers calculated from wall-temperature measurements along the annular test section with and without the wrapped wire for two mass fluxes. The Nusselt number is larger at the mass flux (G) of 1 300 kg/m²s than of 600 kg/m²s. Large localised disturbances in Nusselt number are shown at four locations of the discrete spacer. The enhancement effect of these spacers appears to have impact over a short downstream distance, particularly at the mass flux (G) of 600 kg/m²s. The installation of a wrapped wire along the heated rod has led to increases in Nusselt number (i.e. heat-transfer enhancement) compared to that of a no-wire rod. Reducing the wire pitch from 200 to 100 mm increases further the Nusselt number, but the difference is relatively small.

Figure 3.22: **Schematic diagram of the wire-wrapped annular test section for supercritical refrigerant experiments**

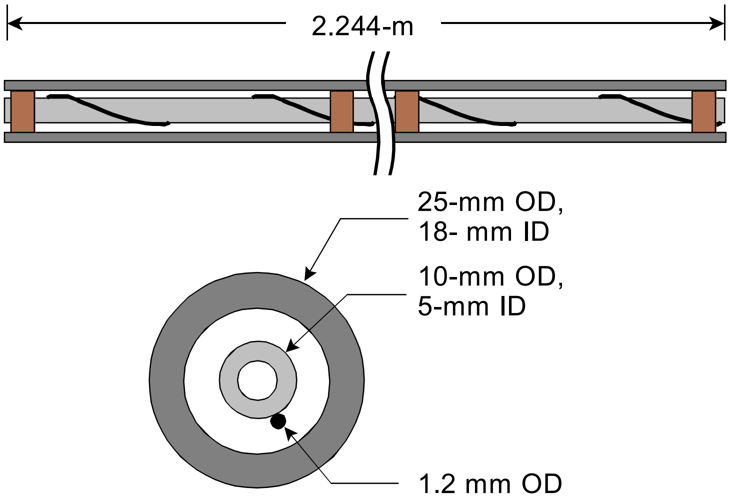
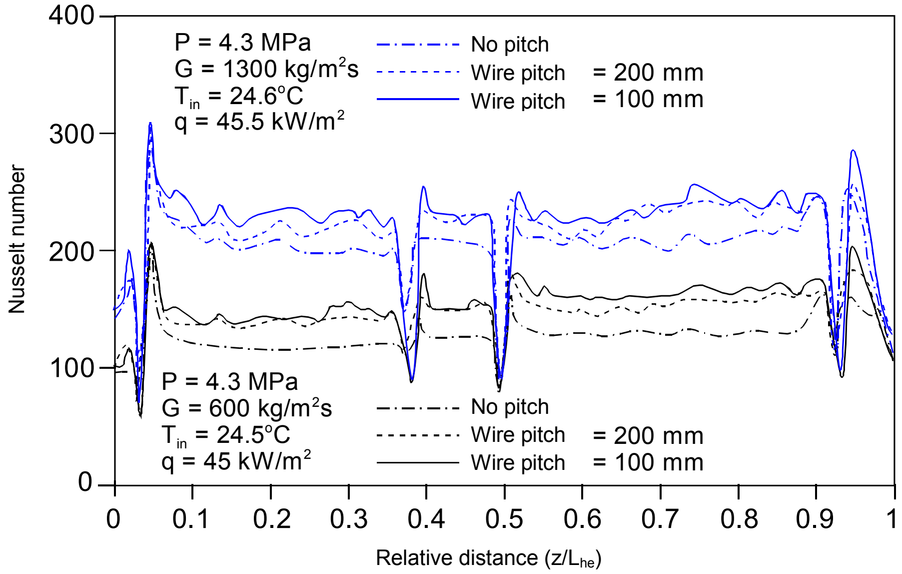


Figure 3.23: **Experimental nusselt number along the annular test section with or without wrapped wire**



Most of the activities in the EU have been focusing on heat transfer in supercritical water and other fluids such as Freons and CO_2. Heat transfer in supercritical fluids is a challenge to model, as properties such as density and specific heat capacity drastically change near the pseudo-critical point. Development of advanced models, that can be implemented in computation fluid dynamic (CFD) codes, are of paramount importance to accurately predict the wall temperatures of the fuel assemblies of an SCWR, such as the European high performance light water reactor (HPLWR).

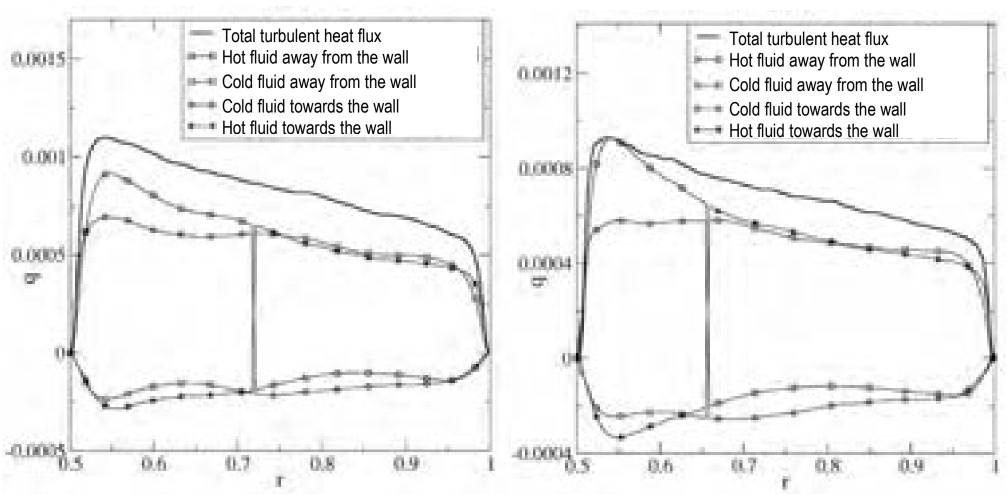
In 2010, the thermal-hydraulics of innovative nuclear systems (THINS) project was initiated. One of its work packages aims to study and improve the performance of existing turbulence models (or develop new ones if necessary). In addition, it focuses on the creation of a reference database based on the Large-Eddy Simulation/Direct Numerical Simulation (LES/DNS) and experiments.

Figure 3.24: **Supercritical jets facility at the Delft University of Technology, the Netherlands. The vessel, which contains six looking glasses for optical access for PIV and Infrared measurements, can withstand the pressure of 5.7 MPa (HFC23)**

An experimental setup, with Freon HFC23 as the working fluid, has been built to obtain detailed hydraulic characteristics of annular flow and of three jets impinging on a wall (representing the phenomena possibly encountered in the upper plenum of the HPLWR). Laser techniques (Laser Doppler Anemometry and Particle Image Velocimetry [PIV]) and a fast infrared camera will be used to measure near wall velocities and temperatures of the wall. The latter experiment aims to study thermal fatigue in supercritical flows. Experimental results are anticipated in the beginning of 2015.

A five-year Dutch research programme (funded by the Dutch Technology Foundation STW, "Stichting Technische Wetenschappen") 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. Geometries such as annular flow and a Rayleigh Benard cell will be used. Some results from a DNS of a supercritical CO_2 annular flow (with a hot and a cold wall) can be found in Figure 3.25.

Figure 3.25: **DNS of supercritical CO_2 annular flow in the case of a high and a low temperature wall. The graphs show the turbulent heat flux for different near-wall flow phenomena**



Source: Peeters, Pecnik, Boersma, Rohde and Van der Hagen, ETMM10 Conference Proceedings (2014).

Figure 3.26: **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 Hungary, Italy and the Netherlands, research is performed on the steady-state characteristics and stability of natural circulation, supercritical loops. These studies make use of codes and research facilities such as the Dynamic Radiography Station of the Budapest reactor and the ANCARA and DELIGHT of the Delft University of Technology.

In 2013 and 2014 an international benchmark study on supercritical heat transfer was organised by the Delft University of Technology and supported by GIF. Ten participants from the EU and Canada performed *blind* calculations on supercritical water flow through a heated seven-rod bundle. The Japan Atomic Energy Agency (JAEA) provided the operational conditions beforehand and the experimental data afterwards in order to prevent any prior knowledge during the course of this study. During a workshop in Delft in June 2014, all data were revealed to the participants. Results will be presented at the 7th International Symposium on SCWR (ISSCWR-7) in Helsinki in 2015.

Figure 3.27: **Participants of the international benchmark on supercritical bundle flow in Delft, the Netherlands**

Materials and chemistry

The M&C PMB has been focusing on selection and qualification of candidate alloys for all key components in the SCWR. This includes general corrosion and stress corrosion cracking tests in autoclaves and loops as well as development work on test facilities. In addition, modelling of oxide film behaviour has been performed to better understand the fundamentals of general corrosion resistance.

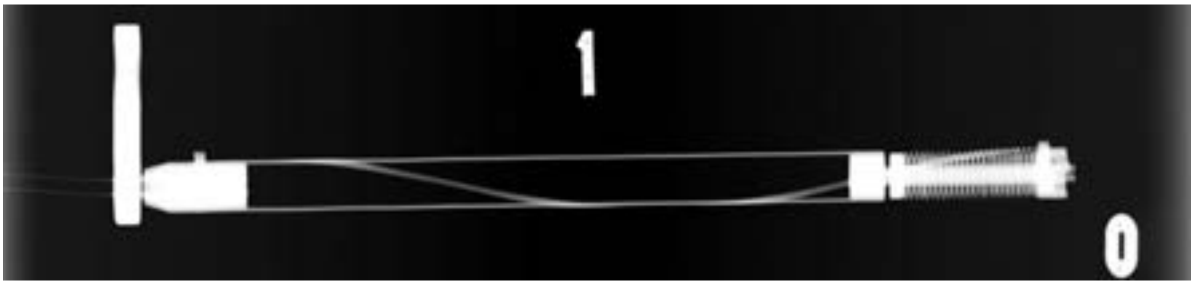
A major activity of the M&C PMB has been the organisation of a Round Robin corrosion test exercise between PMB partners (Canada, Japan and Euratom) to compare the results of corrosion tests in different test facilities. Each laboratory used a standard test protocol and coupon preparation method with the coupon materials originating from the same batch. The tests were completed in 2013 and the results have now been analyzed and will be reported by the end of 2014. It was concluded that considerably more variation in the data was observed than was expected.

At JRC-IET Petten and VTT, Finland, SCWR research was conducted within EU FP7 project SCWR FQT with three main objectives: assessment of general corrosion and SCC resistance of selected materials including structural integrity assessment of a fuel rod mock-up in case of LOCA. The final general corrosion tests were focused on assessment of the effect of surface finish and water chemistry. Corrosion exposures up to 3 600 h duration were carried out at 550°C/25 MPa with 150 or 2 000 ppb of dissolved oxygen or with 1.5 ppm of dissolved hydrogen. The effect of surface finish was studied using plan-milled, sand-blasted and standard polished specimens. The results indicated no significant benefit using hydrogen water chemistry compared to tests with dissolved oxygen. The beneficial effect of surface cold work due to sand-blasting or plan-milling was clearly demonstrated. The SCC resistance of austenitic stainless steels 08Cr18Ni10Ti (equivalent of AISI 321), AISI 347H and AISI 316L was investigated by conducting slow strain rate tensile tests (SSRT) at 550°C in SCW. In addition, development and assessment of a reference electrode for corrosion potential measurement in sub- and supercritical water and evaluation of crack growth rates of austenitic stainless steels in sub- and supercritical water using pneumatic bellows-based loading devices were performed within the internal project IntAg LWR (Integrity and Ageing of present light water reactors and future water-cooled reactors) at JRC-IET.

SCC susceptibility was evaluated using three criteria: loss of mechanical properties compared to those in inert environment, SEM analysis of the main fracture surface, and SEM analysis of secondary cracks. Crack growth rate tests were performed under well-defined stress conditions following the NULIFE guidelines with instrumentation allowing *in situ* crack growth measurement. The focus was on the temperature range close to the critical point of water where the highest susceptibility to SCC could be expected. 316L was selected as the best performer based on the SSRT and SEM results. Initial crack growth test results showed no significant increase of crack growth rates as the temperature increased through the critical temperature.

Three fuel rod mock-ups were manufactured by CVR, Czech Republic (Figure 3.28). Three types of test were perfumed at JRC-IET based on possible accident scenarios. Test 1 simulated a Loss of Coolant Accident. The autoclave was rapidly depressurised to $p_{Aut} < 1$ MPa while the internal pin pressure was held at 20 MPa. Test 2 simulated loss of internal pressure. The third test simulated long-term operation of the fuel pin mock-up at operational SCW parameters with temperature $t = 450$°C and $p_{Aut} = 25$ MPa. Both p_{Aut} and p_{Pin} were held constant for more than 600 h, i.e. $p_{Aut} = 25$ MPa, $p_{Pin} = 15$ MPa.

Figure 3.28: **Radiographic 2D X-ray image of the fuel rod mock-up (test 1) taken before depressurisation test in supercritical autoclave**



An iron/iron oxide electrode was developed by IFE OECD Halden Reactor Project for *in situ* corrosion monitoring up to 700°C in SCW. The first two prototypes were installed in JRC-IET SCW autoclave. The first tests focused on electrochemical potential (ECP) measurements of AISI 316L RC(T) vs. Fe/Fe_3O_4 in sub- and supercritical water up to 600°C. Long-term electrode stability and sensitivity to changes of dissolved oxygen content were evaluated. Both sensors survived long-term SCW exposure to 600°C, but further investigations are needed, in particular verification of impedance characteristics.

At VTT, Finland, the overall objective of Academy of Finland projects NETNUC (New type nuclear reactors, 2008-2011) and IDEA (Interactive modelling of fuel cladding degradation mechanisms, 2012-2016) has been assessment of the general corrosion mechanism using a deterministic model of the oxide layers. In 2014, the kinetic and transport parameters of inner and outer layer growth on two austenitic and two ODS alloys (Sanicro 28, 690, MA956, PM2000) exposed to SCW were estimated using an upgraded model that assumes that growth of the outer layer is governed by the transport of cations through the inner layer via an interstitialcy mechanism. The updated model can accurately reproduce the depth profiles of constituent elements in the inner and outer layers, as well as in the diffusion layer situated between the inner layer and the bulk substrate. Kinetic and transport parameters of inner and outer layer growth were estimated for oxidation times from 600 to 2 000 h at 650°C. Most of the rate constants and diffusion coefficients decreased with oxidation time at a constant temperature. A hypothesis based on microstructure evolution was proposed to explain the effect of film aging on these parameters.

China joined the SCWR M&C PMB in 2014 as observers. Nuclear Power Institute of China (NPIC), Shanghai Jiao Tong University (SJTU) and University of Science and Technology (USTB) are the major organisations actively involved in SCWR materials research in China. Each organisation has well-defined tasks and collaborations between them enhanced the Chinese research capabilities in 2014. Research was funded by the EU-China collaboration SCWR-FQT/SCRIPT project or by other EU or national projects from government.

NPIC announced a new CSR1000 conceptual design, and short-listed candidate materials for the fuel assembly. Stainless steels 316L and 310 are presently considered to be the major materials for building the test reactor. Technical specifications have been defined, and tests were conducted to evaluate the mechanical and corrosion performance of these materials in SCW. Four new test facilities, including two slow strain rate tensile testing machines, one corrosion fatigue testing machine, and a recirculating autoclave, have been setup for evaluation of SCWR candidate materials.

One crack growth rate testing system dedicated to stress corrosion and corrosion fatigue testing of candidate materials for in-core structure and pressure boundary materials in SCW was built in the corrosion laboratory at SJTU. This new as well as existing SSRTs and recirculating autoclaves were used to measure the general corrosion rate and SCC susceptibility of 316L, 347, HR3C, 310, 310-DOS, 18Cr-ODS steels in SCW, and the effects of dissolved oxygen and hydrogen on SCC of 316L.

At USTB, research mainly focused on design and fabrication of novel materials for SCWR cladding applications. 18Cr-ODS, 304-ODS, 316-ODS and 310-ODS steels were prepared by mechanical alloying and hot isostatic press sintering. Results of microstructure observation, mechanical tests and corrosion evaluation showed promising performance of 18Cr-ODS F/M steels and ODS austenitic stainless steels.

In 2014, the Canadian materials and chemistry programme focused on selection and evaluation of five candidate fuel cladding alloys (347 SS, 310 SS, Alloy 800H, Alloy 625 and Alloy 214) in preparation for a Check-and-Review of the Canadian concept by a panel of Canadian experts. A major step forward in fuel cladding material selection was made in 2013 by the adoption of a collapsible fuel cladding concept, which significantly reduced the mechanical properties requirements for the fuel cladding.

The effect of SCW density (pressure) on general corrosion was evaluated in comparative tests of 304 SS, 310 SS, A-286 and Alloy 625 at 625°C at 29, 8 and 0.1 MPa. For alloys with greater than ~20 wt.% Cr, superheated steam at temperatures well above the critical temperature is a reasonable surrogate for high density SCW. This is particularly important for the Canadian SCWR, where the peak fuel cladding temperature may be as high as 800°C; there are currently no autoclave or loop facilities that can operate at this temperature at 25 MPa. The ability to use steam data allows data on materials performance acquired in support of the American and Russian nuclear steam reheat programmes in the 1960s and 1970s to be used. A number of long-term (up to 5 000 h) corrosion tests were performed at temperatures between 450 and 800°C to fill in gaps in the available data. Analysis of these new data and literature data showed that all five candidates would give acceptable general corrosion performance for the 3.5 year lifetime of the fuel cladding if the surface finish is optimised. A major knowledge gap is the effect of flow on the corrosion rate. The effect of surface oxides (formed by base metal oxidation and by deposition from the coolant) on heat transfer was also identified as a knowledge gap.

Creep testing showed that the Larson-Miller Parameter (LMP) is suitable for creep strength predictions for the five candidate alloys; experimental results validated the LMP predictions made for SS 347H and SS 310S. The crept specimens were examined by TEM to investigate precipitation and deformation; 310S samples tested for creep at 800°C showed precipitation of sigma phase even in samples exposed to high temperature for short periods of time. Crept samples also contained $M_{23}C_6$ precipitates, which formed along grain boundaries or within grains, mainly on Ti(CN) precipitates as heterogeneous nucleation sites.

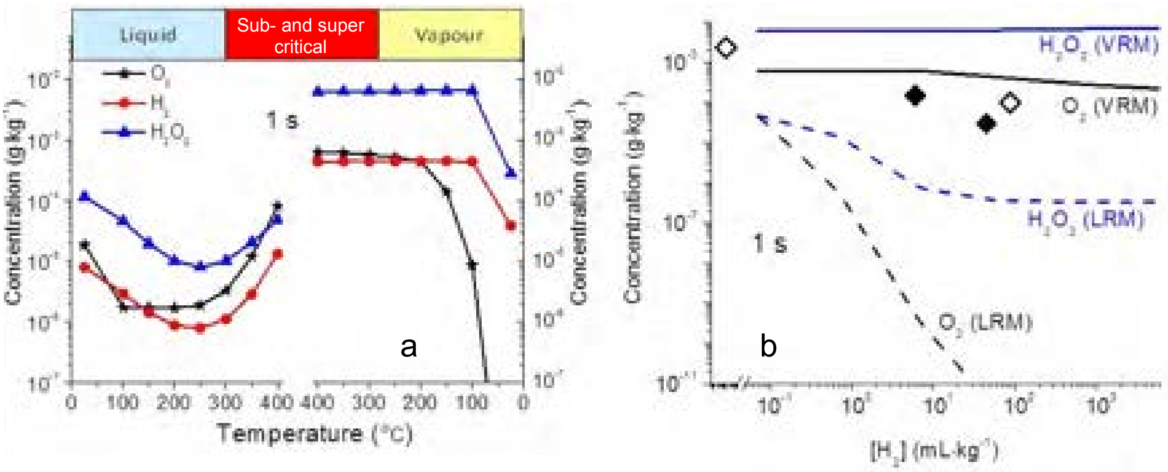
Under neutron irradiation, the ductility of stainless steels is significantly reduced, with elongation values of ~10% or less at temperatures above 700°C. The significant reduction in ductility after irradiation coupled with the approximately 1.5% total deformation expected over the lifetime of the Canadian SCWR fuel cladding defined a preliminary series of SSRT experiments. All samples strained to 5% in SCW at 500°C but without prior tensile cold-working showed no cracking. This was confirmed for 310S and Alloy 800H in tests at 625°C; in these tests samples were strained to 5% and then held at that strain level for at least 300 h. These results, coupled with the extensive experience with these alloys in superheated steam, suggests that SCC will not be a major issue for 310S and 800H when used in annealed condition (i.e. without cold-working) as the fuel cladding. This will need to be confirmed in in-pile testing during material qualification testing.

The damage dose in dpa was determined for the five candidate alloys and also over the fcc phase of the Fe-Ni-Cr ternary diagram. Two methods were used to obtain the number of Frenkel pairs produced from the displacement cascade: the standard Norgett-Robinson-Torrens (NRT) model and molecular dynamics (MD) simulations. The dpa values obtained by MD simulations were lower than those predicted by the NRT model, consistent with published data. The NRT model generally predicts increasing damage dose with increasing Ni for constant Cr levels. Damage dose derived from MD simulations is influenced by both the alloy composition-dependence of primary defect generation and reaction cross-sections. A low dpa region is observed for alloy chemistries between 20-30 wt.% Ni and 0-15 wt.% Cr. The amount of He generation for the candidate alloys was evaluated. The He concentration increases from ~1 appm at 0.5 years to 7-49 appm for the outer fuel pin and 16-121 appm for the inner fuel pin at 3.5 years (the in-service life of the fuel cladding). He generation was found to be approximately linear with the initial amount of ^{58}Ni.

The overall assessment of the five alloys in terms of their corrosion, SCC/IASCC, creep, ductility, strength and void swelling performance indicated that while none of the candidates can meet all of the performance requirements, there are sufficient data available for all but Alloy 214 to suggest that they be evaluated further in detailed material qualification testing. While Alloy 214 meets several of the performance requirements, the high nickel content and high cost, coupled with a lacklustre corrosion performance, suggest that it be dropped from further consideration.

Major advances were made in the modelling of water radiolysis at both the microscopic (reaction yields) and macroscopic levels. At the macroscopic level, a liquid radiolysis model (LRM) and a vapour radiolysis model (VRM) were developed using reaction sets similar to those in existing models for liquid water and water vapour radiolysis (Figure 3.29a). The model was then used to predict the effects of H_2 addition on concentrations of oxidising species in the Canadian SCWR core (Figure 3.29b). The VRM model was in good agreement with data from the Beloyarsk NPP in superheated steam.

Figure 3.29: **Concentrations of oxidising species produced by water radiolysis in the Canadian SCWR core predicted by the LRM and VRM models**



(a) Effect of H_2 addition on the concentration of oxidising species predicted by the models. (b) Symbols represent plant data from Beloyarsk NPP, which operated with nuclear steam reheat (open symbols: H_2 produced by radiolytic decomposition of NH_3, solid symbols: direct H_2 addition).

In support of the FQT project, a review of activity transport in SCW was conducted. A large amount of data on release of fission gases and iodines in superheated steam is available from the US nuclear steam reheat programme, but no data could be found on transport of other fission products. To address this knowledge gap, leaching tests of SIMFUEL in SCW were performed to identify fission product species capable of dissolving from the fuel matrix and determine risk that the fuel itself might dissolve in SCW. Strontium and barium were the only species released in significant quantity; Zr, Ru, Rh, Pd, Ag, Sn, Th and U were detected in one or more tests at close to the Method Detection Limit, suggesting low solubilities under SCWR conditions.

CVR, Czech Republic, has continued to execute national project PRAMEK, focused on compatibility studies of common steels for power industry in water at supercritical parameters. Although the primary intent of this project was the selection of a suitable material for steam generator parts in supercritical fossil-fuelled power plants, the results are fully transferable to SCWR concept as well. In particular, this project concerns the ferritic-martensitic and martensitic steels complying DIN standards 7CrMoVTiB10-10, X20CrMoV11-1, X12CrMoVNbN9-1, X10CrMoVNbN9-2 and austenitic steels of DIN standards X10CrNiCuNb18-9-3 and highly alloyed X6CrNiNbN25-20. Multiple working media parameters were selected in order to choose the best water chemistry regime suitable for the above-mentioned steels. Working temperature and pressure were typical supercritical fossil-fuelled power plants parameters, i.e. 600°C and 25 MPa. The water chemistry was varied in order to obtain three different testing environments; the untreated demineralised water with neutral/slightly acid pH, the alkalised and deoxidised demineralised water with pH = 9 and lastly the alkalised demineralised water (pH = 9) with controlled content of oxygen.

Figure 3.30: **Electro-potential kinetic repassivation curve of as-received and exposed X10CrNiCuNb18-9-3 steel**

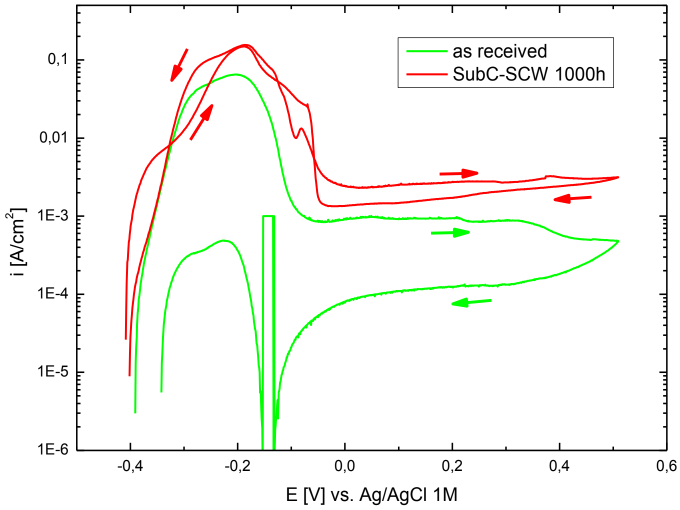
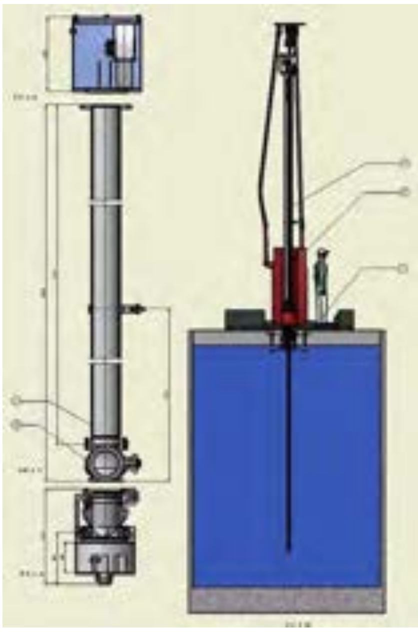


Figure 3.31: **Design of the fuel handling system for the fuel qualification test facility**



In brief, the results show that the corrosion processes occurring in supercritical water are a combination of electrochemical corrosion in water and direct high-temperature oxidation. Predominantly, the content of chromium in the steels determines their corrosion/oxidation resistance. Depending on the content of chromium a more or less protective oxide layer on the surface of the steel is formed, preventing its further reaction with oxidising environment. The corrosion resistance can be enhanced by further surface treatment which is the case of tested steel DIN X10CrNiCuNb18-9-3 which had been shot-peened prior to exposure to supercritical water. Such a surface treatment provides enhanced corrosion resistance given by formation of very compact and high-chromium content oxide layer. However, possible issues related to relatively high carbon content, have been identified when employing X10CrNiCuNb18-9-3 steel. During exposure at 600°C this steel undergoes strong sensitisation making it susceptible to intergranular corrosion. Figure 3.30 shows the electro-potential kinetic repassivation curve of this steel from which the level of sensitisation of 100% was calculated. Therefore such steel must be heat-treated very carefully in order to ensure the stabilisation of its structure for the application.

The PRAMEK project is close to being finalised, so the results obtained shall be evaluated and studied carefully in order to provide a clear recommendation concerning the selection of suitable material for particular working parameters and environment.

Fuel qualification test

The collaborative projects between Euratom (SCWR-FQT project, funded through the EC's 7th Framework Programme) and China (SCRIPT project, funded by Chinese Atomic Energy Agency, CAEA) in which the experimental facility for the qualification of fuel under HPLWR evaporator conditions will be finalised by the end of 2014. The majority of the objectives foreseen by the projects have been achieved; the results obtained in 2014 are as follows:

- The design of the facility has been completed; some details have been finalised or refined, e.g. the connection of the loop to the active ventilation and draining systems, design of the purification and measurement systems and additional independent emergency pumps have been added to the loop for redundancy. The fuel rod mock-up design has been optimised to fit into the autoclave at JRC-IET; results of the tests are summarised in the Materials and Chemistry section. The fuel handling system has been designed for dismantling the active channel in case of a failed experiment. The handling procedure has been set up, including the disconnecting of the coolant lines from the pressure tube by freezing the headpiece. The freezing procedure has been successfully tested, proving that the coolant lines may be safely disconnected and blinded and thus all the fission products present in the coolant will be retained inside the loop. Figure 3.31 shows the design of the fuel handling system.
- Construction of the experimental hall, where the loop will be housed, has been finished.
- Structural analyses of the fuel assembly have been performed showing that the rods will be slightly bended due to thermal stresses. Displacement of the rods will press them against the assembly box; however, the resulting stresses will cause plastic deformation of the assembly box and relax.
- Safety analyses for a postulated accident beyond the design basis have been performed to assess primarily the resulting radioactive source term. The postulated accident is a multiple accident that would result in failure of all three safety barriers and release of a part of the radioactive inventory into the loop. Results show that the radioactive source term would be significant, and although the accident is very improbable, redundant independently driven emergency pumps have been added to the loop. The remaining consequences of this accident, such as the hydrogen source term, steam explosion and risk of a condensation hammer have been assessed showing no significant risks for the integrity of the loop.
- Material tests of the candidate cladding materials have been finalised. Results showed that surface cold work is beneficial for the performance of the tested materials in supercritical water and therefore, several methods of surface treatment, such as shot peening, sand blasting and plane milling were tested and further optimised. The fuel rod mock-up has been tested in autoclave for general corrosion, performance during LOCA and a loss of internal pressure. The results are shown in the materials and chemistry section.
- The pre-qualification test was planned to be performed during 2014 in China with an electrically heated test section in a supercritical water facility. However, the test has been delayed and therefore, the previewed CFD and system code validations will not be performed during the lifetime of the projects. However, the data may be used for the previewed purposes when available. The fuel assembly will be tested at steady-state conditions as well as during a number of depressurisation transients.

- The results of safety analyses provided by the codes ATHLET and APROS have been compared; the results agree quite well for the accidents relevant for the fuel qualification test facility. Furthermore, system codes have been validated for the conditions similar to the fuel qualification test using data available in literature. It has been found, however, that APROS cannot properly model the transition from supercritical to subcritical conditions. An unsuccessful attempt has been made to implement an analytical model of the transient into the APROS code; the problem was identified to be caused by two aspects: improperly defined closure laws and invalid treatment of the phase change during depressurisation.
- The final outcome of the project will be presented at the 7th International Symposium on SCWR (ISSCWR7) from 15-18 March 2015 in Helsinki.

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3.4 Gas-cooled fast reactor (GFR)

R&D activities under the GFR System Arrangement did not make much progress in 2014, with no meetings of either the SSC or the PMB for the "Conceptual Design and Safety" project. This is in part due to the significantly decreased budgets supporting this system. However, two activities are reported here, one related to the support for the development of the GFR demonstrator ALLEGRO in the Visigrad group of countries (the Czech Republic, Hungary, Poland and Slovak Republic), the other related to cross-cutting fuel development in Japan.

3.4.1 The ALLEGRO gas-cooled fast reactor demonstrator project

The GFR concept was mainly based on studies performed in France in the late 1990s and was further developed within the EU 5th and 6th Framework Programmes respectively. It also included the development and safety assessment of a small experimental plant called at the time ETDR (Experimental Technology Demonstration Reactor). This plant was regarded as a necessary stepping-stone to a full-sized GFR in order to test the high-temperature fuel required by the latter. The concept was further analysed and refined by the EU FP7 GoFastR project: the ETDR has been renamed ALLEGRO and a number of design changes were introduced, e.g. the power was raised from the original 50 MWth to 75 MWth. ALLEGRO would function not only as a demonstration reactor hosting GFR technological experiments, but also as a test pad of using the high-temperature coolant of the reactor in a heat exchanger for generating process heat for industrial applications and a research facility which, thanks to the fast neutron spectrum, makes it attractive for fuel and material development and testing of some special devices or other research works.

The three respective nuclear research institutes of the Central European region (ÚJV, Řež, Czech Republic, MTA EK, Budapest, Hungary, and VÚJE, a.s., Trnava, Slovak Republic) agreed in 2010 to start a joint project aiming at the preparation of the basic documents in order to form the basis for a later decision on the construction and operation of the ALLEGRO gas-cooled fast reactor in one of the countries. CEA, France, supports this effort by various means, especially by transferring the documents of the earlier design efforts (under appropriate Non-Disclosure Agreements) to the project participants. NCBJ, Świerk, Poland, joined the project in 2012, i.e. ALLEGRO is supported in all the four Visegrad-4 (V4) countries.

The project ALLIANCE launched in 2012 with the aim to continue the elaboration of basic documents needed for high-level decisions and licencing of ALLEGRO. The main nuclear parameters (like power density, burnup etc.) would be similar to those of the planned 2 400 MWth power GFR. The core built up from the initial fuel type will be replaced by a core of ceramic fuel for the second half of ALLEGRO operation.

Safety analysis performed within the previous EU GoFastR project covered the consequences of most initiating events and most of the ALLEGRO relevant issues were analysed. Safety principles of the ALLEGRO reactor will be based on the WENRA requirements and the study of GIF, added to the actual national safety rules of the hosting country. Moreover, in formulating siting requirements and requirements concerning the design to reduce the impact of external hazards, the results of the European stress tests following the Fukushima events were applied. Nevertheless the current design of ALLEGRO does not fully satisfy these requirements. One of the main reasons is that the safety margin of the stainless steel cladded mixed oxide (MOX) fuel chosen for the initial ALLEGRO core of 75 MWth power is rather low and cannot provide the necessary protection against core melting after a Fukushima-type accident (though the margin is acceptably large concerning Design Basis Accidents, i.e. accidents which may occur with a very low but not negligible probability). The modification of the original design is therefore under discussion, and the original timescale of the preparatory phase is now extended until 2018.

The main components of the new strategy are as follows:

- Reduce ALLEGRO power from 75 MWth to cca. 10 MWth and find the optimum core configuration.

- Optimise nitrogen injection (launch time, duration) and the backup pressure in guard containment.
- Increase main blowers inertia to avoid short-term peak temperature for the LOCA+ blackout case and/or to develop a design with a gas turbine in the secondary side coupled to the primary blowers (this is the solution also advised for GFR).
- As a consequence of potential fuel supply difficulties it was also decided to use UO_2 pellets in AIM1 cladding instead of MOX pellets. It has no significant effect on safety but it may advantageously influence fuel management.

Figure 3.32: **The V4G4 centre of excellence for the GFR**

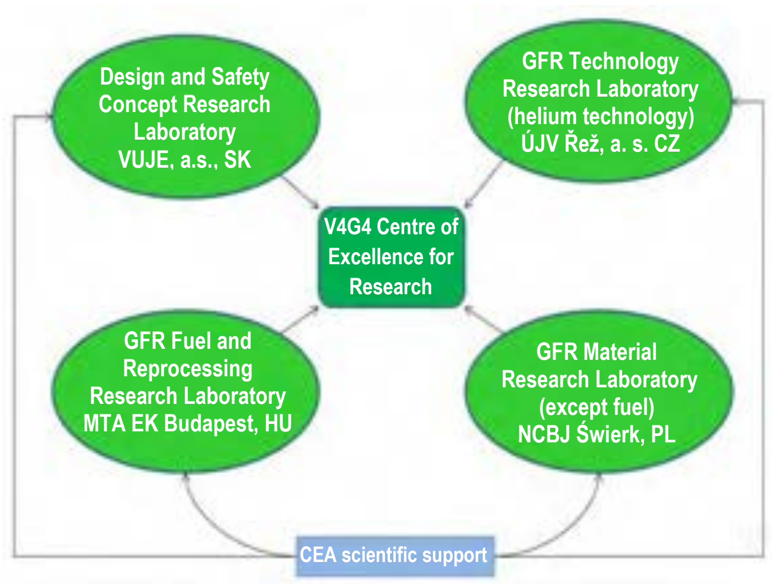
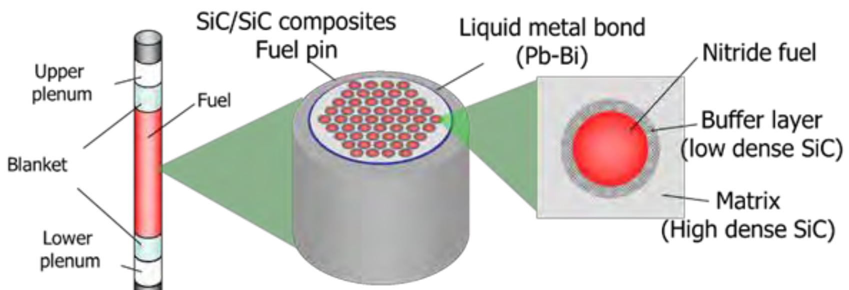


Figure 3.33: **GFR core design concept using fuel pin**



Fuel pin	SiC/SiC composites
Bond material	Liquid metal (Pb-Bi)
Fuel matrix	High dense SiC (> 95%TD)
Fuel buffer layer	Low dense SiC (~ 50%TD)
Fuel	Nitride fuel

The ALLEGRO consortium will be represented by a newly created legal entity, V4G4 Centre for Excellence, with the main goal to establish R&D facilities to investigate fuel development issues, helium technology-related problems, issues related to structural materials and to construct a non-nuclear 1:1 mock-up of ALLEGRO. According to the original intentions the main subject of the V4G4 co-operation is to generate experimental results on the new facilities for developing the gas-cooled fast reactor demonstrator ALLEGRO. If the financing can be ensured, experiments, system qualification and other preparatory works will provide by 2018 the governments of the V4 region a sound basis to decide on launching the design, licensing and construction of the ALLEGRO reactor, on selecting the site of the reactor and on financing and governance matters.

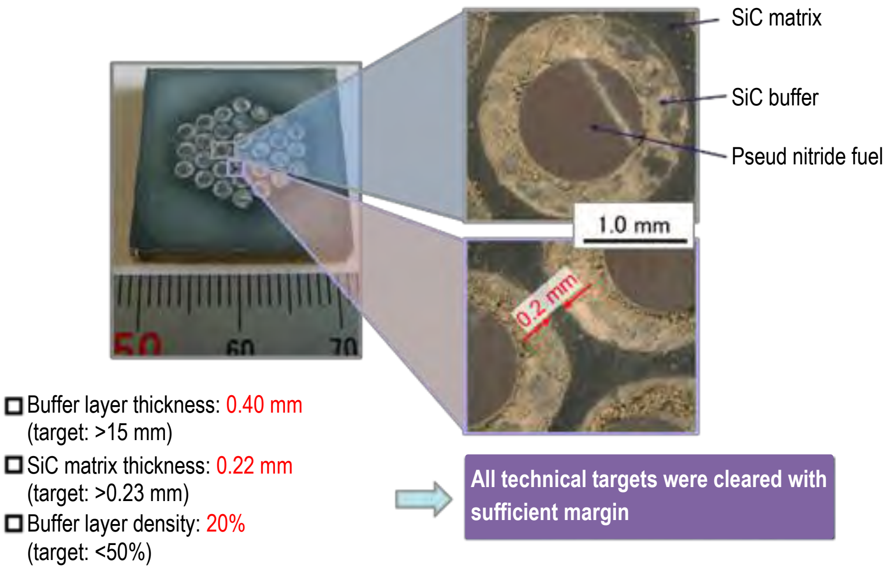
3.4.2 Development of high burn-up fuel with SiC matrix in Japan

Nitride fuel and oxide fuel are candidate for helium gas-cooled fast reactor to achieve thermal efficiency over 40%. The conceptual design of fuel, which consists of nitride fuel of around 50%, highly dense SiC matrix and porous SiC buffer layers is encouraged according to the report for "Feasibility Study on Commercialized Fast Reactor Cycle System (FS)" (JAEA/JAPC, 2006) as shown in Figure 3.33. However there is no technique to form dense SiC in the small gap between fuels with accurate arrangement of fuels. The objective of this work is to develop basic fabrication technique for the high burn-up fuel. The cylindrical fuel is proposed in this work instead of spherical fuel to achieve high dense fuel and accurate arrangement.

Wall thickness between fuels were estimated to 100~300 μm to achieve approximately 50% fuel density. It is almost impossible for machining without fracture. In this work, dense SiC matrix was formed with C bars, which were aligned for the positions of fuels, by hot pressing. The SiC nano-powder and sintering additives were used to form dense SiC matrix. The SiC matrix with C bars was annealed in air at 700°C. Holes for fuels and buffer layers were formed by decarburisation. Pseudo-nitride fuels were inserted into the holes with SiC nano-powder and polycarbosilane (PCS). The SiC matrix with pseudo-fuels were annealed in Ar at 1 500°C without pressure and porous SiC buffer layers were formed from SiC nano-powder and PCS as shown in Figure 3.34.

This work was the result of "Development of high burn-up fuel for gas-cooled fast reactor (GFR)" entrusted to "Kyoto University" by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

Figure 3.34: **SiC composite fuel**



3.4.3 Development of oxidation resistant novel silicon carbide composites in Japan

Silicon carbide (SiC) is very attractive engineering ceramics in particular for high temperature use and nuclear application due to high temperature strength, oxidation resistance, chemical stability, low activation, radiation resistance and so on. Silicon carbide composites are expected to be used as the core materials for GFR. Silicon carbide composites have pseudo ductile fracture behaviour by debonding and sliding at fibre/matrix interphase. However the carbon (C) as fibre/matrix interphase is the weakest link for severe environments including oxidation. The objective is to develop pseudo ductile SiC composites without C fibre/matrix interphase.

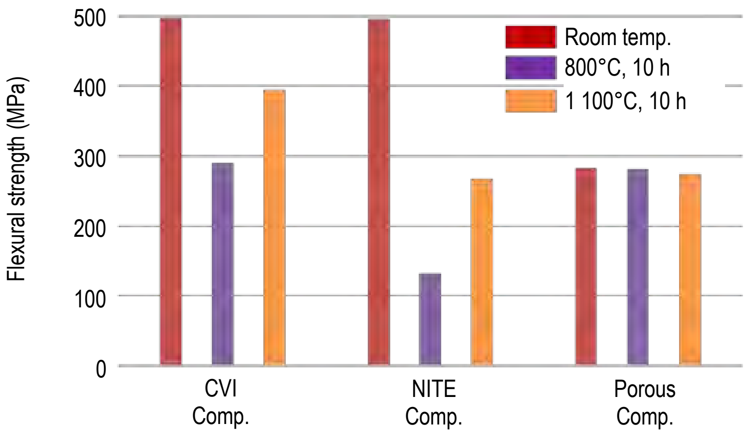
The SiC matrix was formed with carbon powder by liquid phase sintering method with sintering additives. The porous SiC matrix was formed following decarburisation process at 700C in air. The porosity in the matrix was controlled by the amount of carbon powder. The porous matrix SiC composites and conventional SiC composites with C fibre/matrix interphase were exposed in air and weight changes were characterised by TG analysis up to 1 100C. Mechanical properties were characterised by three point flexural test. Microstructure and fracture surfaces were observed by FE-SEM.

The porous matrix SiC composites consisted with just crystalline SiC fibre and crystalline porous SiC matrix without fibre/matrix interphase like C. Crystalline structure is requirement for nuclear application. The composites showed pseud-ductile and complicated fracture behaviour. Flexural strength was approximately 300 MPa for the composites with 30% porosity. Significant degradation was not observed following exposure at 1 100C in air, while conventional SiC composites with C fibre/matrix interphase had significant weight loss and strength degradation as shown in Figure 3.35.

Silicon carbide composites require relatively weak fibre/matrix interphase like C. The control of thickness and quality of the interphase is very difficult, although it is the key to determine mechanical properties of the composites. The porous SiC matrix composites showed pseud-ductile behaviour without the interphase. It is easy to fabricate a uniform material and reduce material cost significantly. The C interphase is the weakest link. The porous material just consists with SiC and applicable for various severe environments.

This work was performed under contract with Toshiba Corporation in “Research and Development of Innovative Technologies for Nuclear Reactor Core Material with Enhanced Safety” entrusted to Toshiba by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

Figure 3.35: **Effect of high temperature exposure in air on flexural strength of CVI, NITE and porous SiC composites**



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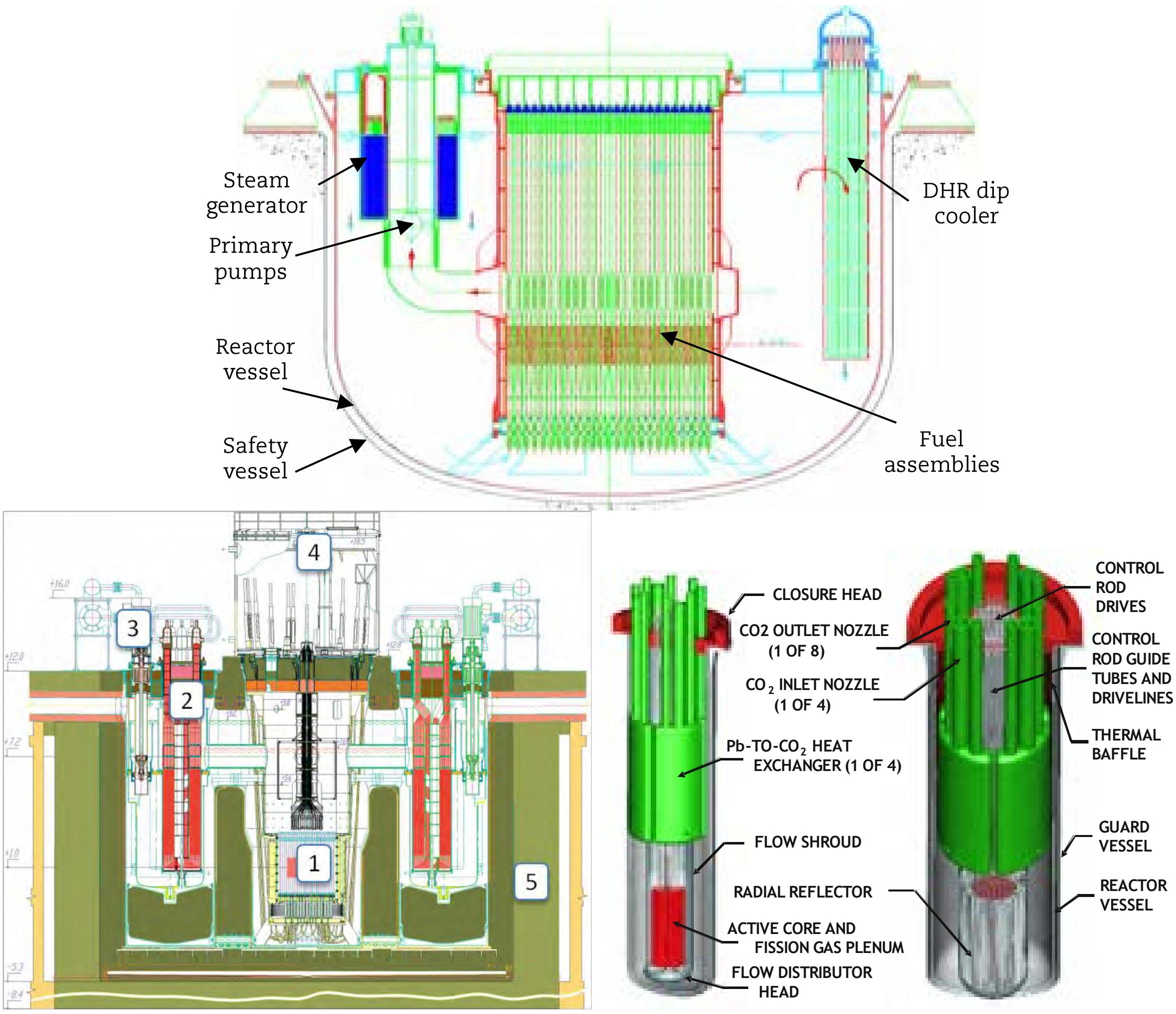
3.5 Lead-cooled fast reactor (LFR)

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 600 MWe (ELFR EU), intended for central station power generation. 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 co-ordination of the efforts carried out by participating countries has been one of the key points of LFR development.

Figure 3.36: **The three reference systems of GIF LFR – ELFR, BREST, SSTAR**



1. Core; 2. Steam generator; 3. Pump; 4. Refueling machine; 5. Reactor vault.

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

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, final draft from SSC has been prepared and sent to GIF Expert Group and is expected to be issued by the beginning of 2015.

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;
- 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 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 an 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 six 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. Actual members (MOU signatories) of the GIF-LFR-PSSC are: Japan, the Russian Federation and Euratom. The PSSC leverage also on the active participation of observers from: the United States, Korea and China.

The main activities of the PSSC during 2014 are related to the organisation of two PSSC meetings, the issue of the LFR White Paper on Safety in collaboration with GIF RSWG as well as the discussions on LFR Safety Design Criteria, presently under development. The 15th GIF-LFR-pSSC was held in Genova (Italy) hosted by Ansaldo on 5-6 May 2014. The main topic of discussion was the draft of the LFR white paper and strategy for SDC development. The 16th meeting of PSSC was held in Hefei (China) hosted by the FDS-Team, one of the observers of PSSC activities. The main topic of discussion was the preparation of the SDC draft and the request by the RSWG for the development of the terms of reference for GIF system safety assessment, presently under development. Activities on LFR System Research Plan led to the preparation of a draft report hopefully available for the Expert Group Review early in 2015. During 2014 a paper dedicated to GIF LFR activity was issued as part of a special issue of Progress in Nuclear Energy (Alemberti, 2014c). In May 2014 a Co-operation Agreement (CooA) was signed between the BREST and LEADER projects, respectively by Nikiet and Ansaldo. The purpose of this CooA is to improve the scientific tools, databases, design techniques, and the joint discussion of safety approaches and solutions, that is, to contribute to the progress of the Lead Fast Reactor technology implementation as a whole. Several meetings with participation of technical experts on both sides will be organised starting in 2015.

Main activities in the Russian Federation

In the frame of the development of the LFR, the Russian Federation plays a key role. The Generation IV LFR Technology Roadmap issued in 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 activities carried out in the Russian Federation in 2014 have been widely spread over many technological aspects of the LFR technology and are briefly summarised below:

Design: The pool-type BREST-OD-300 has an integral layout of the lead circuit components accommodated in the central and four peripheral concrete steel-lined premises.

The reactor core is composed of hexagonal shroudless FAs with fuel rods clad in ferrite-martensite steel. For the power distribution and coolant heat-ups to be leveled in the radial direction, the reactor core is designed as two radial zones, filled by the FAs, differing in the fuel element diameter only. Combination of the core breeding ratio ~1 with a small reactivity margin (less than βeff) ensures that the FA power and the coolant heat-ups are stable throughout the lifetime.

Safety: Combining the properties of a lead coolant and a uranium-plutonium nitride fuel with high dense and high thermal conductivity are the basis for fulfillment of the natural safety requirements with the exception of catastrophic consequences of an uncontrolled power growth after full implementation of the reactivity margin caused by possible equipment failures and personnel errors. A safety analysis has shown that transients exceeding the design reactivity margin do not lead to the reactor runaway with an uncontrolled power growth leading to severe accidents.

The deterministic safety analysis of the reactor transients shows that both normal and abnormal operational regimes accompanied by multiple failures of systems, including a simultaneous complete failure of the two reactor shutdown systems, do not lead to severe accidents. To V&V the accuracy of calculations activities are under way to verify experimentally the neutronic codes and the thermal-hydraulic characteristics of components on test beds, including BFS-IPPE and liquid metal coolants.

Thermal-hydraulics: The series of vibration, hydraulic and aerodynamic tests with the FA mockup and FA bundle have been performed in different regimes. The oscillation and resonance frequency values, mass- and heat-transfer data were obtained.

Coolant: Since the results of the earlier activities for the conceptual justification of the lead coolant technology proved it to be sufficiently mature, recent efforts have been focused on the development and testing of specific process systems and components.

The mockups of oxygen detectors, a gas handler (for various parameters), a hydrogen igniter, a mass-transfer apparatus, and lead and gas filtering materials have been tested.

Fuel: Pre-irradiation testing to investigate the best possible structure in terms of grain size, as well as other fuel parameters like: open and closed porosity ratio, measurements of heat conductivity, thermal expansion coefficient and mechanical characteristics in a working range of temperatures have been performed. Mathematical models of the fuel behaviour under irradiation were developed. In 2014, height FAs with experimental (U-Pu)N fuel elements were manufactured and installed into for irradiation in-pile tests in the BOR-60 (5 FAs) and BN-600 (3 FAs) reactors to obtain swelling, gas release, creep, mass transfer and fuel-cladding interaction parameters.

Materials: The series of vibration, hydraulic and aerodynamic tests with the FA mockup and FA bundle have been performed in different regimes. The oscillation and resonance frequency values, mass- and heat-transfer data were obtained. The initial 300-hour series of the erosion tests of plain-type fuel elements in grids has been conducted to study mechanical properties of the fuel cladding. The results indicated no damage occurring. Mockups of absorber elements have been irradiated, post-irradiation tests conducted, and representative parameters that define the serviceability of the absorbers employed to a burn-up of 12% (as 10B) have been determined.

Main activities in Japan

The development of LFR has not been included in the new basic energy plan issued in April 2014 in Japan. Thus, fundamental studies for the development of LFR were continued primarily in Tokyo Institute of Technology. The concepts of LFRs that have been proposed and studied in Japan are the portable small reactor, LSPR (50 MWe), JNC/JAPC middle-size LFR (750 MWe), and the innovative direct contact type reactor, PBWFR (150 MWe). In the PBWFR, feed water is directly injected into the primary coolant of hot LBE at the outlet of the reactor core. The advantages of this LFR concept are not only reduction of construction cost by means of simplified structure but also the avoidance of corrosion and erosion of the components.

However, it was necessary to overcome these issues:

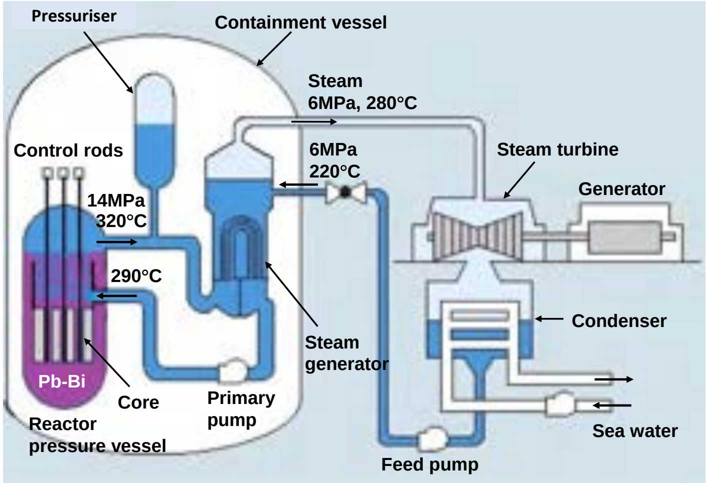
- contamination of a steam turbine system with 210Po (alpha-emitter);
- carry-over of LBE droplets from LBE surface and damage of steam turbine blades;
- mechanical problem of upper structure due to sudden boiling and vibration.

Thus, an innovative direct contact type LFR without boiling like PWR was proposed and studied. This LFR system is called Pressurised Water Lead-Bismuth-Cooled Fast Reactor (PLFR).

As for fundamental studies for the development of LFR, studies on corrosion of welded steel under tensile stress in static LBE, characteristics of metal diffusion in LBE in a capillary tube and the performance of oxygen sensor for LBE were performed. The results of the studies were presented at the 22th International Conference on Nuclear Engineering Advances in Nuclear Science and Engineering (ICONE22) held in Prague in July and at the 2014 Annual Meeting of the Atomic Energy Society of Japan (AESJ) held in Tokyo in March.

A special meeting was held for the developers of Gen IV reactors and officials of MEXT and MITI in Japan in August. The purpose of the meeting was to exchange the information of Gen IV reactor technologies among the developers and officials. The concepts of LFRs proposed in GIF and the status of the development of LFR technology were presented there.

Figure 3.37: **Concept of pressurised water lead-bismuth-cooled fast reactor (PLFR)**



Main activities in Euratom

Following the signature of the FALCON (Fostering ALfred CONstruction) Consortium Agreement on December 2013 by ANSALDO, ENEA (Italy) and ICN (Romania) the main activities of the newly formed consortium were dedicated to a critical review of the ALFRED design able to produce a final "conceptual" configuration of the LFR European Demonstrator as well as to organisation and funding aspects of the development of lead technology in Europe.

In December 2014, CV-REZ (Czech Republic) joined officially the FALCON consortium, adding its expertise in the development of innovative and sustainable energy infrastructures. The consortium successfully involved a number of additional European partners through the signature of a number of Memorandum of Agreements (MoA) expanding around Europe as much as possible the interest in the development of lead technology. In the frame of the MoAs all activities are performed in-kind by the parties.

About MYRRHA, during 2014 the FEED contract, awarded in October 2013 to a consortium formed by AREVA, ANSALDO, EMPRESARIOS AGRUPADOS and GROMTJ, was actively pursued. The activities of the consortium in the FEED contract are centred on the design of the balance of plant activities such as building, containment, hot-cells, auxiliary systems and supporting systems of the primary system while SCK CEN is carrying out the design of MYRRHA primary system. SCK•CEN also defined with the Belgian Safety Authority the detailed content of the pre-licensing phase and is drafting the different deliverables identified for the pre-licensing phase. Both the activities on primary side and balance of plant will be object of a strong effort during 2015.

Euratom was also funding during 2014 several projects under development like ARCADIA, MAXSIMA, SEARCH, MATTER, ESNII+, while more projects have been proposed at the last H2020 call of 17 September and are presently under evaluation.

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Many papers dedicated to LFR technology and reactor developments at the conference: Innovative designs and technologies of nuclear power: 3rd International Scientific and Technical Conference, 7-10 October 2014, Moscow (ISTC NIKIET-2014).

3.6 Molten salt reactor (MSR)

3.6.1 Main characteristics of the system

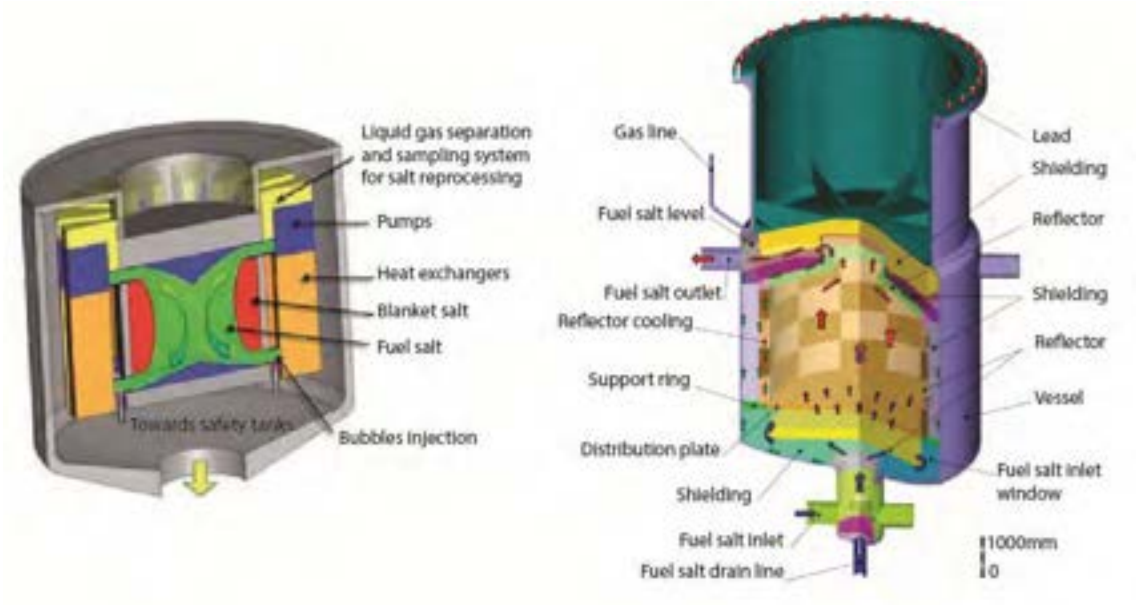
MSRs have two main subclasses. In the first subclass, fissile material is dissolved in the molten fluoride salt and it serves both as fuel and coolant in the primary circuit. In the second subclass, the molten fluoride salt serves as the coolant to a carbon moderated, ceramic fuel core similar to that employed in VHTRs. In order to distinguish the reactor types, the solid fuel variant is typically referred to as a FHR. Within the GIF, research is performed on the first subclass, the molten salt fast reactor concept, under an MOU signed by Euratom, France and the Russian Federation. The United States and China, observers in the PSSC of the MSR, are currently working on FHR concepts.

The molten salt fast reactor concept

From the outset MSRs were thermal-neutron-spectrum graphite-moderated design concepts. Since 2005, liquid-fuelled MSR R&D has focused on fast-spectrum MSR options combining the generic advantages of fast neutron reactors (extended resource utilisation, waste minimisation) with those related to molten salt fluorides as both fluid fuel and coolant (low pressure, high boiling temperature and, optical transparency). Recent MSR developments in Russia 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 3.38). The fast neutron spectrum molten salt reactors open promising possibilities to exploit the 232Th-233U 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 3.38: **MSFR and MOSART concepts**



Fluoride salt-cooled high-temperature reactor (FHR)

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, ceramic 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 United States for a first-generation, large power output (3 400 MW[th]), central station type FHR. 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.

3.6.2 R&D objectives

Partners of the MSR PSSC are involved in the Euratom-funded EVOL project (Evaluation and Viability of Liquid Fuel Fast Reactor Systems). 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 fibre ceramic composites.
- FHR specific fuel elements and assemblies.
- Tritium release prevention technologies.
- Salt Redox control technologies.

Figure 3.39: AHTR reactor building layout

3.6.3 Main activities and outcomes

Fuel composition options

The feasibility of using transuranic (TRU) elements from LWR used fuel as fissile material in an MSR with homogenious core has been evaluated by Rosatom in last years. Particularly, molten LiF-BeF_2 and LiF-NaF-KF salt mixtures are considered as candidate solvents as TRU burner/breeder respectively with and without U and Th support. The status and characteristics of the MOSART and MSFR designs operated in the Th-U cycle (see Table 3.2) were described in details in Gen IV Annual report 2013.

Other Li,Na,K/F MSR design after starting with fuel composition based on transuranic (TRU) elements from LWR used fuel can be fed by MA only. This option will have very high equilibrium loading of fissile nuclides (>30 t). Li,Na,K,U,TRU/F molten salt fast reactor (MSFR) started with TRUs from LWR used fuel can operate as a breeder in U-Pu fuel cycle (Degtyarev et al., 2014). At equilibrium, molar fractions of UF_4 and PuF_3 in the fuel salt are near 0.2 and 0.07, respectively, which is enough to provide a critical system with CR>1.

When substantiating any MSR concept, it is necessary to have detailed information on physical and chemical properties of candidate fuel salts and its compatibility with container materials and reprocessing scheme. Due to the solubility limit, the main concerns for MSR designs are to be expected with actinides and lanthanides dissolved together as trifluorides. For MSR system, the maximum temperature of fuel salt in the primary circuit made of special nickel-based alloy is mainly limited by tellurium intergranular corrosion under strain depending on salt Redox potential. For available Ni-based alloys (e.g. HN80MTY alloy), this temperature does not exceed 1 000-1 023K. Minimal temperature of the fuel salt in MSR primary circuit is also a key factor not only in terms of melting point, but the joint solubility for actinides trifluorides and tetrafluorides in the solvent for this particular temperature must also be taken into consideration. The experimental data on phase diagram for Li,Na,K,U,TRU/F system as well as solubility of TRU trifluorides in quaternary LiF-NaF-KF-UF_4 salt mixture are not available yet. In order to provide reasonable heat up for fuel salt in the core, fuel salt minimal temperature in the primary circuit and its melting temperature should not exceed 873K and 823K, respectively.

Table 3.2: **Main characteristics of the MOSART and MSFR designs**

Fuel circuit	MOSART (MARS)	MSFR (EVOL)
Fuel salt, mole%	LiF-BeF_2+1$TRUF_3$ LiF-BeF_2+5ThF_4+1UF_4	78.6LiF-12.9ThF_4—3.5UF_4-5$TRUF_3$ 77.5LiF-6.6ThF_4-12.3UF_4-3.6$TRUF_3$
Temperature, °C	620-720	650-750
Core radius/height, m	1.4/2.8	1.13/2.26
Core specific power, W/cm^3	130	270
Container material	Ni-Mo alloy HN80MTY	Ni-W alloy EM 721
Removal time for soluble FPs, yrs	1-3	1-3

Joint actinide and lanthanide fluoride solubility

In previous studies individual solubility for actinides and lanthanides fluorides in the melts was measured by isothermal saturation and reflectance spectroscopy. The technique developed provides reliable determination of equilibrium in the system melt-solid state and measurement with relative error less than 10%. 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). Particularly, it was found that two beryllium fluoride containing solutions LiF-BeF_2 and LiF-NaF-BeF_2 with BeF_2 concentration 27 mole% provide close values for individual solubility of PuF_3 in the 825-1 000K temperature range. The solubility of some other actinide fluorides, including AmF_3 in the molten LiF-BeF_2 salt mixtures was also measured. For two beryllium fluoride containing solutions (BeF_2 concentration from 27 to 34 mole%), the 241Am analysis showed that behaviour of americium is almost identical to that of plutonium.

Recently, the joint solubility of PuF_3 and CeF_3 in the 873-1 023K temperature range, was measured for 78LiF-7ThF_4-15UF_4 and 72.5LiF-7ThF_4-20.5UF_4 melts (see Table 3.3). In this case, logarithms of the molar concentration of CeF_3, PuF_3 as well as (CeF_3 + PuF_3) vs. 1/T (K) in the studied ternary melts LiF-UF_4-ThF_4 are not linear. Near the liquidus temperature for 78LiF-7ThF_4-15UF_4 and 72.5LiF-7ThF_4-20.5UF_4 salts, the CeF_3 significantly displace plutonium trifluoride at their joint dissolution. This suggests that the use of CeF_3 additives in the fuel LiF-ThF_4-UF_4-PuF_3 salt can provide effective removal for PuF_3.

The analysis shows that, unlike individual solubility of PuF_3 and UF_4 in 46.5LiF-11.5NaF-42KF eutectic, logarithms of the molar concentration of PuF_3 and UF_4 (joint solubility) in function of 1/T(K) cannot be described by linear function in the 823-1 023K temperature range. Results of measurement of joint solubility of PuF_3 and UF_4 in the 46.5LiF-11.5NaF-42KF eutectic also showed (see Table 3.4), in comparison with individual one, a significant decrease in the temperature range close to the melting point (823 to 873K). At 873K, individual solubilities of PuF_3 and UF_4 in

LiF-NaF-KF eutectic were found respectively 11.1 and 24.6 mol.%, and at their joint dissolution only 2.9 and 3.5 mol.%, respectively. This distinction decreases when temperature is increased.

Table 3.3: **Joint solubility of PuF_3 and CeF_3 in LiF-ThF_4-UF_4 fuel salts, mol.%**

Temperature, K	72.5LiF-7ThF4-20.5UF4		78LiF-7ThF4-15UF4	
	PuF_3	CeF_3	PuF_3	CeF_3
873	0.35±0.02	1.5±0.1	1.45±0.7	2.6±0.1
923	4.5±0.2	2.5±0.1	5.6±0.3	3.6±0.2
973	8.4±0.4	3.7±0.2	9.5±0.5	4.8±0.3
1 023	9.4±0.5	3.9±0.2	10.5±0.6	5.0±0.3

Table 3.4: **Individual and joint solubility of PuF_3 and UF_4 in LiF-NaF-KF eutectic, mol.%**

Temperature, K	Individual Solubility, mol.%		Joint Solubility, mol.%	
	PuF_3	UF_4	PuF_3	UF_4
823	6.1±0.6	15.3±0.8	1.16±0.06	1.75±0.09
873	11.1±1.1	24.6±1.2	2.9±0.1	3.5±0.2
923	21.3±2.1	34.8±1.7	13.2±0.6	11.0±0.6
973	32.8±3.3	44.7±2.2	19.1±1.0	17.3±0.9
1 023	-	-	21.0±1.1	19.0±1.0
1 073	-	-	22.5±1.2	20.0±1.1

Viscosity and liquidus temperature

As applied to MOSART and MSFR designs operating in Th-U fuel cycle the viscosity of the different molten salt mixtures has been measured at the temperature ranging from liquidus up to 1 160K by the method of torsional oscillations attenuation of the cylinder with the melt under study. The dependences of kinematic viscosity (ν, 10^{-6} m^2/s) vs. temperature (T, K) for molten salt mixtures are given in Table 3.4. In the temperature range where the melts behave like normal (single phase) liquids, the experimental viscosity values were approximated by the expression A * exp [B/T]. By least squares method the parameters of model were obtained. The kinematic viscosity root mean square (RMS) estimated in the assumption about dispersion homoscedasticity is (0.04÷0.20) 10-6 m^2/s. Effect of CeF_3 and BeF_2 addition on viscosity was also studied (see Table 3.5). In most cases, the presence of CeF_3 (from 1 to 10 mole%) or BeF_2 (from 2 to 5 mole%) in the eutectics mixtures decreased its viscosity at the cold leg of the measured temperature range.

Addition of 30 mol% UF_4 decreased the kinematic viscosity of the 46.5LiF-11.5NaF-42KF eutectic (mol%) within all temperature range of measurements (see Table 3.6). Increase of the CeF_3 concentration in the molten salt mixture from 5 up to 10 mol% or addition of the 10 mol.% CeF_3 to the 46.5LiF-11.5NaF-42KF eutectic (mol%) already contained 30 mol.% of UF_4 did not affect significantly on the melt viscosity. However significant increase in liquidus temperature was detected, i.e. up to 825K, to 950K and to 1 025K respectively by descending temperature experiments from 1 200K. This does not mean that the gross fuel salt could be solidified but some parts of actinide fluorides could become solid state. These too high melting temperatures make these molten salt mixtures practically not suitable for application in U-Pu MSR.

Table 3.5: **Kinematic viscosity vs. temperature (*T*, K) for different molten salt mixtures containing thorium and beryllium fluorides**

Composition, mole%	*T*, K	$\nu \cdot 10^6$, m²/s	RMS·10⁶, m²/s
78LiF–22ThF$_4$	898-1 119	1.980 exp{3689·(1/T-0.9698E-3)}	0.042
71LiF–27ThF$_4$–2BeF$_2$	866-1 073	2.075 exp{3093·(1/T-1.033E-3)}	0.10
75LiF–20ThF$_4$-5BeF$_2$ with 3 mole% CeF$_3$	851-1 093	2.1905 exp{1877·(1/T-1.013E-3)}	0.12
	966-1 115	2.037 exp{1465·(1/T-0.9627E-3)}	0.053
75LiF–20BeF$_2$–5ThF$_4$	924-1 158	1.996 exp{3159·(1/T-0.9593E -3)}	0.038
15LiF–58NaF–27 BeF$_2$ with 1 mole% CeF$_3$	723-1 063	3.267exp{3042·(1/T-1.086E -3)}	0.21
	723-1 070	2.6375exp{1870·(1/T-1.084E -3)}	0.116

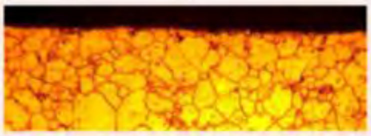
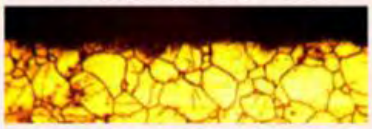
Table 3.6: **Kinematic viscosity (ν) vs. temperature (T) for 46.5LiF–11.5NaF–42KF melt (mol%) without and with UF$_4$ and CeF$_3$ additions**

CeF$_3$ mol%	UF$_4$ mol%	*T*, K	$\nu \cdot 10^6$, m²/s	RMS·10⁶, m²/s
0	0	727-1 144	2.479 exp{3658*(1/T(K)-1,116E -3)}	0.200
5	0	761-1 159	2.156 exp{1948*(1/T(K)-1,0536E -3)}	0.140
10	0	824-1 148	1.670 exp{2940*(1/T(K)- 0,9826E -3)}	0.070
0	30	948-1 173	0.9172 exp{3859*(1/T(K)-0.9501E-3)}	0.025
10	30	1 023-1 193	0.9180 exp{3989*(1/T(K)-0.8966E-3)}	0.031

Materials compatibility and salt chemistry control

Recent study with molten LiF-BeF$_2$ salt mixtures (mole%) fuelled by 2 mole% of UF$_4$ and containing additives of metallic Te or Cr$_3$Te$_4$, included 250 hrs tests with exposure of Ni-based alloys specimens at temperatures up to 800°C without mechanical loading. The Ni-based alloys selected for testing have the following compositions (in mass%): original US Hastelloy N (Mo-16.28, Cr–7.52, Fe – 3.97, Ti – 0.26, Si -0.5) and Russian HN80MTY alloy (Mo–13, Cr–6.8, Al–1.1, Ti-0.9). The corrosion facility allows to test the alloy specimens in the nonisothermal dynamic conditions with difference of the fuel salt temperature in the upper and near-bottom parts of test section about 40°C. Chemical analysis determined by ICP–AES in a typical frozen sample of melt before corrosion test showed the content of the major impurities (in mass%) as follows: Ni-0.005; Fe-0.024; Cu<0.001; Cr-0.001; O<0.05. In our tests the [U(IV)]/[U(III)] ratios in the fuel salt were changed in the range from 30 up to 90. As it can be seen in Figure 3.40, after Hastelloy N exposure without stress at 760°C in Li,Be,U/F with [U(IV)/]/[(III)] =60, a significant Te intergranular corrosion (IGC) can be observed. For the fuel salt with [U(IV)/]/[(III)] ratio = 90 at 800°C, the tellurium IGC for the HN80MTY alloy (the k parameter) is by about ten times lower as compared to original Hastelloy N. The studies have shown, that the Ni-based alloys IGC is controlled by the U(IV)]/[U(III)] ratio, and its dependence on this parameter is of threshold character. Providing control of the [U(IV)]/[U(III)] ratio, it is possible to drastically minimise the tellurium intergranular corrosion.

Figure 3.40: **Microstructure of surface layer for Hastelloy N and HN80MTY specimens without loading after 250 hrs exposure in Li,Be,U/F fuel salt for U(IV)/U(III) ratios: 30 and 60 (at 760°C) as well as 90 (at 800°C)**

U(IV)/(UIII)	Hastelloy N, enlargement × 160	HN80MTY, enlargement × 160
30 without loading at 760°C	No 	No 
60 without loading at 760°C	K = 3500pc × µm/cm ; l = 69µm 	No 
90 without loading at 800°C	K = 4490pc × µm/cm ; l = 148µm 	K = 530pc × µm/cm ; l = 26µm 

Thermodynamic data

Methods for synthesis of actinides fluorides for thermodynamic and electrochemical studies in molten salt media are being investigated in ITU (Institute for Transuranic elements, Karlsruhe, Germany) since 2013. Experimental equipment consists of HF gas line connected to a dedicated glove box equipped with a horizontal fluorination reactor and a high-temperature electrolyser (see Figure 3.41). The installation enables carrying out solid-gas reactions of actinide oxides, metals, chlorides etc. with pure HF gas up to temperatures of 1 200°C and electrochemistry of actinides in molten fluoride salts, using the advantage of purification the melt from moisture by gaseous HF, all under pure argon atmosphere.

During the verification experiments, pure NdF_3 and PrF_3 have been synthesised from the oxides. Successively, the work has focused on methods for synthesis of ThF_4 and UF_4 from ThO_2, UO_2 and $UO_2C_2O_4$. According to XRD analysis, both products have been achieved with purity higher than 99.5% (powder of pure synthesised UF_4 see in Figure 3.42). Electrochemical studies in eutectic LiF-CaF_2 mixture at a temperature of 850°C have shown possibility to prepare electrochemically pure carrier salt and studies on ThF_4 and UF_4 in this melt are ongoing.

To understand the molten salt fuel behaviour under normal and off-normal operating conditions and thus to assess the safety features of the primary circuit, a systematic knowledge of physico-chemical properties is needed. For that reason determination of physico-chemical properties of molten salt reactor fuel has continued culminating in the experimental validation of the full thermodynamic assessment of the LiF-ThF_4-UF_4-PuF_3 system which is the key system for the fuel of the molten salt fast reactor concept. The model has been based partially on novel experimental data obtained for ThF_4-containing salts and partially on similarities between the CeF_3 and PuF_3 components. The resulting lowest melting temperature of the quaternary system has been confirmed by experiments proving the reliability of the model.

Figure 3.41: **Experimental equipment installed in ITU (from left to right): an argon glove box, HF gas installation, electrolyser and fluorination reactor**

Figure 3.42: **Initial material (left) and final product (right) from UF_4 synthesis by HF fluorination of UO_2**

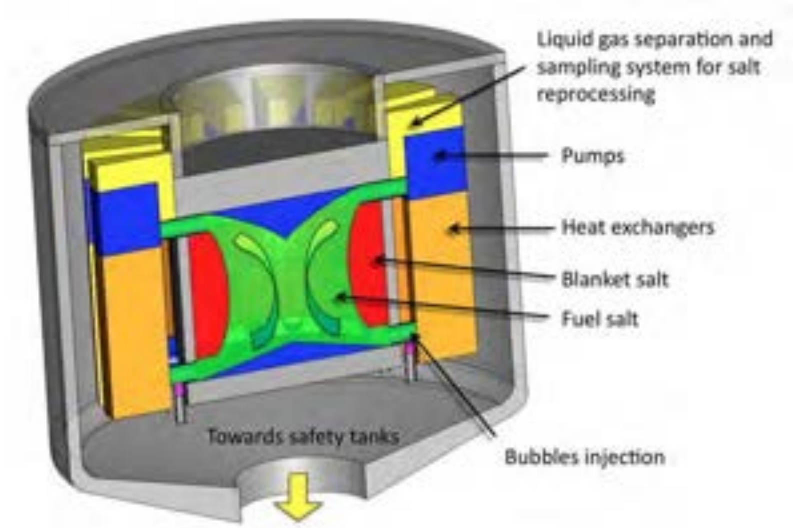
Not only the melting temperature is important, but also other properties like heat capacity or vapour pressure at elevated temperatures, the former for the assessment of the heat transfer within the primary circuit, while the latter to estimate the volatility of the fuel salt under accidental conditions. A systematic approach has been used to understand the general trend of the heat capacity behaviour in a multi-component molten salt system. After the series of LiF-AlkF (Alk = Na, K, Rb, Cs) binary alkali salts that have been analysed in recent years the LiF-CaF_2 system has been measured highlighting for the first time the heat capacity behaviour in the monovalent-divalent cationic fluoride system. A significant excess heat capacity has been found in the mid-range of the (Li,Ca)Fx liquid solution, confirmed by two calorimetric techniques, the drop calorimetry and the differential scanning calorimetry. The vapour pressure of the LiF-ThF_4 has been measured using the Knudsen effusion mass spectrometry yielding the activity coefficients of the LiF and ThF_4 end-members in this binary system which is a measure of their stability in the molten mixture.

Preliminary safety assessment of the MSFR

The reference MSFR is a 3 GWt reactor with a total fuel salt volume in primary circuit of 18 m3, operated at a max fuel salt temperature of 750°C. As shown in the sketch of Figure 3.43, the fuel salt flows from the bottom to the top of the core cavity (note the absence of solid matter in core). After exiting the core, the fuel salt is fed into 16 groups of pumps and heat exchangers located around the core and it travels through the circuit in about 3-4 seconds. The fuel salt considered in the simulations is a molten binary fluoride salt with 77.5% of lithium fluoride; the other 22.5% are a mix of heavy nuclei fluorides. This proportion, set throughout the reactor evolution, leads to a fast neutron spectrum in the core. The total fuel salt volume is distributed half in the core and half in the external part of the fuel circuit.

The fuel circuit is connected to a salt draining system which can be used for a planned shutdown or in case of incident/accident leading to an excessive increase of the core temperature. In such situations the fuel salt geometry can be passively reconfigured by gravity draining of the fuel salt into tanks located under the reactor and where a passive cooling and adequate reactivity margin can be obtained. This MSFR system thus combines the generic assets of fast neutron reactors (extended resource utilisation, waste minimisation) with those associated to a liquid-fuelled reactor.

Figure 3.43: **Schematic conceptual MSFR design, with the fluoride-based fuel salt in green and the fertile blanket salt in red**



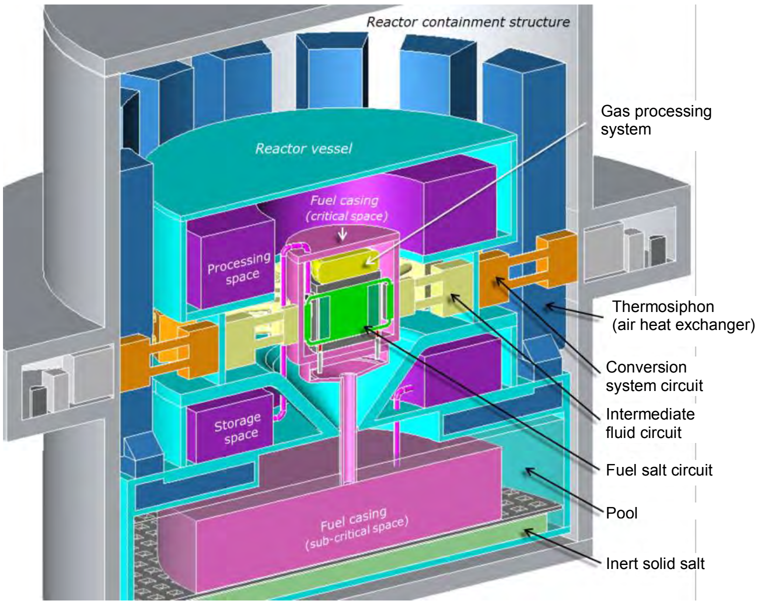
The design characteristics of the MSFR have been evaluated regarding safety issues. An example has been chosen here to illustrate this approach: one of the assets of the liquid-fueled MSFR systems is the homogeneity of the fuel. In a general way, this type of reactor can be placed in a category with all the reactors that run with a circulating fluid fuel (whether gaseous or liquid). These are referred to as homogeneous reactors. Since the 1960s, it has been shown that, in the case of homogeneous reactors without reactivity reserve, control rods are not necessary to control reactor operation. The MSFR, which is self-controlled thanks to its negative temperature feedback coefficients and the absence of in-core reactivity reserve fits in this category and, consequently, control or safety rods are not included in the design being considered. Contrary to a PWR, it does not require neutron flux shape control since the fuel is permanently homogenised and the coolant, here the fuel salt itself, can undergo large temperature increases (100°C to 200°C) with no risk of a boiling crisis susceptible to threaten the integrity of the cladding.

In the frame of the EVOL (Evaluation and Viability of Liquid fuel fast reactor systems) Euratom project of FP7 and through a PhD thesis of Grenoble Institute of Technology, the work-package “Design & Safety”of EVOL has thus addressed a first safety evaluation of the MSFR concept with a deliverable dedicated to the safety approach for a liquid-fuelled reactor, leading to a proposition of fundamental safety guidelines and the transposition of defence-in-depth principles. Coupled to a deliverable related to transient calculations and combined with the studies performed in the coupled MARS project of ROSATOM, this has provided preliminary conclusions on the safety performances of the concept and highlights issues requiring special attention in future design and safety assessment activities.

A transposition to the MSFR of the nuclear safety criteria and methodologies that were developed in the frame of solid-fuelled reactors has been proposed, in particular with the definition of the barriers. Like other safety notions, the transposition of the confinement barriers first mandates more general consideration of the origin and application of this concept. Eventually, these barriers will have to be redefined according to their usefulness for each reactor design rather than hunting for an equivalence with PWRs. The fuel is in a completely different configuration so that the confinement barriers cannot be identical. An extensive study adapted to the sequence of potential accidental events will have to determine or confirm the number of confinement barriers necessary in the case of the MSFR as well as their configuration. However, as a first step and as a pedagogical illustration describing the overall facility, the three fuel salt confinement barriers in the MSFR can be identified by analogy with PWRs as shown in Figure 3.44:

- pink: the fuel circuit (heat exchangers, pumps,...) and the draining system (the tanks and pipes) totally within the fuel casing;
- light blue: the reactor vessel, the intermediate circuit and the draining system's cooling pool;
- grey: the reactor containment structure (the building) and the emergency cooling chimney, not shown on the drawing.

Figure 3.44: **Layout of the MSFR reactor systems showing the three containment barriers (pink, light blue, grey)**



A possible transposition of the containment barriers for the fuel salt as well as for the gas processing system has been proposed. Other systems which also contain radioactive materials are to be studied, in particular the fertile blanket salt system including the storage and processing of the associated gases, as well as all the related inter-system transfers.

The conceptual differences between these two reactor types have led to develop a general methodology allowing a more thorough preliminary identification of possible MSFR accidents in the fuel circuit, as for:

- LOF (Loss of Flow): In the fuel circuit Loss of Flow accident, we gather all the accidents that are not associated to a slowing down or stalling of the intermediate fluid circulation and are not due to a loss of fuel.
- LOH (Loss of Heat sink): In a Loss of Heat sink accident, the fuel salt circulation is maintained but its cooling is no longer ensured.

- TLOP (Total Loss of Power): In the event of on-site Total Loss of Power all the pumps are stalled in the fuel, intermediate and conversion circuits; all active systems connected to the power supply are assumed non-operational; in this type of accident, the on-site security power supply is considered deficient as well.
- TOP (Transient Over-Power or OVC [OVer-Cooling]): An OVer-Cooling accident increases the reactivity and, as a consequence, the power generated because the reactor's thermal feedback coefficient is negative.
- LOLF (Loss Of Liquid Fuel): In the Loss of Liquid Fuel accident, a significant leak of the fuel salt outside the fuel circuit is considered.
- RAA (Reactivity Anomalies Accident): Since the reactivity reserve is very small in the MSFR, reactivity-related accidents have to do with reactivity anomalies rather than with accidents of the TOP type (control bar ejection). In fact, reactivity variations incurred in this reactor are much smaller than they are in a PWR.

Careful consideration of the QSR (Qualitative Safety features Review) questionnaire and the ISAM methodology has led to start identifying the weak points of the concept. Some safety-related items specific to the MSFR were singled out during the preliminary reflection around the QSR and were found beneficial. For example, in its present state, the design includes a large number of passive systems and procedures, something that is sought after for Generation IV reactors. As opposed to today's reactors, some of these items are purely safety oriented while others are also used during normal operation. There is a need for careful consideration of this last aspect conjointly with nuclear safety experts to asses if it is an asset or not.

The MSFR comprises other systems that will have to be analysed separately, such as the fertile blanket, the gas processing system, as well as other processing and radioactive material storage systems.

Finally, the method used to build the MSFR model for systemic risk analysis has been discussed and excerpts of the analysis were chosen and explained, bringing to light their importance. Even though the tools presently available do not allow the same level of analysis for all the accidents considered, this first qualitative and quantitative analysis has led to a better understanding of the MSFR safety characteristics at a very preliminary stage of the concept development, to allow design adaptations to minimise the identified risks and to reduce their consequences (notion of "inherent approach").

Molten salt technological studies

In the MSFR concept, a continuous cleaning gaseous process is proposed to extract non-soluble fission products (noble metals) dispersed in the flowing liquid fuel salt as well as the gaseous fission products. Helium bubbles with low volumic ratio (about 0.5%) are injected at the lower part of the core, and flow with the salt throughout the whole core towards a liquid/gas separator system. In order to begin studies on this bubbling process and components, an experimental project (FFFER (Forced Fluoride Flow for Experimental Research), CNRS/LPSC, France) was launched, based on the construction of a “laboratory” loop operated with about 80 litres of LiF-NaF-KF molten salt at 600°C.

Bubbling studies imply use of large pipes (55 mm diameter) and fast enough (about 1 m/s) flow (biphasic flow), and specific components: gas injectors and liquid/gas separator. Loop design and sizing follow directly from this fact. In addition all essential functional parts are present (forced circulation system, liquid level control, ultrasonic liquid velocity measurement) and have required technological developments. Studies dedicated to bubbling components and ultrasonic measurement have been previously carried out on water mock-up.

Test off the whole setup has been carried out in July 2014 and first run has begun in November 2014. Data on components behaviour and liquid/gas separation rate are at first collected using moderate flow.

Figure 3.45: **Molten salt loop dedicated to bubbling process studies (FFFER project)**

Figure 3.46: **UC Berkeley Compact Integral Effects Test Facility. The facility uses a simulant fluid (DowTherm A) to enable experimental evaluation of liquid salt thermal and hydraulic performance**

FHR-related activities

US MSR activities continue to be limited to the solid fuel MSR subclass (i.e. FHRs). The United States has both national laboratory and university led projects. The university projects are co-ordinated through the DOE Nuclear Energy University Program. The largest current university programme is a collaboration between the Massachusetts Institute of Technology (MIT), the University of California at Berkeley, and the University of Wisconsin. The project has included developing an advanced FHR reactor concept featuring a pebble bed core with on-line refuelling coupled to an open-air Brayton power cycle. The project also includes materials compatibility and irradiation testing as well as conceptual design of an FHR test reactor.

Other university projects include demonstration of a liquid salt direct reactor auxiliary cooling system (DRACS) at the Ohio State University and evaluation of FHR core configurations with higher fissile material loadings by the Georgia Institute of Technology. Two new university collaboration projects (one lead by MIT and the other by Georgia Tech) to resolve FHR technology issues were awarded during 2014.

Figure 3.47: **Early phase construction of Ohio State University's liquid salt pumped loop**

National laboratory led efforts during 2014 have included updating the small modular advanced high-temperature reactor (SmAHTR) design concept, beginning to evaluate the required technologies for an FHR coolant cleanup system, and continuing evaluation of the AHTR thermal and hydraulic performance. Also, Oak Ridge National Laboratory (ORNL) and the Shanghai Institute of Applied Physics (SINAP) have signed a bilateral agreement to co-operate on the development of FHRs. The agreement supports the broader memorandum of understanding signed by the DOE and the Chinese Academy of Sciences (CAS) on co-operation in Nuclear Energy Sciences and Technologies signed in December 2011.

Developing FHR industry consensus standards also is also continuing. Both ASTM standards on the material characteristics of continuous fibre ceramic composites (CFCCs) as well as development of ASME standards on the use of CFCCs for core support structures continue. In addition, an ANS standard on the design safety of FHRs is under development.

Additional information on FHR technologies is available on ORNL FHR web pages. www.ornl.gov/science-discovery/nuclear-science/research-areas/reactor-technology/advanced-reactor-concepts/fluoride-salt-cooled-high-temperature-reactors.

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Chapter 4. Methodology working group reports

The three GIF MWGs – 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 to evaluate 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.

The methodology developed by the EMWG is based upon the economic goals of Generation IV nuclear energy systems, and consists of the following:

- Cost Estimating Guidelines for Generation IV Nuclear Energy Systems, Rev. 4.2 (GIF/EMWG/2007/004), www.gen-4.org/gif/jcms/c_40408/cost-estimating-guide lines-for-generation-iv-nuclear-energy-systems;
- G4ECONS (Generation IV Excel Cost Calculation of Nuclear Systems) software package;
- Users Manual for G4ECONS Version 2.0 (GIF/EMWG/2007/005).

The cost estimating guidelines provide a uniform set of assumptions, a uniform Code of Accounts (COA) in developing cost estimates for advanced nuclear energy systems. It discusses the development of all relevant life cycle costs for Generation IV systems, including the planning, research, development, demonstration (including prototype), deployment, and commercial stages. These guidelines form the basis for the software model G4ECONS, a Microsoft Excel-based tool used to calculate the levelised unit cost of energy products including heat and electricity. The software also includes an economics model for the application of energy products for co-generation applications. It provides the ability to analyse applications other than electricity production, such as hydrogen production or desalination, or a combined production of electrical and non-electrical energy production. An additional module was developed to calculate the cost of fuel cycle services. The combination of the software and guidelines facilitate the development of consistent, comprehensible and comparable cost estimates to be performed by the system development teams.

Previous years

In September 2007, the EMWG released the methodology for public as well as GIF application. A CD is available from NEA containing the complete methodology. To date, over 190 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 Cost Estimating Guidelines and the G4ECONS software have been demonstrated through sample calculations for both Generation III and Generation IV systems. Several papers demonstrating the implementation of the cost estimating methodology were presented by EMWG members at the GLOBAL 2009 conference and GIF Symposia held in Paris, France, in 2009 and in San Diego, California, United States in 2012. Several members of the EMWG presented papers in various international meetings and published articles in scientific journals.

In 2009, the EMWG also developed a standard training presentation for the application of the methodology. The training presentations are modularised so as to be useful for presentation from a management level to a detailed user's level. EMWG members are prepared to give this presentation to GIF groups as requested. Level-1 training presentations have been given to several GIF groups. See, for example, http://web.ornl.gov/~webworks/cppr/y2001/pres/125294.pdf 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.

The G4ECONS software has been extended to fuel cycle applications. Several studies were done to demonstrate an approach for estimating the cost of sectors of nuclear fuel cycles.

2014 activities

The EMWG had its 30th meeting on 21 May 2014 in Paris, hosted by the NEA, and its 31st meeting on 28-29 August 2014 in Vancouver, Canada, hosted by the Atomic Energy of Canada Limited.

The focus of these meetings was on two main initiatives of the EMWG, namely, the collaboration with IAEA INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) and the development of the next version of G4ECONS. During the GIF-INPRO interface meeting held in Vienna on 4-5 March 2014, it was decided to continue further collaborations to examine the two methodologies; G4ECONS and INPRO's Nuclear Economics Support Tool (NEST) for potential harmonisation. The two methodologies were compared using common sets of input data for a Gen III+ LWR system and a Gen IV SCWR system, both with open fuel cycles. Sensitivity analyses were performed using both G4ECONS and NEST over a range of uncertainties in the capital, fuel and operating costs as well as with respect to discount rate, construction period and operational life of the plant. Both G4ECONS and NEST yielded comparable LUEC and total investment costs. The benchmarking exercise was useful to understand the differences in the two methodologies for potential harmonisation opportunities in the future. Additional benchmarking activities are planned for more advanced nuclear systems using closed or partially closed fuel cycles.

A version of G4ECONS that can incorporate uncertainty is still being updated. EMWG discussed the updates and improvements for the next version of G4ECONS during its 31st meeting in Vancouver. The primary focus is to streamline the user interface to simplify the data entry sheets and to clarify the uncertainty analysis outputs.

EMWG members from Canada presented a paper on the uncertainties in the economics analysis of Generation IV system at the Pacific Basin Nuclear Conference in Vancouver, 24-28 August 2014. The paper (www.gen-4.org/gif/upload/docs/application/pdf/2014-09/pbnc_paper-aecl.pdf) discussed the results of the economics assessment of an SCWR concept using G4ECONS and highlighted the impact of the uncertainties in the capital and operating costs of the Generation IV systems at concept stage on the levelised unit electricity cost.

The EMWG maintains contacts with SSCs through participation of the representatives in the Experts Group and Policy Group meetings. EMWG held a joint meeting with the SCWR SSC in Vancouver on 29 August 2014. The EMWG members made presentation on the use of G4ECONS tool and the results of economic analysis of an SCWR concept using G4ECONS.

EMWG co-chair, Harrison, participated in a meeting organised by the China GIF Liaison Office and the China Institute of Atomic Energy (CIAE), entitled "Seminar on the Economic Modelling and Safety Assessment Methodologies for the Gen-IV Nuclear Energy Systems", held in Beijing, China, on 5-7 November. He presented the EMWG methodology and demonstrated the use of the G4ECONS tool to a group of representatives from the China nuclear industry, including Tsinghua University, CIAE, the China Nuclear Energy Association (CNEA), and the Shanghai Institute of Applied Physics (SINAP). The audience showed a high level of interest in the topic of economic modelling and familiarity with the G4ECONS tool. The EMWG continues to monitor the use of the methodology and encourages feedback on its use and possible improvement and maintains a watch on the international activities and studies on the nuclear economic and cost matters.

4.2 Proliferation resistance and physical protection assessment methodology

The PRPPWG has developed a methodology for evaluating the performance of advanced nuclear systems against the proliferation resistance and physical protection goals of GIF. The methodology is described and documented in a publicly available document posted on the GIF open website since 2011 (Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, Rev. 6, GIF/PRPPWG/2011/003). It offers a generic, mature and comprehensive framework for evaluating through measures and metrics the PR and PP characteristics of advanced nuclear systems. As such, it may help the GIF SSCs and other designers of Gen IV systems in improving PR and PP robustness of their concepts at an early stage of design development.

In 2014, based upon this robust foundation, the group focused its activities on outreach aiming at disseminating information on the methodology to a wide range of potential users within and outside GIF, broadening the number of its users and eventually reflecting feedback from those users in an enhanced version of the approach.

A set of frequently asked questions (FAQ) to the group which was posted on GIF public website early in 2014 provides interested experts and policy makers with insights on key features of the methodology and its results when applied to an advanced nuclear system. The document is intended as a first step in familiarising SSCs and PMBs with the PR&PP Methodology in order to promote its use within GIF and eventually to a broader audience.

Pursuing the goal of publicising the methodology, the group prepared a collective paper which was presented at the "Symposium on International Safeguards: Linking Strategy, Implementation and People", held at the IAEA in Vienna, Austria, on 20-24 October 2014. The paper outlines the main elements of the PR&PP methodology and provides an overview of its applications to various advanced nuclear systems. It summarises progress made by the group in enhancing the approach and facilitating its use, and outlines the path forward for increasing the accessibility of the methodology to a broader audience and making its results more relevant for designers and policy makers (IAEA CN-220 289).

In order to support dissemination of information on the methodology, a comprehensive bibliographic list of papers presented in international events and articles published in scientific journals was compiled and finalised by mid-2014. The bibliography which lists only publicly available documents is aimed at a broad audience and has been cleared by the GIF Policy Group for posting on the public GIF website and distribution at various international meetings.

The group is investigating ways and means to reflect lessons learnt from studies carried out using the methodology within and outside GIF. Results obtained illustrate how early assessment of PR and PP characteristics of nuclear systems can identify areas where further R&D is needed and help designers in achieving good performance in an efficient manner. The group recognises that the methodology is not a simple analytic approach and requires the appropriate subject-matter experts for aspects of the analysis. However, it provides high-level guidance to potential users (e.g. the FAQ) to make the advantages of the methodology understandable. The group has found that even a simplified, qualitative application of the methodology can provide valuable insights during conceptual design. To address concerns about the potential complexity of PR&PP assessments, the group has developed a training workshop and supporting materials that includes a set of simplified guidance and illustrative examples of results for the benefit of analysts as well as decision makers.

In connection with the 25th meeting of the group, held at the NEA in Issy-les-Moulineaux, France, on 10-11 December 2014, a training workshop was organised for potential users of the methodology including industry, government, and academia. Feedback from the workshop will provide additional insight on required evolution of the methodology to make it more user-friendly.

The concept of 3 S (Safety, Safeguards, and Security) is being considered as more and more relevant by various experts and policy makers. In this context the PRPPWG is developing a white paper on the impact that the integration of the three issues within a unified framework may have on the methodology.

Robust safeguards are essential for the development and implementation of Gen IV nuclear systems. The group is maintaining cognisance of technology developments and good practices that would foster safeguards-by-design of the GIF systems, looking for opportunities to integrate the concept within its methodology.

The PRPPWG was represented in GIF Experts and Policy Group meetings held in 2014 where progress of work and main activities were reported to the GIF governance. Those meetings offered opportunities to pursue the dialogue with members of the SSCs, exchange information and continue to investigate the possibility of launching a case study on one or more GIF nuclear systems as a means to test the relevance of the methodology for helping R&D teams and designers.

The PRPPWG maintains a close co-ordination with the other GIF Methodology Working Groups (MWGs) as instructed by the GIF governance to build a coherent framework for evaluating the GIF nuclear systems against the goals set up in the Roadmap. In this connection, the group took the lead in developing an abstract for the GIF Symposium to be held in May 2015 which is embedded in ICONE 23 Conference (Makuhari Messe, Chiba, Japan, 19-20 May 2015). The abstract summarises the achievements, status and trends of the three MWGs.

Especially important is the ongoing co-operation with the Risk and Safety Working (RSWG) to ensure synergy and complementarity between the methodologies developed by the two groups. At each meeting of the RSWG, the PRPPWG is either represented or send a report on recent activities relevant from the safety viewpoint. It is planned to organise joint sessions of the two groups whenever possible, as it was done in 2012 in Obninsk, Russia.

In the light of the essential role of the IAEA in the area of safeguards and proliferation resistance, the close collaboration with the IAEA has been maintained since the inception of the PRPPWG. One or more representatives of the IAEA have been observers in the group, facilitating exchange of information and cross-fertilisation whenever relevant. Members of the group are participating in IAEA activities and bring in feedback on potential impact of the evolution of safeguards concepts and approaches on the methodology.

The 2014 annual interface meeting between GIF and INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles), provided a forum for exchange of information and for identifying future collaborative efforts. Taking into account the outcomes from the INPRO/PROSA (Proliferation Resistance and Safeguardability Assessment tools) project, in which several members of the group were involved, it was determined that the INPRO and GIF/PRPPWG approaches to PR evaluation have different goals and scope and will continue to be complementary but distinct, each one benefitting from the other and producing results relevant for its target audience.

The group continued to monitor national and international activities which may have an impact on its future work and could require adaptation of the methodology. In particular, the group will give consideration to developing simplified guidance, training materials and illustrative examples which could make the methodology more usable by target audiences in GIF.

4.3 Risk and safety assessment methodology

The Risk and Safety Working Group (RSWG) is among the methodology groups established in GIF with the mandate to perform cross-cutting activities in collaboration with and support of all six system steering committees with the goal of providing an effective and harmonised approach to the safety assessment of Generation IV nuclear systems. As part of its objectives, the RSWG has developed an Integrated Safety Assessment Methodology (ISAM) intended for use throughout the concept development and design phases to influence the course of the design evolution. The methodology consists of a set of analysis tools that apply selectively throughout the design process and are expected to yield an objective understanding of effectiveness of safety-related design provisions and other safety-related issues that are important to decision makers.

In view of the application of the methodology and in response to the comments of stakeholders, in 2014 the RSWG has finalised and published a Guidance Document for application of the ISAM (GDI). The guidance document provides a more detailed description and justification about the integration of the different ISAM tools and practical guidelines for its application. In addition the document focuses on the application of ISAM tools on specific cases that can provide the potential user with insights on the usefulness of the methodology for safety assessment. The guidance document can serve as a tutorial for using ISAM tools assuming that the reader has already a general knowledge of ISAM and has access to the main report.

Also in 2014, the RSWG worked with the GIF Task Force (TF) on development of Safety Design Criteria (SDC) for SFR systems. With several RSWG representatives also members in the task force, the group provided feedback on the comments received on the SFR SDC Phase 1 report, after having been sent to external organisations for review. The RSWG is also contributing to the 2nd phase activity for development of the safety design guidelines (SDG) in close collaboration with the SDC TF. The group will review the SDG reports as they are released and provide recommendations on the safety approach and safety assessment for the Gen IV reactor system. As more feedback from international organisations are expected in the coming years, the RSWG will continue to support the TF in the interaction between the GIF community and the international/national organisations.

Some progress was made in preparation of the risk and safety white papers for the Generation IV systems with completion and approval of the LFR document. The white papers developed jointly between RSWG and each system steering committee (SSC) are being prepared to also achieve a better co-operation between RSWG and the individual SSCs so that a common vision about the safety of Gen IV system can be brought forward. It is recognised that the designs of the six systems have different levels of maturity; hence, the content and the level of detail for the white papers may be significantly different. The RSWG is strongly committed to help the individual SSCs in finalising this task.

As part of the interpretation of the lessons learnt from TEPCO's Fukushima Daiichi nuclear power plants and their application for Generation IV safety work, in 2014, the RSWG was asked by the GIF Policy and Experts Group to review the GIF Safety Goal 3 that states the elimination of need for offsite emergency response. The PG and EG have expressed their desire to maintain the current language for the safety goal and have requested RSWG to address this safety goal through design measures. While offsite emergency response will likely remain a regulatory requirement for Generation IV systems, the RSWG considers that the architecture should manage and mitigate consequences of severe conditions through combination of active and passive safety features, prevention of cliff edge effects, enhanced containment function with built-in safety provisions to mitigate severe conditions, and provision against hazards with due consideration of potential for common cause failures.

In 2014, the EG has endorsed a new task related to the safety assessment of the six GIF reactor concepts and asked the RSWG to co-ordinate it. The objective of this task is to review and identify the main safety advantages and challenges of the six systems with the aim of providing a snapshot of the major safety concerns of the technologies. This review will assess the current status of safety-related R&D and allow selecting direction of future R&D needs for each system. To help with this task, the RSWG will closely work with the individual GIF SSCs in the preparation of a safety assessment document for their respective system. The RSWG has set up a table of contents and a tentative plan to finalise the report by early 2016. The RSWG will serve as a co-ordinator among the six systems by reviewing the SSC proposals and verifying the information provided regarding the safety concerns of the technologies. The aim is not to promote present status of performance, but to clearly identify on the one hand the strengths of the systems relying on existing technology and equipment, and on the other hand not-yet-mature technologies that need further development in order to satisfy the GIF safety goals.

In 2014, the RSWG has experienced important changes in membership composition as well as the replacement of one of its co-chairs. This has created an opportunity for a fresh look at the group's activities moving forwards. However, the decade-long promotion of a consistent approach to safety, risk and regulatory issues among Generation IV systems still remains the RSWG's primary mission. New directions and fresh thinking on how to accomplish this objective are necessary. The advisory role of RSWG to the PG and EG on interactions with the nuclear safety regulatory community, international organisations and relevant stakeholders remains crucial. The RSWG continues to maintain and reinforce its interfaces internally with the individual SSCs and the PRPP methodology working group and externally with the regulators, IAEA, INPRO, and MDEP 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 2014, the SFR task force (TF) received the international reviews on the Safety Design Criteria (SDC) Phase-I Report and also proceeded to develop the Safety Design Guidelines (SDG).

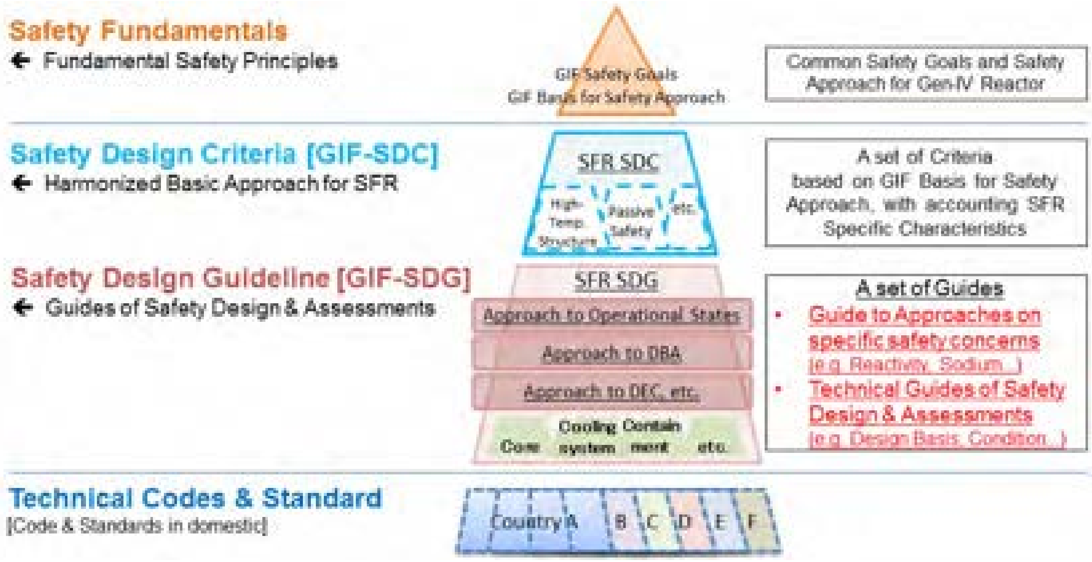
The SDC Phase 1 Report, which had been prepared by the TF, was approved by the GIF policy group in May 2013. The SDC Phase 1 Report was then circulated to international organisations (i.e. IAEA, MDEP, OECD/NEA/CNRA) and regulatory bodies of the SFR developing states under the GIF (i.e. China, EC, France, Japan, Korea, Russia, United States) in order to enhance interactions from the regulatory bodies as an external review process for the feedbacks. The review results on the SDC Phase-I Report, provided by the end of 2014, were from the IAEA, the USNRC, and the China NNSA. The review results vary from general comments (e.g. safety approach as the Gen IV reactor systems and relationship between safety and security) to detailed specific recommendations for individual criteria related to technical characteristics of SFRs (e.g. sodium-fire and its consequences, parameters important to transient, DBA and severe accident). The TF held its meetings in February, June and October 2014, and conducted a thorough analysis of the review results. The TF response to these reviews and recommendations are summarised as feedback to the international reviewers. Additional interaction with international organisations, e.g. OECD/NEA/CNRA is foreseen in 2015 for further input for continued improvement of SFR SDC based on regulatory insights.

Whereas the publication of the SDC Phase 1 Report in May 2013 is the remarkable achievement related to the safety design of the Generation IV SFR reactor systems, an important incentive and motivation for further technical interpretation and clarification of the SDC were raised. Based on these incentives/motivations, the Phase 2 activity of the SDC TF for the development of safety design guidelines (SDG) was started in September 2013 and important progress was made throughout 2014. The SDG is conceived as a detailed guideline documents one level lower of the SDC in a hierarchy of the safety standards as show in the Figure 5.1. It is intended to support practical application of the SDC in the Generation IV SFR design, and it will include “quantification of key aspects” and “clarification on technical issues for common understandings.” There are two expected outputs as the SDG. The first output is a report on “Guidelines on Safety Approach and Design Conditions of Generation IV SFR systems” (so-called “Safety Approach SDG”) which is for guiding safety approaches according to the SDC. It is based on the general safety approach and technical issues listed in the SDC Phase 1 Report, and the primary contents are set on “prevention and mitigation of severe accidents (issue related to fast reactor core reactivity)” and “accident conditions to be practically eliminated (issues related to loss of heat removal)”. As the important interim output in 2014, based on the common understandings on special phenomenological and consequential features, the SDC TF summarised “sets of design guidelines for two issues” and “provisions with designs options” in a worksheet, and prepared the first draft for the “Safety Approach SDG.” The draft report will be updated with the GIF internal review process and will be finalised in 2015.

In order to discuss SDC/SDG with the stakeholders, the fourth joint GIF-IAEA Workshop on “Safety of Sodium-Cooled Fast Reactors” was held on 10-11 June 2014 at the IAEA headquarters. The main purpose of the Workshop was to present and discuss: i) the status of the SDC Phase 1 Report review by national Regulators and International Organisations; ii) the implementation of current SDC by the designers of innovative SFRs concepts; iii) the status of SDC Phase 2 development; and iv) specific safety design criteria e.g. practical elimination of accident situations, design extension conditions, sodium void reactivity effect.

The next SDC TF meeting is planned in May 2015 and several additional meetings are foreseen to complete the SDC TF 2nd Phase activity by 2016.

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



5.2 Sustainability

Background

From the outset of GIF, challenging technology goals for Gen IV nuclear energy systems were set in the following areas: sustainability, economics, safety and reliability, and proliferation resistance and physical protection. These were subdivided into eight sub-goals which served as the basis for the development of criteria/metrics to assess and compare candidate Gen IV systems. The criteria were developed to formulate a number of factors that indicate performance relative to the goals, and it can be interpreted as a more specialised goal. Metrics were devised to evaluate concept performance against these criteria using specific measures.

In the area of sustainability, there are two specific goals: resource utilisation, and waste minimisation and management. The suggested metrics in the Gen IV roadmap are the use of fuel resources for resource utilisation, and waste mass/volume/heat load/radiotoxicity and environmental impact (annual dose, acute dose for the current and future generations). The fuel cycle crosscut group identified that the methodology for sustainability assessment already existed in the preparation of environmental impact assessments, and in principle, no further development of evaluation methods was deemed necessary. In accordance with this finding, no “sustainability methodology working group” was launched, whereas in other areas, the EMWG, RSWG, and PRPPWG were established.

As a result, in the context of GIF, no established sustainability assessment results for Gen IV systems are yet available. There is no reference methodology for sustainability assessment, and no ongoing effort to assess the sustainability of the selected Gen IV systems. Currently, the assessment of sustainability of those systems is left up to the concept developers for each reactor concept. This implies that, for instance, in terms of SFR concepts, ESFR, KALIMER, and JSFR sustainability assessments will be based on different methodologies, with various assumptions and uncertainties. To derive a consistent comparison it is essential to have a common agreed set of reference parameter values for reference plants/facilities involved in the sustainability assessment. It is also obvious that without a sustainability analysis for Gen IV systems, the sustainability of nuclear energy systems cannot be addressed in detail.

Recognising this situation, the idea of setting-up a "sustainability evaluation task force" was discussed at the EG meeting in May 2014. Results from this discussion were reported on at the PG meeting in May 2014. The PG decided to create an interim task force (TF) on sustainability and suggested that discussions on how to proceed should take place at the PG meeting in December 2014. According to PG recommendation, the GIF Technical Director sent out a nomination request to the PG members in September 2014. By November 2014, five nominations were received for the TF, and draft terms of reference were then distributed for review. Once the comments from the TF members were reviewed the TD decided to seek high-level guidance on TF actions, especially in the area of definition and work scope.

Draft action plan (~May 2015)

The suggested work scope of the TF is as follows:

- review of existing documents related to sustainability methodologies and assessments including INPRO reports;
- establishment of definition for sustainability;
- establishment of definition for systems/facilities;
- development of plan for GIF sustainability methodology development;
- development of sustainability assessment plan.

There were extensive discussions on these topics during the EG meeting in December 2014. With valuable input from EG and PG members, the issues of definition and scope are now clearer. One of the important decisions was to keep the current narrow definition of sustainability for the first phase of the TF's work, and then the wider or more common definitions of sustainability assessment will be dealt with in the 2nd phase of the TF's work. These may include models that take into account national data, country-specific strategies for the back end of the fuel cycle, and the geological repository for high-level waste. During the PG meeting in December 2014, it was indicated that a country-wise analysis may be prone to arbitrariness, and again raises questions on the value of the assessment itself and its usage. The TF will have to find a way to overcome the innate inconsistency problem when an individual assessment result is to be compared with others and to add values to it.

The future path authorised in the December 2014 PG meeting is as follows:

- TD will complete the establishment of TF;
- TD will provide support for TF activities to implement an action plan approved by PG;
- TD will organise a sustainability session during the next GIF/INPRO interface meeting;
- Co-chair of the TF will make a status report at the PG/EG meetings in May 2015.

During the early stages of establishing the TF, frequent advice and guidance from PG/EG will be necessary. Moreover, expectations on the sustainability TF need to be shared among the EG, PG, and Interim Sustainability TF members for a fruitful outcome of TF activities.

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. The 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 long-term strategic issues, including regulatory, commercial or technical considerations. At its meeting in May 2014, the Policy Group asked the SIAP to consider and advise the GIF on the supply chain for Gen IV systems, including identification of issues or gaps in the supply of non-LWR reactor materials, structures, systems, and components. The SIAP was also asked to consider what the GIF community could do to enhance knowledge management in advanced reactor R&D, given the history of knowledge management in the LWR industry.

On the topic of supply chain considerations, the SIAP agreed that this is an important issue and major risk to the Gen IV schedule, both for prototype and commercial deployment. Concept developers are advised to be realistic about the choice of materials so that the supply chain can be established in a reasonable amount of time, thus SSCs and Working Groups should consider the availability of materials and industrial practices. Specific considerations that should be addressed early in the R&D phase with potential suppliers and operators include:

- preliminary supply chain for materials and equipment within different disciplines (mechanical, electrical, instrumentation, etc.);
- feasibility of materials, components, instruments given the expected conditions (pressure, chemistry, radiation);
- manufacturability of materials on a production scale and qualified for use in a nuclear plant;
- development of specifications for materials and procure materials for a prototype at completion of preliminary design;
- initiation of qualification of materials for the commercial design, including supplier qualification.

The SIAP encouraged development of international standards and specifications to support supply chain development in multiple countries. Including human capital as part of the supply chain, the SIAP encouraged identification of qualified personnel needed for all phases of the deployment cycle. The SIAP particularly noted that systems with new types of fuel will require early attention to the front end of the fuel cycle due to the long time required to develop and qualify new fuel forms and fabrication capabilities.

On the topic of knowledge management, the SIAP highlighted the following areas:

- public information: GIF systems should ensure availability of general information for the benefit of future participants;
- continuity of information/lifetime management of information used in the design and verification of Gen IV systems;
- design with decommissioning in mind (IAEA guidance);
- capture of expert knowledge in a manner that “survives” changes in personnel.

The SIAP encouraged working with IAEA and other organisations performing similar activities.

Beyond the two requested topics, the SIAP provided input on additional topics that could affect the direction or prioritisation of GIF R&D, including those related to challenges associated with financing of prototype and demonstration facilities and establishment of a regulatory structure. The SIAP requested a briefing on GIF efforts to harmonise safety licensing of Gen IV concepts.

Chapter 7. Other international initiatives

7.1 International project on innovative nuclear reactors and fuel cycles (INPRO) and other interactions with the IAEA

GIF/INPRO interface meeting

The GIF has been working on crosscutting areas with the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) of the IAEA since 2009. The INPRO mainly focuses on the following areas to ensure that nuclear energy is available to contribute to meeting the energy needs of the 21st century in a sustainable manner: National long-range nuclear energy strategies; Global nuclear energy scenarios on sustainable nuclear energy; Innovations in nuclear technology; and the Dialogue Forum.

In 2014, the annual GIF-INPRO interface meeting was held at the IAEA headquarters, 4-5 March, covering various R&D areas. The INPRO action plan for 2014-15 was introduced and there were presentations on GIF reactor system development status followed by safety, proliferation resistance and nuclear security, and economics. There were also INPRO project briefings of possible interest to GIF. As a result of the meeting, the participants discussed and updated the co-operation matrix, mainly for topics related to the three GIF working groups. Presentation materials are available for the public and can be found on the IAEA website.

The 9th GIF/INPRO interface meeting is scheduled for 4-5 March 2015 at the IAEA headquarters. One notable topic to be discussed is the sustainability issue, which is discussed above.

GIF/IAEA safety workshop

In parallel to the annual interface meeting with INPRO, the GIF has co-operated with the IAEA by organising workshops on safety of GIF reactor systems. There have been four safety workshops on sodium-cooled fast reactor covering various issues: Operational and Safety Aspects of SFR; Safety Design Criteria for SFR, and Safety of SFR. The 5th safety workshop is scheduled for 23-24 June 2015 and the Safety Design Guidelines on safety approach and design conditions for the SFR will be discussed in this workshop.

Along with the GIF/INPRO co-operation, the safety workshop continues to serve as an effective GIF-IAEA collaboration and possibilities of workshops on VHTR topics will be explored.

7.2 International Framework for Nuclear Energy Cooperation (IFNEC)

On 16-17 June 2010 in Accra, Ghana, the Partner countries of the Global Nuclear Energy Partnership (GNEP) formally agreed to transform the partnership into the International Framework for Nuclear Energy Cooperation (IFNEC). The transformation from GNEP was agreed upon by the Partners in order to explore mutually beneficial approaches to ensure the expansion of nuclear energy for peaceful purposes proceeds in a manner that is efficient, safe, secure, and supports non-proliferation and safeguards.

As of October 2014, members of IFNEC consist of 63 participant and observer countries and four permanent Observer International Inter-governmental Organisations including the Generation IV International Forum.

IFNEC is led by an Executive Committee, which is made up of ministerial-level officials from Participant countries. The Steering Group manages the implementation of Executive Committee decisions as well as day-to-day and organisational business. IFNEC's working groups carry out specific topical activities. There are currently two IFNEC working groups: the Infrastructure Development Working Group (IDWG) and the Reliable Nuclear Fuels Services Working Group (RNFSWG).

The Executive Committee held its annual meeting on 17 October 2014, in Seoul, Korea. GIF was represented by Dohee Hahn, GIF Technical Director, who co-chaired the Executive Committee meeting.

The Executive Committee re-affirmed the value of IFNEC to each of its participant and observer countries as they develop and deploy nuclear energy resources that are safe, secure, and environmentally friendly and have a high level of proliferation resistance. It needs to be noted that participation in IFNEC Working Group meetings can provide opportunities of collaboration between GIF and IFNEC for the development of future reactors.

7.3 Interaction with regulators (MDEP, CNRA)

In 2014, GIF had productive interactions with national nuclear regulators primarily through the two multinational regulatory organisations that deal with the regulation and design review of new reactors: the NEA Committee on Nuclear Regulatory Activities (CNRA) and the Multinational Design Evaluation Programme (MDEP).

The CNRA is an international committee made up of senior representatives from regulatory bodies. It was created in 1989 to guide the NEA programme concerning the regulation, licensing and inspection of nuclear installations with regard to safety. The CNRA's main tasks are exchanging information and experience among regulatory organisations, reviewing developments which could affect regulatory requirements, and reviewing current practices and operating experiences.

MDEP was established in 2006 as a multinational initiative to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities who are currently reviewing or will in the near future receive application to review new reactor designs. National regulators from fourteen countries are members of MDEP. Current MDEP work area is organised in five reactor design-specific working groups and 3 issue-specific working groups that meet several times a year. NEA facilitates MDEP's activities by acting as technical secretariat for the programme. The MDEP Policy Group (PG) and the Steering Technical Committee (STC) oversee the programme. Members of these committees are key decision makers in their respective regulatory organisations. Since the NEA provides technical secretariat service to MDEP, CNRA and MDEP activities are co-ordinated to preclude duplication.

In January 2014, GIF representatives met with MDEP STC members to discuss the SFR Safety Design Criteria (SDC) Phase-I Report which had been transmitted earlier to MDEP, CNRA and regulatory bodies of countries developing SFRs. The STC members welcomed the high-level discussion and encouraged such future interactions which they deem to be beneficial to both groups. They also noted that member national regulators retain sovereign authority for all licensing and regulatory decisions; hence MDEP will not provide any position on the report. MDEP members US and Chinese regulators have formally submitted their comment on the SDC Phase-I Report.

In December 2014, GIF representatives met with CNRA members to brief them on GIF activities and to discuss possible collaboration with the regulators to address the remaining policy and technical issues at early stage to facilitate future licensing of Gen IV plants. The meeting was attended by NEA Director-General, William D. Magwood, IV, and via video conferencing, the GIF Policy Group chair, Dr John Kelly from Washington. Mr. Thomas J. O'Connor, US Representative to the GIF Policy Group presented an overview on GIF concepts and goals and a proposed path forward for collaboration with CNRA. Dr. John Kelly thanked the group for the opportunity to learn how CNRA operates and expressed his hope that regulators will collaborate with GIF as the US NRC did during the early stages of the AP1000 (Gen III+ passive reactor) development. The chair of the CNRA, Dr Jean-Christophe Niel, thanked the GIF representatives for their presentations and their desire to work with CNRA to pave the way for future licensing of Gen IV reactors. The CNRA and the Committee on the Safety of Nuclear Installations (CSNI) will discuss in 2015 the opportunity of creating a joint ad-hoc group to address regulatory and safety issues related to Gen IV reactor designs.

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 the public health and the environment.*
Economics-1	*Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.*
Economics-2	*Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.*
Safety and Reliability-1	*Generation IV nuclear energy systems operations will excel in safety and reliability.*
Safety and Reliability-2	*Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.*
Safety and Reliability-3	*Generation IV nuclear energy systems will eliminate the need for offsite emergency response.*
Proliferation Resistance and Physical Protection	*Generation IV nuclear energy systems will increase the assurance that they are very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.*

These goals guide the 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 for Generation IV Nuclear Energy Systems* (2014), which can be downloaded at 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 Systems
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 arrangement
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)
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 (Korea)
MOST	Ministry of Science and Technology (People's Republic of 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 (Korea)
NRI	Nuclear Research Institute (Czech Republic)
NSSC	Nuclear Safety and Security Commission (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)
VTT	Valtion Teknillinen Tutkimuskeskus (Technical Research Centre of Finland)

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This eighth edition of the Generation IV International Forum (GIF) Annual Report highlights the main achievements of the Forum in 2014, and in particular progress made in the collaborative R&D activities of the eleven existing project arrangements for the six GIF systems: 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 for the lead-cooled fast reactor and the molten salt reactor is also reported. In May 2014, China joined the supercritical-water-cooled reactor system arrangement; and in October 2014, the project arrangement on system integration and assessment for the sodium-cooled fast reactor became effective. GIF also continued to develop safety design criteria and guidelines for the sodium-cooled fast reactor, and to engage with regulators on safety approaches for generation IV systems. Finally, GIF initiated an internal discussion on sustainability approaches to complement ongoing work on economics, safety, proliferation resistance and physical protection.

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