ANNUAL REPORT 2015

GEN IV International Forum℠

Foreword from the GIF Chair

It is 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. GIF experienced a number of important changes in 2015, notably a transition in leadership and a renewal of our legal basis for collaboration. External outreach also expanded significantly in accordance with the GIF Strategic Plan. Last but not least, a significant collaboration milestone was marked with the receipt of the 1 000th deliverable under GIF collaboration.

The co-founder and first GIF Chairman, William D. Magwood, IV, became Director-General of the Nuclear Energy Agency (NEA) in late 2014. The NEA, among its many roles, acts as the Technical Secretariat for GIF. We would like to congratulate Mr Magwood on this important role in global nuclear energy. Thierry Dujardin, the Acting Deputy-Director General at the NEA and long-standing face of the NEA for the GIF Policy Group, retired in March 2015. Henri Paillere, Senior Nuclear Analyst, assumed Mr Dujardin's GIF responsibilities, in addition to the GIF support that he has already provided for the past several years.

Four additional leadership transitions should also be noted in GIF during the year 2015. Hideki Kamide succeeded Kazumi Aoto as a Vice-Chair of the Policy Group. Haeryong Hwang was elected Chairman of the Senior Industry Advisory Panel (SIAP), following a two-year vacancy in that position. Francois Storrer succeeded the recently retired Jean-Claude Bouchter as GIF Policy Director. Following Dohee Hahn's acceptance of the position of Director of the Nuclear Power Division at the International Atomic Energy Agency (IAEA), Alexander Stanculescu replaced Mr Hahn as GIF Technical Director. In his new role at the IAEA, Mr Hahn will be the principal IAEA interface with GIF. I would like to thank Mr Aoto, Mr Dujardin, Mr Hahn, Mr Bouchter, Mr McFarlane and other GIF leaders who have moved on this year from their dedicated service to GIF.

The year 2015 marked the tenth anniversary of the signing of the Framework Agreement that allowed collaborative research and development to be organised under the GIF banner. Because the Framework Agreement was valid for only ten years, a Framework Agreement Extension was developed and has been signed by a majority of active GIF members. In 2016, our attention will turn to extending the system arrangements, which currently have a ten-year expiration. These essential legal documents were the subject of much discussion at the two GIF Policy Group meetings in Chiba, Japan, and Saint Petersburg, Russian Federation. Currently, active members in GIF are Canada, the People's Republic of China, the European Atomic Energy Community (Euratom), France, Japan, Korea, Russia, South Africa, Switzerland and the United States. Argentina, Brazil and the United Kingdom are inactive members but remain cognisant of the Forum's activities. In October, Australia presented its petition to become a GIF member to the Policy Group. The Policy Group will consider this petition in 2016.

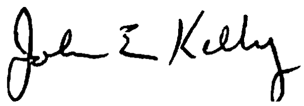
In 2015, the third GIF Symposium was held in conjunction with the 23rd International Conference on Nuclear Engineering (ICONE23) at Makuhari Messe, Chiba, Japan. In addition to providing the latest updates on progress in five of the generation IV systems, GIF organised a panel on fast spectrum testing, as well as a panel on the role of nuclear energy in national energy policies, and it reported on progress within the GIF task forces and working groups.

GIF maintains a long-standing, collaborative relationship with the IAEA with a traditional emphasis on the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). Co-operation on evaluation methodologies for economics, safety, physical protection and proliferation resistance has been ongoing for several years. This year, GIF co-operation with INPRO was expanded to include other IAEA technical departments, while maintaining close ties to INPRO. GIF and the IAEA/INPRO held their ninth interface meeting in March 2015 to discuss areas of mutual interest in technology status, convergence of assessment methodologies and progress on items in the co-operation matrix that summarises the agreement between the two organisations. GIF and the IAEA also sponsored the fifth workshop on the safety of sodium fast reactors (SFRs) in June 2015. This year, the workshop emphasised design criteria, design guidelines and practical approaches to achieving these goals.

Three GIF task forces remained active in 2015. The most advanced is the task force that developed safety design criteria (SDC) for the SFR. The SFR SDC report had been previously distributed for external review to national regulators and international organisations, and in 2015 a new report offering guidelines on implementing the design criteria was completed. The NEA helped GIF begin a dialogue on the safety of advanced reactors with the NEA Committee on Nuclear Regulatory Activities (CNRA) and the NEA Committee on the Safety of Nuclear Installations (CSNI). Subsequently, these two NEA committees created the Ad hoc Group on the Safety of Advanced Reactors, which will, *inter alia,* help identify needed safety research in anticipation of licensing. The Sustainability Task Force completed its phase 1 assignment, as reported later in this report. The Education and Training Task Force was revised to include an early focus on webinars and adding value to international schools that have already been successfully established.

Progress continues on research and development (R&D) for the six GIF advanced reactor systems. After a year's hiatus, activity on the gas fast reactor is resuming under new leadership. A new member from Korea has joined the Lead Fast Reactor System Steering Committee, which operates under a Memorandum of Understanding. The other four systems continue to make steady progress, as described in this report.

Finally, I have the privilege of informing you that François Gauché will be the new GIF Chair. I congratulate Mr Gauché and his team and look forward to working with them. I have been honoured to serve as Chair of the Policy Group for the last three years, and I trust that the GIF members will afford Mr Gauché the same great level of support that I have enjoyed.



Dr John E. Kelly
GIF Policy Group Chairman

A tribute

The end of the year was unfortunately a sorrowful one as the GIF lost two distinguished researchers who made great contributions to the work of the Forum: Jan Kysela from the Czech Research Centre, Řež, and Philippe Dufour from the French Atomic Energy and Alternative Energy Commission.

Jan Kysela was a key leader in the generation IV supercritical water-cooled reactor community. He was one of the world's leading water chemists working on water-cooled reactors. He was an experienced, well-respected and insightful researcher, who brought his extensive knowledge to the supercritical-water-cooled reactor (SCWR) community. In addition to his scientific knowledge, Jan had a very deep and practical understanding of the problems encountered in operating nuclear reactors that turned the focus of discussions on SCWR materials and chemistry away from the laboratory and to the realities of what was required to build a functional reactor.

Philippe Dufour contributed greatly to the French fast neutron reactor programme and was directly involved in the progress made on the Phénix and Super-Phénix sodium-cooled reactors. He also led a cost-optimisation programme that had direct implications on the European Fast Reactor Project. His competence in both the operational and accidental aspects of sodium fast reactors (SFRs) was recognised by all, such that he played a key role in many international collaborations, not least within GIF where he chaired the SFR System Steering Committee since October 2013. Philippe Dufour had also been very active in the French ASTRID programme.

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

1.1 Generation IV International Forum 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. The charter was extended indefinitely in 2011. Signatories of the charter are expected to maintain an appropriate level of active participation in GIF collaborative projects.

Among the signatories to the charter, ten members (Canada, France, Japan, China, Korea, Russia, South Africa, Switzerland, the United States and Euratom) have signed or acceded to the Framework Agreement (FA) as shown in Table 1.1. On 26 February 2015, during a signing ceremony hosted in Paris by the OECD Secretary-General, depositary of the Framework Agreement, the Agreement Extending the Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy Systems was signed by four countries (France, Japan, Korea and the United States). In accordance with Article II of the Extension Agreement, as more than three parties have thereby indicated their consent to be bound, the Extension Agreement entered into force on that day. Accordingly, the GIF Framework Agreement is extended for another ten years, until 28 February 2025. The Agreement Extending the Framework Agreement was later signed by Russia in June, by Switzerland in August, and by South Africa in September. Canada, China and Euratom are expected to sign in 2016.

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 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 (SAs) 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 (MoUs) were signed in 2010 by France and the European Union, and the European Union and Japan, respectively. Russia signed the LFR MoU in 2011 and the MSR MoU in 2013. In November 2015, Korea signed the LFR MoU and Switzerland the MSR MoU. Switzerland also withdrew from the GFR SA. The participation of GIF members in SAs and MoU is shown in Table 1.1.

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

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.

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 2015, the first PG meeting was held in Makuhari Messe (Japan) in May, back-to-back with the 3rd GIF Symposium and hosted by the Japan Atomic Energy Agency (JAEA), the second PG meeting was held in Saint Petersburg (Russia) in October and hosted by Rosatom.

Table 1.1: **Parties to GIF Framework Agreement, System Arrangements and Memoranda of Understanding as of 31 December 2015**

Member	Implementing agents	Framework Agreement	System Arrangements				Memoranda of Understanding	
		Date of signature or receipt of the instrument of accession/(extension)	GFR	SCWR	SFR	VHTR	LFR	MSR
Argentina								
Brazil								
Canada	Department of Natural Resources (NRCan)	02/2005 (-)		11/2006				
Euratom	European Commission's Joint Research Centre (JRC)	02/2006 (-)	11/2006	11/2006	11/2006	11/2006	11/2010	10/2010
France	Commissariat à l'énergie atomique et aux énergies alternatives (CEA)	02/2005 (02/2015)	11/2006		02/2006	11/2006		10/2010
Japan	Agency for Natural Resources and Energy (ANRE) Japan Atomic Energy Agency (JAEA)	02/2005 (02/2015)	11/2006	02/2007	02/2006	11/2006	11/2010	
Korea	Ministry of Science, ICT and Future Planning (MSIP) and Korea Nuclear International Cooperation Foundation (KONICOF)	08/2005 (02/2015)			04/2006	11/2006	11/2015	
People's Republic of China	China Atomic Energy Authority (CAEA) and Ministry of Science and Technology (MOST)	12/2007 (-)		05/2014	03/2009	10/2008		
Russia	Rosatom	12/2009 (06/2015)		07/2011	07/2010		07/2011	11/2013
South Africa	Department of Energy (DOE)	04/2008 (09/2015)						
Switzerland	Paul Scherrer Institute (PSI)	05/2005 (08/2015)				11/2006		11/2015
United Kingdom								
United States	Department of Energy (DOE)	02/2005 (02/2015)			02/2006	11/2006		

The third GIF Symposium was held jointly with the 23rd International Conference on Nuclear Engineering (ICONE23). GIF symposia are organised triennially, with the first symposium held in France in 2009 and the second held in the United States in 2012. The 2015 symposium, which took place just over a year after the publication of the updated GIF Technology Roadmap, attracted about 100 participants. After an introduction by the GIF Chair Dr John Kelly, participants discussed progress made in the development of the six GIF systems, as well as in the three methodology working groups. The symposium ended with a lively question and answer session. Presentations from the symposium may be found on the GIF website. 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. The Chair of the EG is known as the Technical Director. The position of Technical Director was held by Korea until October 2015, when a new Technical Director from the United States was nominated.

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.

Figure 1.1: **GIF governance structure in 2015**

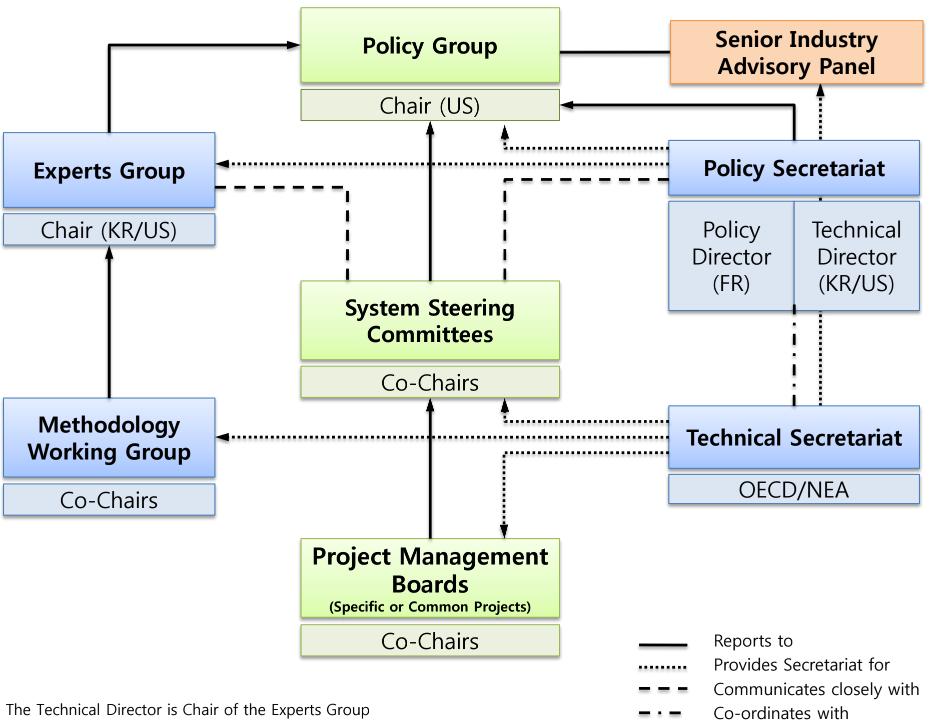


Figure 1.2: **Policy Group in Saint Petersburg (October 2015)**

Three methodology working groups (MWGs), the Economic Modeling 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 time frame. The progress status of three 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 another one dedicated to the issue of sustainability and a third dedicated to the Education & Training activities that are being developed within the GIF.

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. In 2015, a new Chair of the SIAP was elected, Mr Hwang from the Korea Electric Power Corporation (KEPCO).

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 assists the PG on technical matters. The technical secretariat, provided by the Nuclear Energy Agency (NEA), supports the SSCs, PMBs, MWGs, TFs, as well as the SIAP 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, an amendment for the PA for the SFR Advanced Fuel (AF) became effective in October 2015, with partners CEA, CIAE, DOE, JAEA, JRC, KAERI, and Rosatom.

Table 1.2 shows the list of signed arrangements and provisional co-operation within GIF as of 31 December 2015.

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 2015**

	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	X	X		X		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		
GFR SA			X	X	X						
CD&S PA	17-Dec-09		X	X							
FCM PA	Provisional		P	P	P						
LFR MoU			X		X	O	X		X		O
MSR MoU			X	X	O	O	O		X	X	O

X = Signatory P = Provisional participant O = Observer S = Signature process ongoing

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

Chapter 2. Highlights from the year and country reports

2.1 General overview

In a ceremony at the Paris headquarters of the Organisation for Economic Co-operation and Development (OECD) on 26 February 2015, four members signed the extension document of the Framework Agreement (FA) and the extension of ten years until 2025 was put into force. With the signing of all remaining member countries by early 2016, the FA extension will be fully executed. The next step is to work on the extension of the system arrangements that will expire in 2016.

The Australian government's Department of Industry and Science have petitioned to join the Generation IV International Forum (GIF). At the 40th Policy Group Meeting held on 29-30 October 2015 in Saint Petersburg, Russia, representatives from the Australian Nuclear Science and Technology Organisation (ANSTO) presented an overview of their nuclear research and development activities and facilities. The Policy Group is considering the petition and, upon an invitation by ANSTO, a Policy Group delegation will visit Australia in 2016 to assess ANSTO's capabilities.

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. The 9th GIF-INPRO Interface Meeting was held in March 2015 in Vienna, Austria. Topics included GIF reactor system development status, technical development status reports, and safety topical sessions for proliferation resistance and economics.

The third GIF Symposium was held in co-ordination with the 23rd International Conference on Nuclear Engineering in May 2015 at Makuhari Messe, Chiba, Japan. The symposium provided the opportunity to disseminate GIF's various technical activities to a broader audience. Panel discussions facilitated information exchange about national energy strategies from the perspectives of energy security and reduction of greenhouse gas emissions, and on fast neutron spectrum irradiation capabilities.

The 5th GIF-IAEA SFR Safety Workshop was held in June 2015 in Vienna, Austria. As a broader forum with participation of a larger number of designers, regulators, and industry than what is represented under GIF, these GIF-IAEA workshops offer a unique platform for information exchange and knowledge sharing. Main objective was to assist SFR developers and vendors in utilising the SFR safety design criteria (SDC) in their design process for improving the safety – including the use of inherent/passive safety features and design measures for prevention and mitigation of severe accidents. The workshop further expanded its scope into review and feedback of the safety design guidelines (SDG) currently being developed by the GIF SDC Task Force.

There had been interactions between GIF and the Nuclear Energy Agency's (NEA) Committee on Nuclear Regulatory Activities (CNRA) in terms of licensing frameworks for advanced reactors. As a result, an Ad hoc Group on the Safety of Advanced Reactors (GSAR) was formed in co-operation with NEA's Committee on the Safety of Nuclear Installations (CSNI). For now, GSAR is focusing on SFR safety issues. Over the next few years, the group is expected to have continued engagement with GIF and other organisations.

GIF provided a statement for the United Nations Climate Change Conference (COP21) held in December 2015 in Paris, to call for policymakers to acknowledge the real contributions that nuclear energy is making today to the mitigation of carbon emissions from the power sector, and to consider endorsing the deployment of advanced reactors to enhance further decarbonisation of the world's energy mix in the decades to come.

2.2 Highlights from the Experts Group

The Experts Group (EG) advises the Policy Group on research and development strategy, priorities and methodology as well as the assessment of research plans prepared in the framework of the system arrangements.

The Policy Group approved the SFR Safety Design Criteria (SDC) report prepared by the SFR SDC Task Force. This report incorporated comments by the US Nuclear Regulatory Commission (NRC) and the IAEA. In the next phase of its activities, the SFR SDC Task Force is focusing on the development of SFR SDG. The first report on "SFR Safety Design Guidelines on Safety Approach and Design Conditions for Generation IV Sodium-cooled Fast Reactor Systems" (mainly dealing with reactivity loss and heat removal issues) was completed in October 2015. After consideration of comments received by Policy Group members, the report will be circulated to IAEA and GSAR for commenting. Currently, the SFR SDC Task Force is working on three complementary SDG documents, *viz.* on key structures, systems, and components.

The Education and Training Task Force (ETTF) has been revitalised to disseminate open GIF materials via social media, webinars and brochures. The Terms of Reference document has been developed for a term of three years. After identifying the target groups and their needs, the ETTF will disseminate open GIF material, place education and training resources on the GIF website, and make use also of other social media resources, as appropriate. ETTF will develop a prototype of webinar series, which if successful, could lead, along with the other activities, to the establishment of a standing GIF Working Group for a longer-term effort.

The mandate for the Sustainability Task Force is to consider sustainability in the narrow sense of the GIF goal, i.e. resource utilisation and waste management. The task force reviewed the legacy of GIF work on GIF screening and methodology experiences and found that there were no fundamental changes in the understanding of sustainability. The task force (TF) also looked at what activities on sustainability have been performed by the IAEA, NEA, and the US Fuel Cycle Options Study. The expert group members are developing sustainability frequently asked questions (FAQ) that will be published on the GIF website. The GIF state-of-knowledge with regard to sustainability is summarised in this annual report.

The Proliferation Resistance and Physical Protection Working Group (PRPPWG) conducted outreach activities targeting students and the general public and aiming at enhancing their awareness of the issues related to the PRPPWG scope of work. The major accomplishments of the working group were revisions of the methodology based on a sodium-cooled fast reactor (SFR) case study, and in various joint activities with the six Gen IV systems. The PRPPWG has contributed to the efforts of the Risk and Safety Working Group (RSWG), specifically the drafting of the SDC, by ensuring the interface between safety and security.

The main activity of the RSWG consists in the development, in collaboration with the system steering committees, of technology specific (one for each Gen IV system) safety white papers. The SFR White Paper has been completed, and the very-high-temperature reactor (VHTR) one is currently under revision by the respective System Steering Committee. The other Gen IV systems white papers are under development. Another ongoing major effort by the RSWG consists in the preparation of safety assessment reports for all six Gen IV systems. These assessment reports are documenting the outcome of the application of the Integrated Safety Assessment Methodology (ISAM) to specific Gen IV systems.

The Economics Methodology Working Group (EMWG) worked on strategic initiatives and project planning, and identified conditions for success and failure in the project planning.

Switzerland withdrew from the Gas-cooled Fast Reactor System Arrangement and the "Conceptual Design and Safety" (CD&S) project Arrangement. The Gas-cooled Fast Reactor System Steering Committee decided to continue the CD&S project between Euratom and France.

With regard to the supercritical water-cooled reactor (SCWR) activities, the system research plan is being updated to reflect main directions in terms of the three projects, *viz.* thermal hydraulics, materials and chemistry, and system integration and assessment.

The VHTR project plans are being updated to reflect the fact that, upon China signing the Framework Agreement Extension, the Institute of Nuclear and New Energy Technology (INET) is joining the Hydrogen Production and Materials project.

The Seoul National University signed the Memorandum of Understanding (MoU) for collaboration on the lead-cooled fast reactor (LFR). The Paul Scherrer Institute signed the MSR MoU in November 2015.

The AFR-100 design was approved as a new SFR system design track by the System Integration and Assessment project.

During the EG meetings, results of actions in GIF member countries to implement strategic planning recommendations were reviewed in the areas of sharing capabilities and resources, communications, SIAP and engagement with external organisations.

2.3 Country reports

Canada

New developments in Canada Nuclear Liability and Compensation Act updated

On 26 February 2015, Canada's Parliament passed new legislation, the Nuclear Liability and Compensation Act, to replace the existing 1976 Nuclear Liability Act. The new legislation will strengthen Canada's nuclear liability regime to better deal with liability and compensation for a nuclear accident within Canada, and will implement Canadian membership in the IAEA Convention on Supplementary Compensation for Nuclear Damage. The new act will set the monetary limit for operator liability to CAD 1 billion, to be phased in over four years from CAD 650 million at entry-into-force. The new liability amount – increased from the CAD 75 million under the 1976 Nuclear Liability Act – is commensurate with current international standards. The new act is expected to come into force in January 2017, once key regulations and financial security mechanisms are in place.

New developments/industry updates

The restructuring of Atomic Energy of Canada Limited (AECL) has been completed with the transition of AECL's nuclear laboratories to a "government-owned, contractor-operated" (Go-Co) model. In 2015, Canadian National Energy Alliance (CNEA) was selected through a competitive procurement to manage and operate Canadian National Laboratories (CNL). The CNEA, a consortium made up of CH2M HILL, WS Atkins, Fluor, SNC-Lavalin and Rolls-Royce. AECL remains a Crown corporation, with the new mandate to provide oversight of the performance of CNL to ensure the government expectations are met.

The Provincial Government of Ontario announced the CAD 12.8 Billion refurbishment of four units at the Darlington nuclear power plant (NPP). The rebuild will start fall 2016 and will extend life of the units by 30 years. A contract was also signed between the Province of Ontario and Bruce Power to refurbish the remaining six units at the Bruce NPP. The total investment required to refurbish the ten units is estimated to be CAD 25 Billion. The refurbishment of the ten nuclear units results in annual emissions savings for Ontario of between 31.8 to 53.3 mtCO_2e, based on whether the plants are displacing natural gas or coal respectively. In addition, the Province also announced that it will seek approval from the CNSC to operate six of the Pickering station units past the original timeline of 2020; two units until 2022, and four units until 2024.

SCWR progress

In 2015, Canada completed the development of the Canadian Super-critical Water-cooled Reactor (SCWR) concept. It is a light-water-cooled and heavy-water moderated pressure-channel-type reactor that adopts the mix (reactor-grade) plutonium and thorium fuel. It is designed to operate at the pressure of 25 MPa with a core outlet temperature of 625°C matching the conditions of an advanced high-pressure turbine. Two reviews of the Canadian SCWR concept were conducted to assess it against the goals set by the GIF for the generation IV reactor concepts. The first review involved prominent Canadian nuclear industry experts and was held in February 2015. These experts included leaders from industry, nuclear associations, and government stakeholders. The reviewers congratulated researchers on the technical advancements and the innovative features of the Canadian concept. They provided valuable recommendations to improve the concept from an operational point of view. The second review of the SCWR concept was held in October 2015. International members participating in the SCWR system of the Generation IV International Forum were invited to the international expert review of the Canadian SCWR concept. These members are subject matter experts from China, the Czech Republic, Finland, Germany, Japan and the Netherlands, and have been participating in developing the SCWR concepts in China, the European Union (EU) and Japan. Canadian researchers provided presentations on various technology areas to the reviewers in the two-day meeting. Details on the concept and technical issues were discussed thoroughly. Reviewers praised the report on the Canadian SCWR concept and the organisation of the meeting. The Canadian SCWR concept was demonstrated to meet the GIF technology goals on enhancing economics, safety, proliferation resistance, and sustainability using the methodologies developed by the GIF Cross-Cutting Working Groups (except for the sustainability where a working group has not yet been established).

China

Nuclear energy policy

China adheres to the policy of developing nuclear power in a safe and efficient manner. "Safety first, quality first" has consistently been the fundamental policy of nuclear industry in China. To further strengthen the peaceful use of nuclear energy and nuclear safety management, the nuclear legal system has been continuously improving. The Atomic Energy Law (final draft) has been submitted to the State Council for review. The drafting of the Nuclear Safety Law is under progress. Nuclear Security Regulations and Nuclear Power Management Regulation have already been incorporated in the administrative legislation plan of the State Council in 2015.

Operation and construction of nuclear power plants

The in-service nuclear power units have maintained a good record in safety and operation performance, and the projects under construction are progressing as scheduled. By the end of September 2015, there were 26 nuclear power units in commercial operation in China mainland, with total installed capacity of 24.688 GWe. The total nuclear power generation amounted to 124.261 billion KWh from January to September in 2015, approximately 2.96% of the total mixed power generation nationwide.

Twenty-five nuclear power units are under construction with the installed capacity of ~27 GWe. Construction continues on four AP1000 units at two sites in Sanmen and Haiyang. Two EPR units are also being installed on schedule at its Taishan site. Construction of the first demonstration project for Hualong One Reactor (HPR-1000) was started at Fuqing nuclear power plant in May 2015.

Gen IV nuclear energy systems R&D

Very-high-temperature reactor (VHTR)

Research and development (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 grid by the end of 2017. By 2015, the civil engineering work was nearly completed, and equipment installation was started. The full-scale tests of key components and systems are finished. The reactor pressure vessel successfully finished the hydrostatic test and will be shipped to the site soon. HTR-PM adopts the steam cycle, with the steam turbine system driven by two reactor modules. HTR-PM will demonstrate the technology's maturity and near-term market potential of VHTR. The research and development of HTR-PM600 power plant has already started in China. HTR-PM600 is featured as a 600 MWe steam turbine driven by the steam from six reactor modules, with the capability of cogeneration, and each reactor module adopts same design of HTR-PM. This will further push forward the commercial deployment of V/HTR technology.

The irradiation test on fuel samples were finished in 2014, the heating up test on irradiated fuel samples are undergoing. The fuel production line for HTR-PM with capacity of 30 000 spherical fuel elements per year has entered into trial production stage since September 2015.

Sodium-cooled fast reactor (SFR) R&D

China Experimental Fast Reactor (CEFR) was restarted and the gross electricity production has reached 5.8×10^6 kWh by the middle of October, 2015. The No. 2 cold trap was replaced and some maintenance work has been done on the cooling water circulating pump, vacuum system of main condenser, etc. The R&D work of mixed oxide (MOX) fuel of CEFR is ongoing as planned. As one of the design tracks in the GIF SFR SIA project, the pre-conceptual design of CFR1200 was started. The conceptual design of the China demonstration fast reactor (CFR600) project has been completed, and the preliminary design has continued.

Supercritical water-cooled reactor (SCWR) R&D

China has taken part in the GIF R&D activities on the Supercritical Water-cooled Reactor Nuclear Energy System in 2015. Nuclear Power Institute of China and Shanghai Jiaotong University have been authorised to join the Thermal-Hydraulics and Safety (TH&S) Project Management Board (PMB) and Materials and Chemistry (M&C) PMB respectively as representatives of China's SCWR research consortium. In accordance with the requirements of the GIF SCWR System Steering Committee (SSC), the Chinese contribution to the project plans for TH&S and M&C were provided to the respective PMBs.

Research and development on SCWR and pre-conceptual design of the experimental reactor of CSR1000 have been proceeding. Several R&D activities have been started by different universities and institutes. The new project R&D on SCWR technology (phase II) has been accepted by government and will be started in 2016. It aims to finish 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 device etc. One new international benchmark exercise is being prepared jointly in Nuclear Power Institute of China (NPIC) and the Canadian Nuclear Laboratories (CNL) based on the supercritical water (SCW) bundle tests from the Chinese side and Canadian side. This benchmark exercise will be presented to GIF SCWR members to assess computational fluid dynamics (CFD) models. The international peer review on the CSR1000 concept design is planned in 2018.

Lead-cooled fast reactor (LFR) R&D

The Chinese Academy of Sciences (CAS) has launched the Accelerator-driven Subcritical System (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. 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. The KYLIN series lead-bismuth eutectic (LBE) experimental loops has already been constructed. The material corrosion, thermal-hydraulics experiments and key components technology validation are being performed. The engineering design of a lead alloy-cooled non-nuclear reactor named CLEAR-S to validate and test the key components and thermal-hydraulics phenomena of pool-type lead-based reactor has been finished. It is expected to be commissioned in 2016.

Molten salt reactor (MSR) R&D

In 2011, CAS initiated the "Thorium Molten Salt Reactor (TMSR) Nuclear Energy System" project. The aims of TMSR are to develop Thorium Energy utilisation, including non-electric application of nuclear energy based on TMSR in the next 20-30 years. In 2015, TMSR research centre completed the preliminary engineering design of the 10 MWth TMSR-SF1 and the concept design of the 2 MWth TMSR-LF1 as well as pyroprocess complex. The key technology and equipment of TMSR including the high-temperature FLiNaK molten-slat loop, the large nitrate natural circulation circuit, the control rod system, the fuelling and defueling system have been developed and established. An innovative dry-processing flow sheet of Th-U fuel cycle has been designed and its cold consistency has been realised at lab-scale.

Euratom

Euratom's contribution to GIF is based on two different components:

- R&D launched in the framework of Horizon 2020 programme.
- Programmes initiated in the framework of the Sustainable Nuclear Energy Technology Platform (SNETP) based on three pillars:
  - European Sustainable Nuclear Industrial Initiative (ESNII) for Gen IV fast neutron reactors (SFR, LFR and gas-cooled fast reactor [GFR]).
  - NC2I for nuclear cogeneration with a focus on (V)HTR.
  - NUGENIA dedicated to Gen II and III reactors but also addressing SCWR concept.

Horizon 2020 Euratom Research and Training Programme

A new seven-year European Union (EU) research programme called Horizon 2020, agreed and adopted by the EU Parliament and EU Council, started in 2014. Within this frame, the specific "Horizon 2020 Euratom Programme for nuclear research and training activities", supports the EU research in nuclear fission and fusion, including generation IV research. The Horizon 2020 Euratom Programme 2014-15 call for proposals includes a specific cluster "Support Safe Operation of Nuclear Systems" that contains themes linked to generation IV research such as "Improved safety design and operation of fission reactors" and "New innovative approaches to reactor safety". Several projects related to generation IV have been awarded in 2015 and started their implementation phase. These are:

- **Thermal-hydraulics Simulations and Experiments for the Safety Assessment of MEtal cooled reactors (SESAME)**

  This project supports the development of European liquid metal-cooled reactors (ASTRID, ALFRED, MYRRHA, SEALER) in full alignment with the ESNII roadmap. The project focusses on pre-normative, fundamental, safety-related challenges for these reactors.

- **A Paradigm Shift in Reactor Safety with the Molten Salt Fast Reactor (SAMOFAR)**

  This project deals with new innovative approaches to reactor safety for molten fuel fast reactors. The leading organisation is TU Delft (Netherlands) and the project consists of 11 partners from 5 EU countries (including Switzerland) and Mexico. The project, having a budget of EUR 5.2 million started on 1 August 2015 and will last for 48 months.

- **MYRRHA Research and Transmutation Endeavour (MYRTE)**

  The goal of MYRTE is to perform the necessary research in order to demonstrate the feasibility of transmutation of minor actinides in high-level waste at industrial scale through the development of the MYRRHA research facility. The leading organisation is SCK•CEN (Belgium) and the project consists of 27 partners from 7 EU countries (including

Switzerland). The project, having a budget of EUR 12.0 million started on 1 April 2015 and will last for 48 months.

- **Visegrad Initiative for Nuclear Cooperation (VINCO)**

  This project is a regional initiative aiming at developing nuclear research and training capacity building in nuclear technologies in Central European countries. The four participating countries defined their specialisations: helium technology in Czech Republic, design and safety analyses in the Slovak Republic, fuel studies in Hungary and material research in Poland. The leading organisation is the National Centre of Nuclear Research (NCBJ, Poland) and the project consists of six partners from five EU countries. The project, having a budget of EUR 1.1 million started on 1 September 2015 and will last for 36 months.

Sustainable Nuclear Energy Technology Platform (SNETP)

In 2015, the EU agreed on priorities to implement the “Energy Union”, targeted to achieve climate change mitigation, energy security and economic improvements. These “Integrated Strategic Energy Technology Plan (SET Plan)” priorities include a set of potentially effective and competitive low-carbon energy technologies to be developed and deployed in Europe, including nuclear energy. The Sustainable Nuclear Energy Technology Platform (SNETP) is is gathering stakeholders dealing with nuclear fission. As previously mentioned, SNETP is structured in three "pillars", namely: ESNII, NC2I and NUGENIA.

European Sustainable Nuclear Industrial Initiative (ESNII)

The European Sustainable Nuclear Industrial Initiative (ESNII) is devoted to fast neutron reactors with closed fuel cycles for improved sustainability through a better use of the uranium resource and improved management of high-level, long-lived waste via plutonium recycling and partitioning and transmutation of minor actinides. Presently, ESNII includes the study of three technology options that could lead to future industrial deployment: the sodium-cooled fast reactors (SFR) as reference technology, the LFR as a short-term alternative and GFR as a long-term development. Research is pursued in support to the construction of demonstration plants promoted by some European countries: ASTRID SFR in France, ALFRED LFR in Romania, and ALLEGRO GFR in central Europe (Czech Republic, Hungary, Poland and the Slovak Republic).

ESNII prepared several Euratom projects which usefully contributed to GIF projects (ESNII+ and JASMIN). Three were successfully completed in 2015 (THINS, FAIRFUELS and SEARCH):

- **ESNII+ (Preparing ESNII for HORIZON 2020)**

  ESNII+ is a large-scale integrating project with 37 partners from 14 countries including Switzerland and a budget of EUR 10.4 million over 4 years. The aim of this cross-cutting project is to develop a broad strategic approach to support the European Sustainable Industrial Initiative (ESNII) within the SET Plan. The project aims to prepare ESNII towards an efficient European co-ordinated research on Gen IV reactor safety linked with SNETP Strategic Research and Innovation Agenda and Deployment priorities.

- **Joint Advanced Severe accidents Modelling and Integration for Na-cooled fast neutron reactors (JASMIN)**

  JASMIN is a large-scale integrating project with nine partners from five countries and a budget of EUR 5.6 million over four years. This project will support the ESNII on the enhancement of SFR safety, especially towards a higher resistance to severe accidents.

- **Thermal-hydraulics of Innovative Nuclear Systems (THINS)**

  THINS was a large-scale integrating project with 24 partners and a budget of EUR 10.6 million over 5 years. It addressed cross-cutting and specific issues related to liquid metals, supercritical water and helium gas and thus produced useful input for several GIF projects.

- **FAbrication, Irradiation and Reprocessing of FUELS and targets for transmutation (FAIRFUELS)**

  FAIRFUELS was large-scale integrating project with 11 partners and a budget of EUR 7.3 million over 6.5 years. It provided a way towards a more efficient use of fissile material in nuclear reactors with a view to reducing the volume and hazard of high-level long-lived radioactive waste, closing the nuclear fuel cycle.

- **Safe ExploitAtion Related CHemistry for HLM reactors (SEARCH)**

  SEARCH was a medium-scale focused research project with 13 partners and a budget of EUR 5.7 million over 3.5 years. SEARCH supported the licensing process of the planned MYRRHA reactor by investigating the safe chemical behaviour of the fuel and coolant in the reactor.

Nuclear Cogeneration Industrial Initiative (NC2I)

The objective of NC2I is to demonstrate an innovative and competitive energy solution for the low-carbon cogeneration of heat and electricity based on nuclear energy. The targeted outcome is the commissioning of a nuclear cogeneration prototype to deploy this low-carbon energy technology in several energy-intensive industries. NC2I targets all non-electric applications of nuclear energy for lower-temperature applications such as seawater desalination or district heating; and higher temperature industrial applications such as chemicals production, oil refining, hydrogen production or advanced steelmaking. The latter require higher temperature output which can be provided by high-temperature gas-cooled reactors, or the GIF VHTR.

In the United States, the Next Generation Nuclear Plant (NGNP) programme targets an objective of licensing an HTR first-of-a-kind in the next decade. The NGNP Industry Alliance gathers industrial companies interested in the technology. In 2014, the NC2I and NGNP Alliance have established a transatlantic co-operation framework called GEMINI which held several meetings in 2015 on subjects like siting, financing options, early customers and design convergence for a demonstration plant.

NC2I had also prepared several Euratom projects which contributed to GIF projects. Two were successfully completed in 2015:

- **Advanced High-Temperature Reactors for Cogeneration of Heat and Electricity R&D (ARCHER)**

  ARCHER was a large-scale integrating project over four years with a budget of EUR 10 million. The consortium consisted of 14 partners. The activities were aligned with the GIF VHTR project structure, and direct collaboration existed within the project with international partners from the United States, China, Russia and co-operation with IAEA.

- **Nuclear Cogeneration Industrial Initiative – Research (NC2I-R)**

  As a co-ordination action over two years and with a budget of EUR 2.5 million mainly for desktop activities, NC2I-R supported the work of NC2I to gear up towards demonstration. The project had 22 project partners. It was structured in 5 work packages which addressed also non-technical areas, including organisational, legal and economic aspects.

Nuclear Generation II and III Association (NUGENIA)

The R&D dedicated to SCWR is embedded in the NUGENIA platform, an international non-profit association dedicated to the research and development of nuclear fission technologies, with focus on Gen II & III nuclear plants. NUGENIA gathers stakeholders from industry, research, safety organisations and academia, committed to develop joint R&D projects in the field.

One could highlight the following project concluded early 2015:

- **Supercritical Water Reactor – Fuel Qualification Test (SCWR-FQT)**

  SCWR-FQT was a collaborative project with eight partners from six countries and a budget of EUR 1.5 million over four years. The scope of the SCWR-FQT Euratom-China parallel project was to design an experimental facility for the qualification of fuel for the supercritical water-cooled reactor. The facility is to be operated in the LVR-15 research reactor in Czech Republic in the future. All necessary documents required for the licensing of the FQT facility by the Czech regulator was the main outcome of this project. Pre-qualification of the FQT facility will be carried out in China. Testing of a limited amount of commercially available nuclear grade materials which are candidates for fuel cladding were carried out within this project.

Some smaller Euratom contributions to GIF originate from projects initiated by communities outside SNETP, from non-nuclear R&D programmes and from member states. A particularly positive example is the contribution of the non-nuclear EU project SOPHIA (High-Temperature Steam Electrolysis) to the VHTR Hydrogen Production project.

France

Energy Transition Act

The new Law on Energy Transition for a Green Growth was published on 18 August 2015. This law sets the general framework of France energy policy with a number of ambitious objectives for a low-carbon, sustainable and robust energy mix.

Notable long-term objectives include:

- a 40% reduction of greenhouse gas (GHG) by 2030 compared to 1990;
- reduction of final energy consumption by 20% in 2030 and 50% in 2050;
- 32% renewables in final energy consumption by 2030;
- reduction of fossil energy sources of 30% by 2030 compared to 2012.

Regarding nuclear power, these objectives also include specific goals for nuclear with a cap of the authorised installed nuclear capacity at its current level (63.2 GWe) as well as a reduction of the share of nuclear in the electricity mix to 50% towards 2025 (from 75% at present).

In addition, the implementation and technical feasibility of the long-term objectives set out in the law are to be regularly reviewed and updated on a five-year basis. One should mention that the decision on potential nuclear reactor shutdown is left to the operator (EDF), in accordance with the safety authority's decisions.

Evolution of the governance of major French industrial and institutional actors

AREVA

In July 2015, a reorganisation of AREVA was announced, with EDF taking a stake of at least 51% in AREVA's reactor business (AREVA NP), that includes equipment and fuel manufacturing, as well as services for reactors. AREVA would keep a stake of about 15%, allowing for the participation of other minority partners.

In parallel, the French government announced in December 2015 that it will take a majority stake in AREVA naval propulsion and research reactor business (AREVA TA). CEA and France's naval shipbuilder DCNS are expected to both take a minority stake in the company.

Finally, the French government has also announced its will to contribute to recapitalise AREVA as it remains the main shareholder of the company. Following this reorganisation of the company, AREVA is expected to focus more specifically on nuclear fuel cycle business services.

CEA

Mr Daniel Verwaerde was appointed as the new Chairman and CEO of CEA in January 2015.

CIGEO: High-level waste deep repository

The National Radioactive Waste Disposal Organisation (ANDRA) is finalising CIGEO safety options file to be sent to the French Nuclear Safety Authority (ASN) in 2016.

CEA Saclay and Cadarache centres designated International Centres based on Research Reactors (ICERR)

Saclay and Cadarache DEN research centres have become the first to be designated International Centres based on Research Reactors (ICERR), international research hubs under a scheme launched by the IAEA last year.

The designation period covers 2015 to 2020 and is based on the Jules Horowitz Reactor (JHR) under construction in Cadarache, and other key facilities.

CABRI test reactor for light water reactor (LWR) safety studies has restarted on 20 October 2015

Cabri is a research reactor located at the Cadarache Centre, which is used to reproduce the conditions on a sample of irradiated nuclear fuel during a severe accident. This reactor is particularly used to reproduce the conditions of a reactivity-initiated accident (RIA).

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

In 2015, CEA completed the conceptual design phase of the ASTRID reactor. This includes the submission to ASN of a preliminary safety options case for ASTRID by the end of 2015.

Following this important milestone, the project is progressing with the basic design phase until the end of 2019.

One should also mention the strengthening of two important bilateral collaborations on ASTRID project: the broadening of the co-operation between CEA and Japan and an expanded co-operation between CEA and US DOE on ASTRID, including a contract recently signed with Argonne National Laboratory to study gas power conversion system (Brayton cycle).

Exchange of information and experience with Japan in decommissioning of nuclear facilities

CEA signed, in 2015, an MoU with the Tokyo Electric Power Company (TEPCO) and one with the Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF), both centred on the exchange of information and experience in decommissioning of nuclear facilities.

EPR at Flamanville

The construction of the EPR reactor at Flamanville is progressing with most of the civil structure work now completed. EDF's CEO has announced a new timetable to complete and commission the 1 630 MWe unit by 2018 (instead of 2017).

Japan

Current status of nuclear policy

The agency for Natural Resources and Energy of the Ministry of Economy, Trade and Industry (METI) approved the "Long-term Energy Supply and Demand Outlook" (Long-term Outlook) in July 2015, based on the Strategic Energy Plan approved by the Cabinet in April 2014.

In the Long-term Outlook for 2030, Japan is supposed to achieve an improvement in energy self-sufficiency to around 25% and the reduction of energy costs, as well as GHG reduction with a target in line with those of Europe and the United States by promoting energy conservation, introducing renewable energy as much as possible and improving efficiency in thermal power generation, etc. Specifically, the government intends to reduce the GHG emissions by 26% from the FY2013 level, by achieving a share of 20 to 22% of nuclear energy and 22 to 24% of renewables in the electricity generation mix.

Condition of TEPCO's Fukushima Daiichi nuclear power plant

For fuel assembly and fuel debris removal from units 1, 2 and 3, TEPCO began preparatory works to remove fuel assemblies from the spent fuel pools. Work on dismantling the building cover of unit 1 started on 28 July 2015. In unit 3, the fuel-handling machine, the largest rubble in the spent fuel pool, was removed on 2 August 2015. The fuel removal from units 1 and 3 will start in FY2020 and FY2017 respectively, and the preparation at unit 2 is ongoing. Planned work for removing fuel from the spent fuel pool of unit 4 was completed on 22 December 2014.

In April 2015, the Collaborative Laboratories for Advanced Decommissioning Science was established at the Tokai site of Japan Atomic Energy Agency (JAEA) as a base for R&D on technologies that will bring big breakthroughs in decommissioning and human resources development. The Naraha Remote Technology Development Center of JAEA, a demonstration facility for remote operation devices and apparatuses, started partial operation in Fukushima in September 2015.

Licence Conformity Review of nuclear power plants and nuclear fuel cycle facilities

The life of a reactor is basically limited to 40 years according to the revised Nuclear Reactor Regulation Law. It was announced in March 2015 that five units of four nuclear power stations will be decommissioned. Japan Atomic Power Co. will retire its Tsuruga 1, Kansai Electric Power Co. Mihama 1 and 2, The Chugoku Electric Power Co. Shimane 1, and Kyushu Electric Co. Genkai 1.

Unit 1 at Kyushu Electric Power Co.'s Sendai Nuclear Power Station restarted on 11 August 2015 and began commercial operation on 10 September 2015. Unit 2 restarted on 15 October 2015.

The Nuclear Regulatory Authority (NRA) confirmed on 12 February 2015 that units 3 and 4 at Kansai Electric Power Co.'s Takahama Nuclear Station meets the new regulation for restart and it also confirmed the same for the unit 3 at Shikoku Electric Power Co.'s Ikata nuclear power station on 15 July 2015. The NRA is currently conducting safety reviews for commercial operation under the new regulation for 26 units in 16 nuclear power stations.

Regarding the status of safety reviews of nuclear fuel cycle facilities under the new regulation, safety review is underway for Japan Nuclear Fuel Limited's (JNFL) reprocessing plant and MOX fuel fabrication plant. Currently, measures for preventing recriticality and mitigating the impacts of beyond-design-basis accidents are being inspected based on the "defence-in-depth" philosophy. JNFL announced that the construction of the reprocessing plant will be completed in the first half of 2018 and that of the MOX fabrication plant in the first half of 2019.

Update on the Japan Atomic Energy Agency

In April 2015, JAEA set out a mid- and long-term plan for seven years. It is stated in this plan that JAEA should advance R&D of Monju and plan R&D to establish proven technologies of fast reactors (FRs) and international strategy to achieve maximum results through the co-operation in GIF and also with France for the ASTRID programme. It is also stated that JAEA should conduct R&D for reducing the volume and toxicity of radioactive waste and R&D of high-temperature gas reactors and its heat utilisation technology.

JAEA's report "Safety Requirements Expected to the Prototype Fast Breeder Reactor Monju" was reviewed by FR experts from in and outside of Japan evaluated as appropriate. (This result was issued in September 2015)

With regards to the maintenance management of Monju, JAEA, Monju operator, has been working on the improvement of its maintenance management system and quality assurance system since the deficiencies in the maintenance and management of components was found out. The NRA, however, issued a recommendation to the Minister of Ministry of Education, Culture, Sports, Science and Technology (MEXT) on 13 November 2015 to find a new operator to replace JAEA within about six months. Responding to this recommendation, the MEXT intends to review the way Monju is managed.

As for the Experimental FR Joyo, JAEA successfully retrieved the damaged upper core structure (UCS) and the material testing rig with temperature control (MARICO-2) in 2014. The reinstallation of the devices which were retrieved from the rotating plug was completed in June 2015. In order to apply for permission to restart Joyo, safety improvement measures are under consideration.

As for the high-temperature engineering test reactor (HTTR) system, replacement of neutron source is undergoing and preparation is underway for restart after the application for permission under the new regulation was submitted on 26 November 2014.

Korea

Overview: current status of important nuclear activities in Korea

As of today, nuclear power from 24 reactors accounts for 22.5% of Korea's total power generation (21 716 MW). Korea intends to increase its nuclear share to 29% with ten additional nuclear power plants by 2035. In June 2015, Korea decided to permanently shut down the Kori unit 1; the first nuclear reactor in Korea will be shut down in June 2017, after 37 years of operation.

Meanwhile, after a safety review and a stress test to evaluate the plant response to large-scale natural disasters, Wolsong unit 1, a CANDU reactor, was approved for continued operation until 2022 after its design life of 30 years ended in November 2012.

In June 2015, the Public Engagement Commission on Spent Nuclear Fuel Management (PECOS) presented its "Recommendation to the Government on the Issue of Spent Nuclear Fuel". This recommendation was prepared after a 20-month period, commencing in October 2013, and has gathered the opinions of the citizens and residents living in communities near nuclear power plants. Based on the recommendation, the Korean government will formulate its "Basic Plan for Management of Spent Nuclear Fuel" by the end of this year.

In addition, a disposal facility with a storage capacity of 100 000 drums of low- and intermediate-level radioactive waste (LILW) was completed in September of this year after spending 30 years on selecting the site. It is now possible to safely dispose of radioactive waste produced by both nuclear power plants and radioisotope (RI) industries.

Korea has been developing the System-Integrated Modular Advanced Reactor (SMART), a type of small modular reactor (SMR), since 1997. In March 2015, Korea and the Kingdom of Saudi Arabia signed an MoU for establishing a SMART partnership for the joint development and commercialisation of SMART globally. On 2 September, Korea and the Kingdom of Saudi Arabia also signed the Pre-Project Engineering (PPE) MoU, and both countries will continue to work together on the construction of SMART.

R&D on Gen IV Nuclear Energy System

Currently, an advanced nuclear energy system that couples pyroprocessing and generation IV sodium-cooled fast reactors (SFRs) is in the pipeline for the efficient management and utilisation of spent fuel. Korea is concentrating its R&D resources on VHTR projects and is actively participating in the Gen IV International Forum.

Sodium-cooled fast reactor (SFR)

Korea has been developing a prototype generation IV SFR (PGSFR) design according to the long-term development plan for future nuclear energy systems with the aim to finish construction by 2028. Korea 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 the end of 2015 to have an independent and authorised peer review on the safety of a prototype SFR. For the successful development of the PGSFR design, Korea has been actively engaged in international collaborative research activities. As a part of this effort, Korea has been actively participating in the GIF SFR activities.

A large-scale sodium thermal-hydraulic test programme called the Sodium Test Loop for Safety Simulation and Assessment (STELLA) is being progressed by KAERI. As the first step of the programme, a sodium component test loop called STELLA-1 has been completed, which is used for demonstrating the thermal-hydraulic performance of the major components and their design code verification and validation (V&V). A performance test for the heat exchangers of the decay heat removal system and a performance test for the mechanical sodium pump to assess the hydraulic similarity in STELLA-1 was carried out from 2014 to 2015. The second step of an integral effect test loop called STELLA-2 will be constructed to demonstrate the plant safety and support the design approval for the prototype SFR.

To support the design, R&D activities for design and analysis code V&Vs, the development of metal fuel fabrication technology, and irradiation testing of cladding materials in BOR-60 are being conducted.

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

VHTR development in Korea is primarily dedicated to the generation of hydrogen, which has been dubbed as the fuel of the future and an alternative energy source to replace fossil fuels. Hydrogen production using VHTR in conjunction with thermochemical water splitting does not emit GHGs, unlike conventional liquefied natural gas (LNG) steam-methane reforming. Therefore, hydrogen production using a VHTR is a clean and efficient method for reducing dependence on fossil fuels in Korea. KAERI has been developing a VHTR and nuclear hydrogen key technologies since 2006, targeting the demonstration of nuclear hydrogen by 2030.

The key technology development project, which is the basis of the Gen IV VHTR R&D collaboration, is focused on the development of computational tools, high-temperature experimental technology, a high-temperature material database, tristructural isotropic (TRISO) fuel fabrication, and the hydrogen production process. The in-house design and analyses codes have been developed and are under validation. Korea succeeded in the operation of a high-temperature helium loop (HELP) at a high temperature of above 900°C, and is ready to test the performance of high-temperature heat exchangers. A natural cooling test facility was built to evaluate the passive safety of an air-cooled reactor cavity cooling system (RCCS). Five separate tests were successfully finished. An irradiation test of TRISO particle fuel fabricated by KAERI was completed in the "HANARO" research reactor, and post-irradiation examinations (PIE) is now progressing. In addition, a suphur iodine (SI) hydrogen production test was successfully carried out to produce hydrogen at 50 L/hour, which was maintained for eight hours under pressurised conditions. Thus, the key technology development project is proceeding well.

A number of efforts are under way to launch a Nuclear Hydrogen Development and Demonstration (NHDD) project, which is aimed at the design, construction, and demonstration of a nuclear hydrogen system using a VHTR. With the help of the Korean Nuclear Hydrogen Alliance, consisted of 13 domestic industries, KAERI performed an NHDD system concept study to develop a system concept and work plan with industries, and was finished in 2014. Another project for the VHTR system point-design and pre-feasibility evaluation started in 2015. Its objectives are to generate design data of the stepwise and integrated demonstration plant. Based on the results of the project, KAERI will submit an application for pre-feasibility approval to the government to secure a budget of the NHDD project.

Russia

Nuclear power in Russia

At present, 34 nuclear power units are in operation in Russia, with more than 25 GWe total electric power capacity. In 2015 the Russian NPPs' load factor is greater than 85%.

There are nine power units under construction in Russia, and the third Rostov NPP unit has been commissioned and included in the unified energy system of Russia. At Rostov NPP the forth power unit of VVER-1000 type is under construction and VVER-1200-type units of new project NPP-2006 with improved technical and economic indicators are under construction at Novovoronezhskay NPP2, Leningrad NPP2 and Byelorussia NPP. Now the first in the world floating nuclear power unit is under construction at the Baltic shipyard in Saint Petersburg and its commissioning is expected in 2016. It will be used in Chukot and help to substitute disposal capacity of Bilibino NPP.

Strategy of the State Atomic Energy Corporation Rosatom in the area of innovative reactor technologies

R&D in the area of innovative reactor technologies in Russia are conducted on a broad front and cover five of six advanced reactor technologies developing in the framework of GIF, namely:

- SFR;
- fast reactor with heavy liquid metal coolant;
- supercritical water reactor;
- molten salt reactor;
- fast gas reactor.

Investigations on the last three technologies are carried out on the conceptual level aiming at the long perspective. The main activities are focused on the first two technologies of fast reactors with liquid metal coolant.

The strategy of activities on these technologies is being done 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 objectives of this programme are not just limited to the development of the particular projects of reactor facilities, but also covers both the issues of nuclear fuel cycle closure and the issues of experimental substantiation of innovative projects. This approach to the creation of a new technological platform allows to solve problems of nuclear power of the future.

The first phase of the implementation of this federal programme is being completed, during which projects of a commercial large size sodium-cooled fast reactor BN-1200, the demonstration fast lead-cooled reactor BREST-OD-300, and the multipurpose research fast reactor MBIR with sodium coolant were developed. A facility for the production of MOX fuel for the new generation of fast reactors, including the BN-800 reactor, was set up in Zheleznogorsk.

In the second stage of the Federal Target Programme (for the period 2016-2020) it is planned to construct the demonstration fast lead-cooled reactor BREST-OD-300 and the multipurpose research fast reactor MBIR, to master industrial technology for the production of promising dense nitride fuel, to finish upgrading the fast critical facilities (BFS) in IPPE in Obninsk, and to continue the development of computational codes of a new generation to analyse prospective nuclear designs.

So it is possible to conclude on the successful implementation of the federal programme for the development of new technology platforms for Russian nuclear power. But it also becomes clear that the scale of the FTP tasks shows the need to extend it up to 2030. In this regard, the issues concerning the preparation and adoption of the FTP-2 are now under discussion.

Sodium-cooled fast reactor (SFR)

The development of the projects of large size fast reactor BN-1200 and multipurpose research fast reactor MBIR is based on the significant and successful experience of development and operation of sodium-cooled fast reactors accumulated in Russia over the past 60 years, since the first experimental reactors FR-5, FR-10, BOR-60 and completing power plants with reactors

BN-350, BN-600, BN-800, the total operation life of which exceeds 150 reactor-years (or 35% of global experience in SFR).

There are currently three units with sodium-cooled fast reactors in Russia:

- industrial power unit BN-600 (more than 30 years of operation);
- research reactor BOR-60 (about 46 years of operation);
- industrial power unit BN-800 (at the stage of commissioning).

The design lifetime of BN-600 of 30 years was prolonged for 40 years (until the end of March 2020). It demonstrates stable and reliable operation. In the last year the load factor for BN-600 reached more than 86% which is a maximum over the entire period of its operation. Now work has begun on a further extension of the operation life of BN-600. Preliminary results indicate the technical possibility of lifetime extension of BN-600 to 60 years (up to 2040).

Operation of the BOR-60 reactor up to the end of 2019 has been licensed. Work began on the study of the possibility of extending its operation up to the end of 2020 at the time of commissioning a the new research reactor MBIR.

The power unit 4 of Beloyarsk NPP with BN-800 reactor is under commissioning, first criticality took place on 27 June 2014. Currently, permission was received to conduct the phase of power start-up with implementation of the research programme at partial power levels up to 50%, including the connection of a turbogenerator to the grid at 35% of the nominal power. The duration of the phase of power start-up is 110 days.

In the frame of the FTP, the development of the basic design of large size fast sodium-cooled reactor BN-1200 which meets the requirements of the 4th generation reactor power systems was completed as well as the basic design of multifunctional fast sodium-cooled research reactor MBIR to replace the BOR-60 reactor.

The licence for the construction of the reactor MBIR, in accordance with the Federal Targeted Programme, was granted. MBIR will be put into operation in 2019 in RIAR (Dimitrovgrad), and on 11 September 2015, the first concrete for the foundation of the reactor MBIR was poured. Rosatom plans to create an international research centre centred on MBIR.

The construction of the first Beloyarsk NPP unit with the BN-1200 reactor is under consideration.

Fast reactors with heavy liquid metal coolant

In the frame of the FTP the development of the design of the lead-cooled BREST-OD-300 has been completed, and work is underway to confirm the feasibility of this reactor technology with the use of heavy liquid metal coolant.

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

Activities within GIF

In June of 2015, the State Corporation Rosatom on behalf of the Russian government signed the GIF Framework Agreement Extension. On 15 October 2015, the Addendum to the GIF Project Arrangement on SFR advanced fuel was also signed.

Preparatory work to sign the GIF Project Arrangement on SFR CDBOP is underway at the moment. Proposals to join the GIF Project Arrangement on thermal-hydraulics and safety of SCWR are under preparation. In the framework of GIF Project Arrangement on SFR system integration and assessment it is planned to consider the concept of BN-1200 to meet the requirements of reactor units of the 4th generation.

South Africa

The Department of Energy, directed by the Nuclear Energy Policy of 2008 regarding the nuclear new build programme (NNBP) continues to implement the Integrated Resource Plan (IRP2010-30), an electricity plan of government which envisage nuclear as part of the energy mix. According to the plan, nuclear will constitute 23% (9 600 MWe) of the energy sources by 2030.

South Africa signed inter-governmental agreements with nuclear vendor countries that have expressed an interest in participating in the South Africa nuclear new build programme. Government, made up of various departments and state-owned entities representatives, conducted nuclear vendor parade workshops which provided a platform for the vendor countries to show-case their technology offerings and allowed South African professionals to exchange views with their peers on the nuclear new build programme.

South Africa hosted the IAEA Integrated Nuclear Infrastructure Review Mission in 2013. The Mission Report highlighted good practices and some gaps within South Africa's nuclear infrastructure and made recommendations, suggestions for closing those gaps. South Africa developed an Action Plan to address the recommendations of the IAEA INIR mission and implementation of this plan is work in progress.

In 2014, South Africa hosted the IAEA Emergency Preparedness and Review (EPREV) Mission. The mission aimed at assessing the country's readiness to respond to nuclear and radiological accidents and incidents culminated into a report with recommendations, suggestions and good practices. An Action Plan to address the recommendations of the IAEA EPREV Mission was developed and implementation of this plan is work in progress.

In 2015, South Africa hosted the IAEA Integrated Regulatory Review Service (IRRS) Pre-Mission with the aim of strengthening and enhancing the effectiveness of the national regulatory infrastructure for nuclear, radiation, radioactive waste and transport safety through the evaluation of regulatory technical and policy issues against IAEA safety standards and good international practices. A full IAEA IRRS Mission is planned for December 2016.

South Africa has been safely generating electricity using this technology at the Koeberg Nuclear Power Station (KNPS), situated in the Western Cape, for more than 30 years. KNPS has two French-designed reactor units, each with an original design capacity of 900 MW for an operational life of 40 years. The utility is currently planning for the extension on the operational life of the KNPS reactor units to 60 years, an additional 20 years on each unit.

In view of the planned plant life extension for KNPS, Eskom requested the IAEA Safety Aspect of Long-Term Operation (SALTO) pre-mission for KNPS from 17 to 25 November 2015. The SALTO peer review is mainly aimed at assessing the current status of the plant's programmes for long-term operation and ageing management; identifying existing or potential issues in respect of safe long-term operation and proposing measures to address issues identified.

Following the termination of the PBMR project, South Africa withdrew from the VHTR System Arrangement. The status quo is that the PBMR intellectual property and assets remains under care and maintenance pending further decisions by government.

In September 2015, South Africa as a signatory of the GIF Charter, signed the ten years extension of the Framework Agreement.

Switzerland

General decision of Switzerland about nuclear power future

The Swiss government decided to phase out nuclear energy shortly after the Fukushima accidents, which in practice means that the currently operating four nuclear power plants will not be replaced after they reached the end of their lifetime. According to the Swiss licensing regime, the duration of the remaining operation period is determined by safety considerations. This was largely confirmed by the decisions taken by the parliament during 2014 and 2015.

International initiative of Switzerland

On 9 February 2015, at the initiative of Switzerland, the Vienna Declaration on Nuclear Safety was adopted by the Contracting Parties to the Convention on Nuclear Safety (CNS) at the Diplomatic Conference which took place at the Vienna headquarters of the IAEA.

This short document includes the following three statements:

- New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible (consequences) releases of radionuclides causing long-term off-site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.
- Comprehensive and systematic safety assessments are to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the above objective. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner.
- National requirements and regulations for addressing this objective throughout the lifetime of nuclear power plants are to take into account the relevant IAEA Safety Standards and, as appropriate, other good practices as identified inter alia in the Review Meetings of the CNS.

The document is available on the IAEA website: www.iaea.org/sites/default/files/cns_vienna declaration090215.pdf

Activities within GIF

Switzerland withdrew from the GFR SA and the CD&S PA in November 2015. In the same month, Switerland signed the MoU for MSR. Switzerland remains active in the Materials PMB in VHTR.

United States

Nuclear energy continues to be a vital part of the US "all-of-the-above" energy strategy for a sustainable, clean energy future. In August 2015, President Obama presented his Clean Power Plan that represents a historic step in the fight against climate change. The Plan sets a targeted reduction of carbon dioxide emissions from electricity production at 32% below 2005 levels by 2030. The plan allows new nuclear power and nuclear uprates to be counted in complying with emission goals, thereby allowing nuclear energy to be competitive with other energy sources. The White House Summit on Nuclear Energy held in November 2015 underscored the role that nuclear energy plays in the Administration's clean energy strategy. The summit identified several activities to help provide the nuclear energy community with access to the technical, regulatory, and financial support necessary to move new or advanced reactor designs towards commercialisation.

In the area of water-cooled reactors, the United States remains optimistic about the construction of four Westinghouse AP1000 pressurised water reactors (PWRs) at two sites in Georgia and South Carolina, with all four reactors projected to be completed by 2020. The Tennessee Valley Authority (TVA) Watts Bar unit 2 reactor received its operating licence from the US Nuclear Regulatory Commission (NRC) in October 2015. TVA is in the process of loading fuel and remains on target to begin commercial operations in 2016. When on line, Watts Bar 2 will be the first US reactor to be completed since Watts Bar unit 1 began operating in 1996. In May 2015, the NRC issued the combined licence (COL) to DTE Energy for an economic simplified boiling water reactor (ESBWR) at the Fermi unit 3 site in Michigan. The NRC is currently reviewing Dominion's COL application for an ESBWR reactor at the North Anna unit 3 site in Virginia. Both the ESBWR and the AP1000 participated in the Department of Energy's (DOE) Nuclear Power 2010 programme, which cost-shared the development of the design certification application and first-of-a-kind engineering. Dominion Resources has announced its intention to seek an extension of the operating licence of its Surry plant in Virginia for another 20 years, which would mean a total of up to 80 years of operating this reactor. If granted, this would create a precedent for other operators; a final decision would likely be made by the early part of next decade.

The DOE stands firmly behind SMRs as an emerging technology that can meet the Nation's growing energy demands – including replacing retiring fossil power plants – while providing reliable, affordable low-carbon power. To this end, DOE initiated the SMR Licensing Technical Support (LTS) Program to provide cost-shared financial support for the certification and licensing of innovative designs that improve SMR safety, operations, and economics. Notably among SMR LTS Program participants, NuScale has been making progress towards its certification goal, meeting key project milestones such as completion of critical plant component testing and development of plant safety analyses. NuScale is currently on schedule to submit its design certification application to the NRC in December 2016. NuScale has also partnered with Utah Associated Municipal Power Systems (UAMPS) to license the first NuScale SMR, which may be located at or near the Idaho National Laboratory site. A UAMPS COL application for this project is planned for submittal in 2017 with commercial operation set for late 2023. The TVA is preparing an early site permit (ESP) application using the Clinch River site in Tennessee for a generic SMR using conservative parameters from all US SMR vendors with a target submission date to the NRC of March 2016.

DOE's advanced non-light water reactor programme performs research to develop technologies and subsystems that are critical for advanced concepts. These R&D efforts can be broadly captured in five distinct areas: fast reactors, high-temperature reactors, licensing strategies, advanced studies and generic advanced reactor technologies. Three important initiatives are being conducted. First, at the direction of Congress, DOE has initiated a study to evaluate options for a new advanced test or demonstration reactor. In this study, point designs are being developed which will then be evaluated against the study's evaluation criteria and metrics, and summarised in a report scheduled for completion in April 2016. The study will evaluate a broad range of options and make a recommendation on an implementation strategy for deploying advanced reactor testing and demonstration capabilities.

Second, in regards to licensing efforts, DOE has drafted the advanced reactor design criteria (applicable to most advanced concepts) and design criteria sets tailored specifically to sodium fast reactors and high-temperature gas reactors. These design criteria sets were provided to the NRC in December 2014 and are now under review. The NRC anticipates publishing the final guidance in 2016. Also in regards to licensing, the NRC and DOE hosted a two-day workshop in September 2015 to engage the advanced reactor community to explore options for increased efficiency, from both a technical and regulatory perspective, in the safe development and deployment of innovative non-light water reactor technologies. The workshop included participants from government, industry, national laboratories, and nuclear-related organisations. The attendance of over 300 people reflects an increasing interest in advanced reactors as a potential energy source for future electricity and process heat use.

And thirdly, DOE is continuing efforts, begun in 2012, to seek interactions with industry for the development of its R&D programme. In 2013 and 2014, DOE made nine awards totalling USD 16.5 million to industry to perform cost-shared R&D on advanced reactor technologies. On 31 July 2015, DOE issued a funding opportunity announcement to cost-share with industry to support the further development of advanced reactor concepts, with initial funding being made available in the amount of USD 12.5 million, which includes USD 5 million for research to be conducted at DOE national laboratories.

Another important initiative within DOE involves the development of accident-tolerant fuels, a next generation nuclear fuel with higher performance and greater tolerance for extreme, beyond-design-basis events. These fuels would give operators additional time to respond to unforeseen conditions, such as those experienced at Fukushima Daiichi. The Congressionally mandated programme is framed on a three phase approach from feasibility to qualification to preparation for commercialisation and is executed through strong partnerships with national laboratories, universities, and the nuclear industry. The industrial research teams, led by AREVA, Westinghouse, and General Electric, are conducting irradiations of their proposed fuels at the Idaho National Laboratory (INL) advanced test reactor in support of the ultimate goal of being ready for an industry commercialisation phase by 2022.

In support of the nuclear energy industry's long-term viability, DOE is also working to train the next generation of nuclear engineers and scientists by sponsoring research activities at US universities. In June 2015, DOE made 68 awards totalling more than USD 60 million for nuclear energy research and infrastructure enhancements. This year the programme includes 43 awards for approximately USD 31 million for university-led nuclear energy R&D projects, 4 integrated research projects totalling approximately USD 13 million, and 9 infrastructure support awards for approximately USD 8 million. For the 2016 awards, 645 applications have been received for consideration. The NRC has engaged in similar efforts, awarding USD 15 million in grants to more than 30 academic institutions in 2015 as part of its Integrated University Program (IUP). Awards were made to encourage careers and research in nuclear, mechanical and electrical engineering, health physics, and related fields to meet expected future workforce needs. The NRC has awarded more than USD 138 million since the programme began in 2007. New funding opportunities have been issued for the 2016 IUP grants.

As DOE strives to fulfil President Obama's call for reducing global carbon emissions – tackling the threat of climate change while providing affordable and reliable energy for our country – nuclear energy will be a crucial part of the US energy portfolio, accounting today for more than 60% of carbon-free electricity in the United States.

Argentina

Nuclear power in Argentina

Argentina recognises the potential of nuclear power for providing sustainable and clean energy, and supports the development and deployment of nuclear power plants. The country currently has three pressurised heavy water reactor PHWR NPPs (1 750 MWe) representing 5.4 % of the country's installed capacity.

The Argentinean-designed CAREM 25, a prototype SMR of 27 MWe, is under construction and scheduled to be in operation by 2018.

An additional nuclear installed capacity of about 3 000 MWe is being considered by 2030.

Nuclear energy policy related to generation IV nuclear reactors

The National Atomic Energy Commisson (CNEA) is the institution in charge of advising the national authorities on defining nuclear policy and conducting research and technical development in the nuclear energy area. In order to comply this mission, CNEA issued the Strategic Plan 2010 – 2019.

In the item nuclear power reactors of this document, CNEA defined Strategic Objective 3: Implement a follow-up programme of new generation IV nuclear reactors and their fuel cycles technologies to evaluate and create research and development lines. This objective includes three specific objectives: i) conduct studies and evaluations so as to define the generation IV Argentine major interest's line(s); ii) promote the participation in international programmes through the collaboration in specific projects; and iii) develop experimental facilities.

Argentinean participation in innovative nuclear reactors activities and projects

Argentina joined the Generation IV International Forum from its beginning in 2001 and contributed to the development of the Generation IV Technology Roadmap 2002, although it has not joined any R&D project related to the six nuclear energy systems selected. Despite being currently a non-active member, Argentina implemented a follow-up programme of new generation IV nuclear reactors.

Argentina also joined the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), an IAEA co-ordinated project. As a member state of the project, Argentina participated in the development and application of the INPRO methodology for the assessment of innovative nuclear reactors and fuel cycles, in three collaborative projects, and in the INPRO Dialogue Forum.

The country also recently joined, as an observer, the Technical Working Group on Fast Reactors of the IAEA.

CNEA R&D activities related to generation IV nuclear reactors

CNEA performed a comparative assessment of the six reactor concepts selected within the frame of the Generation IV International Forum, in order to choose the appropriate concept(s) for conducting R&D activities in the future. The study selected the Sodium sodium technology as the proper for CNEA.

CNEA carried out joint activities with the CEA (Alternative Energies and Atomic Energy Commission [CEA] of France) for the training of CNEA staff in the performance of neutronic calculation for fast reactors.

CNEA participates in the IAEA CRP (co-ordinated research project [CRP]) "Sodium Properties and Safe Operation of Experimental Facilities in Support of the Development and Deployment of Sodium-cooled Fast Reactors (NAPRO)".

CNEA education and training activities related to generation IV nuclear reactors

CNEA organised, together with the IAEA, two education and training seminars oriented for CNEA staff. One was dedicated to fast reactors science and technology, and the other to sodium-cooled fast reactors science and technology.

United Kingdom

Energy policy and status of nuclear energy in the United Kingdom

Nuclear power plants currently provide about one-fifth 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 light water reactor generating capacity by the late 2020s. This would supplement and eventually replace current nuclear generation capacity, which consists of advanced gas-cooled and pressurised water reactors, as the present fleet is retired.

The choice of reactor technology in this programme lies with the developer. There are currently three LWR designs being proposed for build: the AREVA EPR, the Toshiba-Westinghouse AP1000 and the Hitachi-GE advanced boiling water reactor (ABWR), each of which are at a different stage in the UK licensing process.

Development of UK Nuclear R&D and Innovation Policy

In January 2014 the UK government established the Nuclear Innovation and Research Advisory Board (NIRAB), with the purpose of advising ministers, government departments and governmental agencies on issues related to nuclear research and innovation in the United Kingdom.

During 2014-15, the NIRAB undertook a review of the nuclear research and innovation capability, portfolio and capacity in the United Kingdom. It identified funding and research necessary for the United Kingdom to keep open its options to increase the contribution of nuclear power plant to its energy supply. This supported the development of a programme of nuclear research and innovation for the United Kingdom, along with underpinning business cases, to support priority policies in energy, industrial and scientific capability.

A new UK government was elected to office in May 2015 and has since considered the evidence provided by this exercise. On the basis of this, funding was announced in the 2015 review of government spending for a five-year programme to reinvigorate the UK's nuclear R&D landscape. The UK plans to undertake a wide ranging programme of work, some of which will underpin the development of reactor technology considered by the Gen IV International Forum. As part of this work, we envisage that 1-2 years of scoping and planning will be necessary to determine the most appropriate approach to advanced reactor technology and fuel cycle research.

Investment in research infrastructure

A number of new national facilities for nuclear R&D were established in the UK during 2015 through capital investment programmes provided by the UK's Department of Energy and Climate Change (DECC):

- The Materials for Innovative Disposition from Advanced Separations (MIDAS) laboratory was established at the University of Sheffield. This is a national user facility, which provides academic, public and private sector organisations with access to state-of-the-art equipment to support research in the management and disposal of radioactive wastes from the nuclear fuel cycle. The facility provides access a suite of state-of-the-art equipment for working with radioactive inventories in the processing and characterisation of materials with application in nuclear waste management and disposal. The MIDAS facility is therefore a unique research capability, providing hands-on user access to state-of-the-art equipment and instrumentation, utilising medium-scale radioactive inventories.

- A new pyroprocessing research laboratory is being established at the University of Edinburgh. This will enhance UK-based current pyroprocessing research capability by providing the equipment and infrastructure required to demonstrate each of the essential components of a fast reactor fuel pyroprocessing recycle technology, with a view to allowing subsequent hot cell testing of these processes. Like MIDAS, it was also operate as a user facility.

- The capabilities of the National Nuclear Fuel Centre of Excellence, hosted jointly by NNL and the University of Manchester, are being extended to facilitate the development of advanced fuel materials and manufacturing processed for accident-tolerant fuels (ATF). This is seen as one of the most promising areas of technology to improve the safety and performance of both existing nuclear power plants, as well as new build plant, in the global arena. These fuels will have the ability to withstand much higher temperatures during fault conditions than currently available products, and hence to eliminate or greatly reduce the radiological impact from events involving loss of coolant or containment. Whilst the ATF programme feeds into wider international efforts to develop these fuels for light water reactors, the ceramic compounds and cladding systems are also relevant to Gen IV systems.

- A new process chemistry laboratory is being established at the University of Lancaster for work on beta and gamma active fission products, uranium, thorium and low-level alpha tracers. This is oriented towards near-term demonstration of a safe, economic, efficient and proliferation-resistant aqueous fuel recycle technology at lab-scale, to enable subsequent hot cell testing on spent fuel. This will cover uranium and thorium-based fuel cycles, hydrometallurgical processing and the interface with pyrochemical reprocessing routes.

- A high-temperature materials testing suite (HTF) is being established to provide open access to facilities for fundamental research on structural materials, which have the potential for use in primary and secondary circuits of future reactor systems. This suite is designed to allow materials research for a wide selection of reactor systems that are currently being considered for development, including both gas (especially helium) and liquid metal-cooled systems (e.g. sodium and lead/bismuth). The facility will increase the

UK's capacity for tensile, creep and fatigue testing in the relevant environments (pressurised gas for VHTR/HTR, liquid metal for SFR/LFR and inert atmospheres) at temperatures up to 1 000°C and with temperature cycling. Detailed analysis of tests will be possible through digital image correlation for full field strain measurement (especially of welding), acoustic emission monitoring and potential difference monitoring for crack initiation and growth.

In 2015, it was announced that the United Kingdom and China will work together to co-fund a GBP 50 million nuclear energy research centre, to be headquartered in the United Kingdom. The Centre will be established and run jointly by the UK's National Nuclear Laboratory and the China National Nuclear Corporation (CNNC) and will incorporate projects on a number of different areas of work across the whole nuclear fuel cycle.

Research into small modular reactors

The Nuclear Industrial Strategy, published in 2013, set out the UK government's interest in the potential benefits offered by SMRs. The UK government also recognises that there may be long-term value in SMR technology, in particular its potential for shorter deployment times and to reduce the costs of nuclear power for energy consumers, as well as presenting a possible area of high value opportunity for UK industry. However, SMRs are in the early stages of development and there are no commercially operational examples that can be used to validate their potential.

To begin building an evidence base to support policy decisions on SMRs, the UK government commissioned a feasibility study in early 2014 to assess the technical, economic and commercial case for the deployment of SMRs in the United Kingdom. The study was published by the National Nuclear Laboratory on 3 December 2014 and indicated that SMRs were potentially deployable within a ten-year time frame and that there were opportunities for collaborations between potential SMR vendors and UK industry. However, the study also concluded that further detailed technical analysis of specific SMR designs would be needed to inform any policy decision by the UK government and support any potential investment decisions.

Building on this, DECC commissioned a techno-economic assessment in the summer of 2015, to deliver the necessary evidence base to inform any future government policy. This study will result in an objective assessment of SMRs against predefined criteria, with a collation of a database of evidence. This is not a competition or process for selection of technology, but an analysis of technical and economic capabilities of the technologies. DECC has encouraged all global SMR vendors to participate, to ensure UK government has as clear a picture as possible of the overall potential for SMRs, including those using Gen IV technology. Designs under consideration cover a wide range of reactor types, including LWR, metal-cooled fast reactors, molten salt reactors and high-temperature gas-cooled reactors.

The object of the study is to enable:

- the collation of a robust data set regarding the SMRs in the global marketplace;
- assessment of the technologies as contributors to a UK energy mix and the potential impact on the UK economy;
- assessment of the potential policy options for SMR deployment in the United Kingdom.

The results of this study are due in the spring of 2016.

Research relevant to Gen IV systems

The National Nuclear Laboratory, private companies and universities in the UK carry out work of relevance to GIF systems, some of which is contributed to the work of the systems steering committees and their project management boards via the 7th Framework Programme (FP7) and Horizon 2020 programmes of Euratom.

The following is a (non-exhaustive) list of the main UK R&D activities which are relevant to GIF systems and which were undertaken in 2015 involving the UK's National Nuclear Laboratory.

Experimental research

- Evaluation of SiC fibre and SiC corrosion, erosion and erosion/corrosion experiments relevant to conditions of gas-cooled fast reactors, undertaken as part of EC's FP7 MATISSE programme by the NNL and UK universities.
- Fabrication of carbide fuel for fast reactors, as part of the EC's FP7 ASGARD programme to research closed fuel cycles. In 2015, the NNL installed production line equipment for uranium carbide fuel. This fuel has now been produced and the project has been examining the effect of process additives, the production of dual density pellets and production fuel containing a minor actinide surrogate.
- Processing of unused carbide fuel and recycle of uranium back into new fuel feedstocks is now close to restarting the Dounreay (SFR) uranium carbide reprocessing operations.
- Development of fabrication techniques for nuclear fuels with the potential for Gen III and IV deployment, including uranium nitride fuels. This project includes investigation into spark plasma sintering techniques.
- Work on high-temperature reactor fuel coatings has been carried in partnership with the University of Manchester to understand the mechanism for fission product migration through the coatings. A thermodynamic model of the mechanism of silver migration has been developed.
- Facilities have been demonstrated that have the capability to work with limited quantities of thorium dioxide for fuel manufacture and then return equipment to a usable state for ongoing uranic operations.
- Work on the development of advanced spent fuel aqueous and pyrochemical recycle processes continues in the United Kingdom through involvement in the Euratom FP7 SACCESS project, through the university-led and Engineering and Physical Sciences Research-Council-funded PACIFIC and REFINE consortia and within NNL funded by a combination of EC and internal IR&D-funded programmes.

Historic materials analysis

- A project is underway to collect historic reports on post-irradiation examination data from the Dounreay prototype fast reactor (PFR) fuel and to identify PFR fuel rods that could be analysed to extend the available historic data.
- NNL has reviewed historic CEA measurements on fission product yields produced in samples irradiated in the Phénix reactor. This work will feed into the JEFF-3.3 fission yield evaluation.

Modelling

- Evaluation of the UK's TRAFFIC fast reactor fuel code and development of a mechanistic modelling approach to augment the US MOOSE-BISON code platform for prediction of oxide fast reactor fuels.
- The fuel cycle modelling code ORION has been used to develop a plausible future scenario for 40 GWe of UK nuclear capacity, which assumes reliance on PWR technology during the majority of this century and a transition to a fast reactor fleet. Results show such a transition is possible even with a more "conservative" cooling time of four years for fast reactor spent fuel. A preliminary decay heat repository model has also been developed, which predicts the peak temperature for a given waste inventory density and repository size. The results corroborate with other studies performed by CEA, which suggest significant reductions in repository footprint.
- Recommendations have been made for modelling for the MOX fuel properties catalogue in the Euratom's FP7 ESNII+ project, based on review of previous work on Pu and O_2 redistribution in fast reactor fuel.

Chapter 3. System reports

This chapter gives an overview of the achievements made in 2015 in the research and development (R&D) activities carried out under the four system arrangements (very-high-temperature reactor [VHTR], sodium-cooled fast reactor [SFR], supercritical-water-cooled reactor [SCWR] and gas-cooled fast reactor [GFR]) and under the two Memoranda of Understanding (MoUs) (lead-cooled fast reactor [LFR] and molten salt reactor [MSR]).

3.1 Gas-cooled fast reactor (GFR)

The GFR cooled by helium is proposed as a longer-term alternative to sodium-cooled fast reactors. This type of innovative nuclear system has several attractive features: the helium coolant is a single phase coolant that is chemically inert, which does not dissociate or become activated, is transparent and whilst the coolant void coefficient is still positive, it is small and dominated by Doppler feedback. The reactor core has a relatively high power density, offering the advantages of improved inspection, simplified coolant handling and low void reactivity effects. The high core outlet temperature above 750°C, typically 800-850°C is an added value to the closed fuel cycle.

The reference concept for GFR is a 2 400 MWth plant operating with a core outlet temperature of 850°C enabling an indirect combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high power density necessary for good neutron economics in a fast reactor core. This represents the biggest challenge in the development of the GFR system. The second significant challenge for GFR is ensuring decay heat removal in all anticipated operational and fault conditions.

A necessary step in the development of a commercial GFR is the establishment of an experimental demonstrator reactor for qualification of the refractory fuel elements and for a full-scale demonstration of the GFR-specific safety systems. This demonstrator will be ALLEGRO; a 75 MWth reactor with the ability to operate with different core configurations starting from a "conventional" core featuring steel-clad MOX fuelled pins through to the GFR all-ceramic fuel elements in the latter stages of operation.

In 2010, research institutes from the Czech Republic, Hungary and Slovak Republic, stepped into the ALLEGRO development, with the aim of creating an ALLEGRO Consortium and hosting the demonstrator in one of these countries. Considering the various difficulties to overcome to succeed in building ALLEGRO, the four organisations – ÚJV Řež, a.s. (Czech Republic), MTA-EK (Hungary), VUJE, a.s. (Slovak Republic) and National Centre of Nuclear Research (NCBJ, Poland) decided to create a legal entity, the "V4G4 Centre of Excellence", which is in charge of the international representation of the ALLEGRO project and of its technical co-ordination. The "V4G4 Centre of Excellence" was formed in 2013 oriented on development, design and construction of ALLEGRO demonstrator – with the aim of hosting the demonstrator in Slovak Republic. The "V4G4 Centre of Excellence" is a legal body registered in Slovak Republic.

The "V4G4 Centre of Excellence" is, at present, in charge of the international representation of the ALLEGRO project and of its technical co-ordination (design, safety, R&D).

The funding is currently provided by national resources, Euratom Framework Programmes and EU Structural Funds. The "ALLEGRO project – Preparatory Phase" was launched by the "V4G4 Centre of Excellence" members in July 2015 with the aim to finish the pre-conceptual phase of V4G4 ALLEGRO by 2020 and the conceptual phase by 2025. As a first step, a roadmap of activities in design and safety was elaborated. The formulation of the following documents related to the V4G4 ALLEGRO is underway:

- Design Specifications and Objectives.
- Safety Requirements and Objectives.
- Roadmap for Research and Development.

R&D objectives related to ALLEGRO

Extensive R&D started after 2001 in France at CEA and continued till 2009, when the GFR programme in France was reduced. The main research challenges for ALLEGRO (and in principle also for GFR2400) have, however, remained still valid and are listed below:

- simultaneous improvement of the robustness and simplification of the decay heat emergency removal systems;
- development of sandwich clad fuel concept including pin encapsulation and irradiation of assembled pins/rods;
- studies related to severe accident behaviour of an all-ceramic core – core degradation mechanisms and radionuclide transport/retention in a gaseous environment;
- high-temperature material qualification and component design and qualification;
- experience feedback and current research relating to the high-temperature gas-cooled reactor (HTR) and VHTR concepts may yield numerous solutions of benefit to the GFR. This applies principally for:
  - development of structural materials suitable for high-temperature operation;
  - thermal insulation technology;
  - helium valve technology (in particular fast acting isolation valves);
  - helium blowers;
  - intermediate heat exchanger and steam generator technology (in particular experience feedback from the VHTR);
  - helium purification technologies;
  - development of high power blowing machines.

Main activities and outcomes of ALLEGRO

The Slovak national programme, ALLEGRO Research Centre was launched in 2014 carried out by VUJE a.s. company. In the framework of the National research project in Slovak Republic, oriented to GFR reactor development, demonstration reactor ALLEGRO CEA 2009 concept (75 MWth) was reanalysed. Concept of ALLEGRO, developed at CEA was studied. Data from CEA institute, GoFastR and other EU projects, concentrated at ESNII+ benchmark, were used as a reference proposal. Results will be easily applicable in future ALLEGRO modifications.

Neutronic analyses were oriented on criticality problems, overall characterisation of neutronic features and core properties, optimisation of reactivity control system and adaptation of suitable macrocode for ALLEGRO core calculations. Well known codes SERPENT, SCALE, HELIOS and DYN3D-MG with various cross-section libraries were utilised at analyses.

Cans for ALLEGRO fuel storage (dry and wet) were proposed. Its subcriticality was proven at all operational situations. ALLEGRO breeding properties were evaluated and improved by partial replacement of PuO_2 by UO_2. Comparing homogeneous and heterogeneous transmutation, better results of heterogeneous placement of Am and Cm – at $MgAl_2O_4$ (spinel) or depleted UO_2 matrix – was pointed out.

The reactivity control system was optimised. Not negligible efficiency improvement was reached by replacement of the B_4C by EuB_6. Dividing of the control and shutdown devices (CSD) control rod group into two parts was initiated by too high differential efficiency and consequently by the necessity to ensure safe asymptotic period of reactor after rod movement. The diverse shutdown devices (DSD) rod group was recognised as not independent for core shut down as CSD support is inevitable. The problem was solved by moving the DSD peripheral part two rows closer to the core centre.

The dynamic macrocode DYN3D-MG was adapted for the ALLEGRO core calculations. A cross-section library in form of interpolation tables (26 energy groups) was prepared using the spectral code HELIOS for key core parts. The DYN3D-MG core calculations showed acceptably low deviations from the results obtained by the SERPENT code. Dynamic modelling of insertion of all control rods into the core gave reasonable results. Serious differences of spectral code results at various criticality and core analyses indicated necessity of both the library and the code testing against existing fast reactor experimental data (e.g. available for sodium-cooled FRs). Critical experiments with potential MOX and UPuC fuel for GFR will be inevitable as well.

The analysis of accidents with fuel damage focusses on specifics of severe accidents, identification of appropriate codes to simulate such events up to the development of initial version of the model of ALLEGRO facility. The model was developed for MELCOR code and analysis proved that the model works properly and revealed basic characteristics of the ALLEGRO relevant to severe accidents. It has been concluded that the analyses performed showed that severe accident at the ALLEGRO facility presents a technical problem within expected scope, which is possible to be analysed and solved using present level of relevant knowledge. The resulting source terms allow an initial estimation of the protection zones of the ALLEGRO research reactor and proposal of appropriate countermeasures to solve severe accident consequences within mitigation of potential serious accidents associated with core fuel damage. As a specific topic of the ALLEGRO concept, the reliability of keeping permanent subcriticality of diverse configurations of the core in severe accident was studied. Based on the analyses, it has been proved that in the case of severe accident at the ALLEGRO reactor, there is significant risk of recriticality of the core assembly, especially in the process of core degradation and relocation. This dangerous state can be prevented by insertion of sufficient mass of absorber into the core prior to its geometry degradation.

VUJE a.s. also performed core melting analyses for ALLEGRO CEA 2009 with the first core using MELCOR code. The source term and severe accident mitigation measures were discussed. Recriticality issues were studied for various configurations of the (molten) core. It was stated that recriticality during the meltdown of the first core might occur. Averting of this situation is expected by insertion of sufficient mass of absorber material into the core prior its geometry degradation.

Behaviour of the guard vessel (partially also of the containment) during normal and accident conditions was also studied (pressure, temperature and composition of the atmosphere) including evaluation of consequences of the external and internal events. In addition, considerations were made about the functionality of the containment ventilation system during severe accidents.

A RELAP5 model of the ALLEGRO CEA 2009 concept was also developed and a limited number of scenarios was analysed. The obtained results were compared with results calculated with the existing CATHARE-2 model. In addition, performance of the decay heat removal system of ALLEGRO was qualitatively studied with the aim to identify relevant requirements for this system.

The research organisation ÚJV Řež, a. s. has concentrated in 2015 onto revision of the fundamental safety characteristics of ALLEGRO, especially onto its coolability in accident conditions using passive systems only (passive mode – without active systems using natural convection only). A model of the ALLEGRO CEA 2009 concept has been developed for MELCOR. Various scenarios including core melting were explored such as the station blackout (SBO) and loss-of-coolant accident (LOCA) with SBO. The safety analyses revealed that the investigated concept of ALLEGRO (first core with MOX fuel in stainless steel tubes) exhibits, in passive mode (e.g. station blackout), the following characteristics:

- Pressurised protected transients: Fuel damage may be prevented by natural convection in the decay heat removal (DHR) system, if the fission products inventory is reduced.
- Depressurised protected transients: The heat removal by the DHR system in passive mode is effective only if the pressure in the guard vessel is high enough (typically min. ~2 MPa) and flow resistance in the system (DHR and core) is minimised. The exploration of a high-pressure resistant guard vessel (e.g. a pre-stressed concrete vault) is highly desirable.

Based on the above-mentioned findings, the current activities focus mainly onto the feasibility of the first core with reduced power and the solution of the coolability of ALLEGRO during LOCA (passive mode).

Unprotected transients are planned to be analysed together with MTA-EK and VUJE a.s. It is expected that passive mode will be unable to avoid melting of the first ALLEGRO core, but more heat resistant cladding materials (either metallic of ceramic) might reduce/remove the risk of the potential fuel meltdown.

The Hungarian National Nuclear Program has been launched in 2015 including activities for ALLEGRO safety and core design. The members of the Hungarian ALLEGRO Consortium are: MTA Centre for Energy Research (MTA-EK), Institute of Nuclear Techniques of TU Budapest (BME NTI) and NUBIKI Ltd. The programme has been aimed at elaboration of the core design and thermal-hydraulic analysis with respect to the reduced power demonstrator.

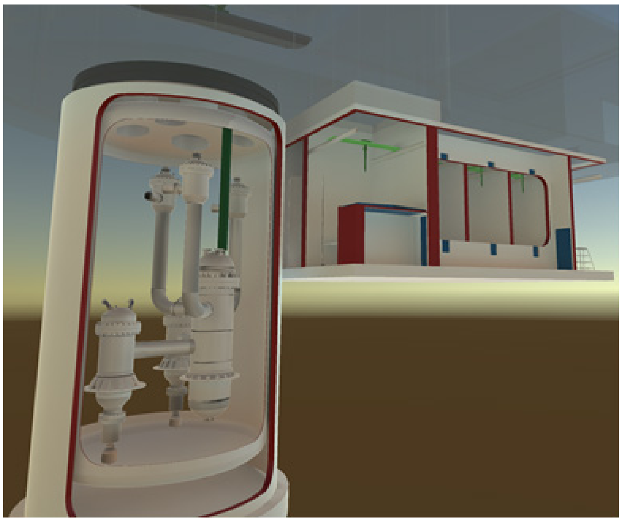
The ALLIANCE project was launched in 2012 by the Euratom with the duration of three years. The project was focused on the preparatory phase for developing the ALLEGRO demonstrator. ALLIANCE covered a number of preliminary studies on fuel management, R&D roadmap and infrastructures needs, and siting, as well as the licensing roadmap, preliminary design and safety analysis. This support action created conditions for efficient exchange of information and co-operation links for improved communication among relevant GFR technology stakeholders, identified and clearly defined the required R&D tasks in terms of experimental programmes and computer codes, and suggested a realistic roadmap for further steps to be taken in order to develop this kind of technology. The main objective of the ALLIANCE project was to put together information on the feasibility of the construction and assessment of design needed following the Gen IV requirements. Though the safety analysis of the 75 MWth ALLEGRO design showed satisfactory features, the lack of passive systems and the low thermal inertia leads to unacceptable results in case of combined LOCA and blackout.

Therefore, a new strategy for the development of V4G4 ALLEGRO has been formulated:

- Feasibility and optimisation of the first core with reduced thermal power aimed at maintaining the ALLEGRO coolable in passive mode in protected depressurised scenarios.
- Increase of the main blowers inertia aimed at avoiding the initial temperature peak during loss-of-cooling scenarios in passive mode (especially during the protected SBO and LOCA and SBO) including the potential development of a turbomachinery concept for secondary circuit (filled with a suitable gas) coupled in a suitable way to the primary blowers. This solution is also advised for the large GFR2400.

- Feasibility and optimisation of the appropriate backup pressure in the guard vessel for the most critical scenarios, especially in LOCA aggravated with SBO in passive mode.
- Solution of potential unprotected transients.
- Development of severe accident mitigation measures in the ALLEGRO design.

Figure 3.1: **ALLEGRO demonstrator**



GIF activites

The GIF R&D activities in GFR resumed in 2015 with 2 meetings of the GFR SSC. Switzerland withdrew from the GFR system arrangement and CD&S project arrangement in November 2015. The system research plan is being updated by France, Japan and Euratom.

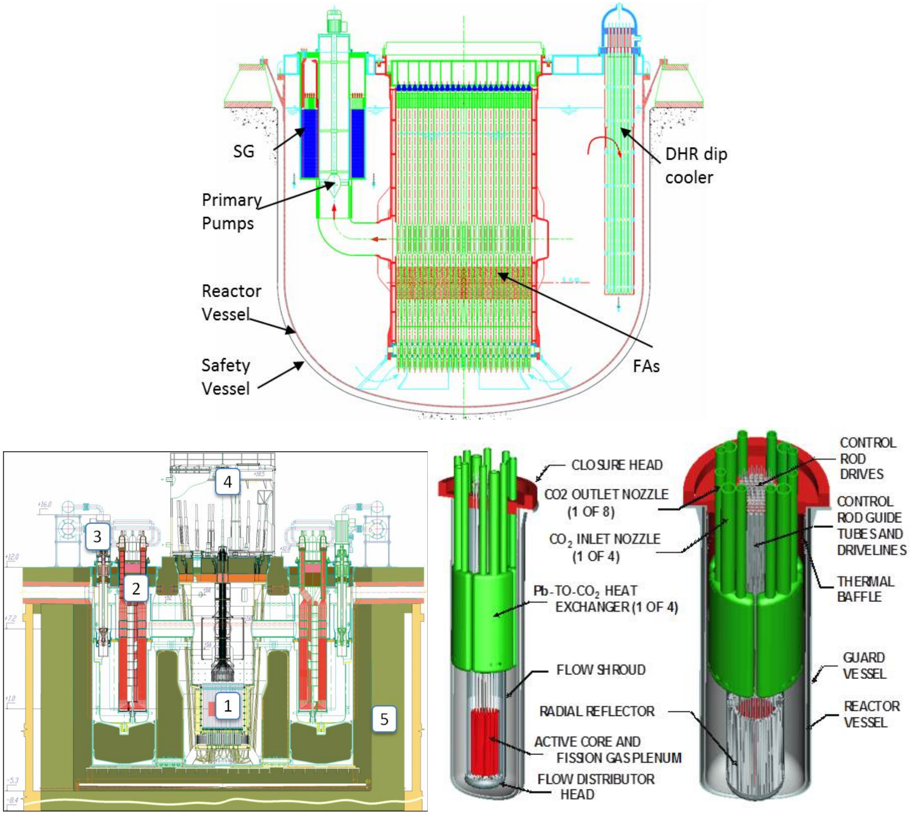
3.2 Lead-cooled fast reactor (LFR)

Main characteristics of the system

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

The system identified by the Generation IV International Forum (GIF) includes three reference systems. The options considered are a large system rated at 600 MWe (ELFR EU), intended for central station power generation, a system of intermediate size (BREST 300 Russia), and a small transportable system of 10-100 MWe size (small, secure transportable autonomous reactor, SSTAR-US) that features a very long core life. The expected secondary cycle efficiency of each of the LFR reference systems is above 42%. It can be noted that the reference concepts for GIF LFR systems cover the full range of power levels, including small, intermediate and large sizes. 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 3.2: **Reference systems of GIF LFR – ELFR, BREST, SSTAR**



1. Core; 2. Steam generator; 3. Pump; 4. Refuelling 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	Supercritical 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

R&D objectives

The system research plan (SRP) for the LFR is based on the use of molten lead as the reference coolant and lead-bismuth as the backup 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 modest primary coolant temperatures and power densities by 2025; and higher-performance reactors by 2040. Following the reformulation of GIF LFR Provisional System Steering Committee (PSSC) in 2012, the SRP was completely revised, and a final draft was prepared by the SSC and sent to the GIF Experts Group for comments. It is expected to be issued in final form in 2016.

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

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 a system arrangement (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 project management boards (PMBs).

System and Component Design project

System design activities are conducted in the following areas: preliminary design of a central station LFR, advancement of the intermediate-size plant design towards early completion, 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 burn-up when compared with that reached in current liquid metal reactors.

In the long term, it is important to confirm the potential for industrial deployment of advanced MA-bearing fuels and the possibility of using fuels that can withstand high temperatures to exploit the advantage of the high boiling temperature of lead in order to increase plant efficiency for electric energy generation and provide the possibility of high-temperature process 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 burn-up) and at high temperature (for increased coolant temperature and power density).

Main activities and outcomes

In 2015, Korea signed the LFR MoU. Signature of the MoU was delegated by the Korean GIF representative to Seoul National University (SNU). SNU was already active as an observer of the LFR GIF activities from the beginning. The current full members (MoU signatories) of the GIF LFR PSSC are: Euratom, Japan, Korea and Russia. The PSSC benefits also from the active participation of its observers: the United States and China.

In 2015, the 17^{th} LFR PSSC meeting was hosted by SNU in Seoul, Korea from 24-26 May. The meeting participants had the pleasure to host Dr John E. Kelly, the Chairman of the Generation IV International Forum. The meeting was as usual dedicated to a worldwide review of technological and organisational aspects in the different participating countries, followed by discussions and working sessions addressing the development of LFR safety design criteria and LFR system safety assessment.

The meeting included a special workshop on small modular reactors (SMRs) organised by SNU. The main objectives of this workshop were to review the worldwide progress in SMR development, project future R&D activities, and promote related information exchange. Following the opening statement of Prof. I. S. Hwang and a welcoming speech by the former Korean congressman Sanghee Rhee, the following seven presentations were given including a Q&A session after each presentation:

- US SMR Development Status and Plan, John Kelly;
- KAERI's SMART R&D, Suhn Choi;

- Lead-cooled SMR of EC, ALFRED, Kamil Tuček;
- Small LFR R&D of China, Minghuang Wang;
- US LFR-SSTAR, Craig Smith;
- LBE-cooled SMR, URANUS, Il Soon Hwang;
- Japanese LFR-based SMR.

The second LFR PSSC meeting, scheduled for 19-20 November at the OECD in Paris was cancelled due to the tragic situation that occurred in Paris on 13 November, especially in consideration of the travelling difficulties of the participants to reach the meeting place. The meeting was rescheduled on 16-17 February at the OECD conference centre in Paris.

The activities of the LFR PSSC during 2015 have been centred on top-level reports for GIF. After the issuance of the LFR White Paper on Safety in collaboration with GIF Risk and Safety Working Group (RSWG) in 2014, the PSSC was very active on four main lines:

- LFR safety design criteria: Development of the LFR safety design criteria (SDC) used the previously-developed SFR SDC report as a starting point. However, it was later realised that the IAEA SSR2/1 (on which SFR SDC was based) did not require many modifications to be adapted for the LFR (note that IAEA SSR2/1 refers substantially to LWR technology). As of the end of 2015, the LFR SDC report had been completed in a final draft form, ready to receive comments by the GIF Reactor Safety Working Group and other working groups.
- LFR system safety assessment: In 2014, the RSWG asked SSCs chairs to develop a report on their systems to analyse them systematically, assess the safety level and identify further safety-related R&D needs. The assessment report was prepared by the LFR PSSC and sent to the RSWG for comments at the end of September. RSWG provided comments in November and the final version is expected to be ready for publication early in 2016.
- LFR PSSC comments to IRSN report on safety of generation IV reactors: In June 2015, the PSSC took the initiative to analyse in detail the above cited IRSN report and provide comments. The committee sincerely appreciated the technically comprehensive review of LFR safety aspects. However, it was also felt that results of recently concluded, as well as ongoing R&D efforts were possibly not available to IRSN when drawing some of the conclusions. The comments provided by the PSSC are expected to be used for further discussions and possible revision of the IRSN report in future. Comments were sent to the newly appointed Technical Director of GIF, Alexander Stanculescu, and to the RSWG inadvance of the Experts Group (EG) and Policy Group (PG) meetings in Saint Petersburg.
- LFR System Research Plan: The LFR SRP was transmitted to the expert group at the beginning of 2015. The LFR System Steering Committee welcomes any comments from the EG to improve the report in order to proceed with the issuance of the document.

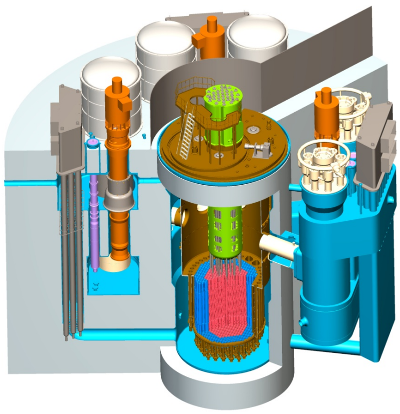
Following the signature in May 2014 of a co-operation agreement between the BREST and LEADER projects, by Nikiet and Ansaldo (on behalf of the LEADER consortium) a first meeting was organised in Genova on 9-11 December 2015. The meeting was attended by four experts on the Russian side (Nikiet) and by a number of participating organisations of the LEADER project. Presentations were made covering both the BREST and ALFRED designs and safety features as well as many specific aspects related to thermal hydraulics, fuel assembly cooling, etc. The meeting was concluded at the ENEA Brasimone laboratory including a visit to the Italian experimental facilities at that site. Several possible fields of collaboration for synergic developments were discussed. Agreements were finalised in the meeting minutes, and the next meeting of the co-operation agreement has been scheduled to take place in the second half of 2016.

Main activities in Russia

An integral review into the innovative reactor technologies of a new generation under consideration in Russia and elsewhere shows that the concept of a fast neutron reactor with a heavy liquid-metal coolant meets higher safety and fuel supply requirements. Under development in Russia is BREST-OD-300, an intrinsically safe pilot demonstration lead-cooled reactor with uranium-plutonium nitride fuel. The BREST-OD-300 reactor is also viewed as a prototype of future commercial BREST-type reactors for large-scale naturally safe nuclear power. Therefore, the selection of the BREST-OD-300's basic designs and performance, including the power level of 700 MW(th), a two-circuit configuration of the heat removal system with subcritical water-steam used as the secondary circuit fluid, as well as of other designs, is dictated not just by the intent to demonstrate the natural safety properties of this reactor technology but also by the requirements for providing the continuity of the fundamental designs for future BREST-type concepts of a higher power.

The development and construction of the BREST-OD-300 reactor is one of the tasks under the energy strategy of Russia for the period up to the year 2030 approved by the Russian government order No. 1715-r, dated 13 November 2009, "Development Strategy of Russian Nuclear Power in the First Half of the 21st Century" approved by the Russian government on 25 May 2000, the Federal Target Program "Nuclear Power Technologies of a New Generation for the period of 2010-2015 and up to the year 2020" approved by the Russian government in 2010, and the Proryv project (2011) that collects projects for the strategic achievement of targets in the formation of naturally safe nuclear power technologies based on fast neutron reactors and a closed nuclear fuel cycle.

Figure 3.3: **BREST-OD-300 primary system configuration**



During 2015, the design of the BREST-OD-300 reactor was developed (as technical design mode) including substantiation on the basis of experiments at small- and medium-scale test facilities, as well as on the results of calculations using verified software. A large amount of integrated computational and experimental activities have been performed to justify the solutions adopted for the design of the reactor core, and the major reactor components and systems, as well as to justify the neutronic and thermal physical characteristics and processes in the reactor.

The neutronic codes MCU-BR and FACT-BR have been verified on the sets of experiments previously performed at the BFS critical installations (five configurations) and the BN-350, BN-600 and Joyo reactors. The first benchmark model of the BREST-OD-300 reactor core with a 500 kg nitride fuel load was created at the BFS-1 stand. The central insert of the model is similar in composition and spectral characteristics to the reference reactor. The E/C ratio was measured for a multiplication factor Keff less than 0.2% $\Delta k/k$, and the standard deviation of the radial and axial fission reaction rate distributions for ^{239}Pu, ^{238}U, ^{235}U was less than 4.0%.

The parameters of gas release from lead into the gas volume at a temperature of 500°C have been measured at the specialised installation for ^{210}Po, ^{124}Sb, ^{110m}Ag, ^{123m}Te, ^{131}I and ^{115}Cd. The obtained data will improve significantly the calculation modelling of radiation conditions (by two orders of magnitude).

The data for the cell-type and computational fluid dynamics (CFD) codes (BEAM-LMC and FLOWVISION) were verified on liquid-metal and aerodynamic stands. Deviations of the hydraulic resistance and heat transfer factors are 10% and 16% respectively. It was shown that taking into account these deviations the maximal temperature of the fuel element cladding does not exceed the design limits.

The computer modelling takes into account the design core parameters with the initial fuel load and subsequent loads over the 30 years of reactor operation within the closed fuel cycle conditions with transition to the equilibrium mode. The basic principles of the equilibrium core mode are confirmed. The maximum reactivity margin at the rated power (including the neptunium effect) was 0.65 βeff, and the stability of the neutron fields was evaluated in terms of the relative change of fuel assembly power over a micro-campaign: <1% (centre) and <3% (periphery). The maximum linear power of the fuel rods was determined to be 420 W/cm (at the centre) and 340 W/m (at the periphery).

An integrated programme is under way for the computational and experimental justification of mixed nitride uranium-plutonium fuel. As the result of the programme, the serviceability of the fuel elements is expected to be justified with respect to the major performance characteristics of the BREST-OD-300 reactor core. By 2015, five experimental fuel assemblies have been fabricated and installed for irradiation in the BOR-60 reactor, seven more EFAs with experimental (U-Pu)N fuel elements have been fabricated and installed for irradiation in the BN-600 reactor, and one fuel assembly has been withdrawn from the BN-600 reactor.

It is planned that, prior to 2020, 15 EFAs will be tested in the BOR-60 reactor and 15 more in the BN-600 reactor for the fuel qualification and licensing.

The shortened heat exchange pipe of the emergency core cooling system (ECCS) model was tested at the experimental setup. The obtained data will be used for verification of codes, and calculation of systems with air field's type heat exchangers. The wall temperature, thermal power, hydraulic resistance versus coolant rate and inlet temperature for the model will be evaluated under conditions close to expected ECCS operating parameters. The experimentally determined net power of one ECCS pipe at a coolant temperature 600°C (~100 kW) is sufficient for operation.

Work has been performed to prove that a single BREST-OD-300 steam generator tube break cannot grow into a multiple break and to obtain experimental data for software verification. The purpose is to justify experimentally BREST-OD-300 safety in the event of the steam generator's heat exchange tubes losing integrity (steam escape into the lead). For the steam generator tubes from EP302-M steel, the allowable temperature and breaking pressure values exceed the levels (including with regard for damageability) expected in the worst cases of severe accident accompanied by a power growth.

An automated control and protection system (ACPS) simulator has been built and integrated tests have been conducted on its basis to support the development of the control and protection system (CPS) detailed design. Developed and perfected video frames of the data display system for the main and backup control rooms have been obtained. Protection and automatic regulation algorithms have been developed and tested, and CPS regulators have been checked for stability of operation in conditions of different transients.

Analysis of reactor safety is fulfilled for the transients during normal operation and for the elaborated scenarios of development of noncompliance of normal operation. It was shown that practically none of the considered initial events of anticipated operational occurrences accompanied by postulated multiple failures of systems and components or personnel errors lead to violations of the unit's safe operating limits. It is possible that the maximum design limit will be exceeded for some of the reactor core fuel elements in the event of the most conservative scenarios of initial events accompanied by multiple failures of systems and components or personnel errors, but no fuel cladding or fuel melting takes place and the circulation circuit remains intact.

Analysis of transient processes in BREST-OD-300 shows a possibility of exclusion of severe accidents which would require evacuation and displacement of inhabitants; these analyses consider first physical properties of the coolant, fuel, other reactor components, and also the technical design, directed at it realisation.

Main activities in Japan

Fundamental studies for the development of LFR were continued primarily in Tokyo Institute of Technology, especially for reactor core design.

Lead or lead-bismuth eutectic (LBE) coolant has a possibility of excellent advantage in burn-up characteristics in fast reactors, because it is possible to make the neutron spectrum hard and the leakage from the core small. These characteristics are expected to make it easy to realise the once-through fuel cycle fast reactor with the principle of a breed-and-burn concept. The CANDLE reactor emphasises burning as one of the ideas in breed-and-burn concepts. Several neutronic analyses were performed for the LBE-cooled CANDLE reactor. It focused especially on a method to maintain the integrity of the fuel elements while undergoing very high burn-up.

The fundamental research results were presented at the American Nuclear Society 2015 Winter Meeting held in Washington, DC in November and at the 2015 Autumn Meeting of the Atomic Energy Society of Japan held in Shizuoka, Japan in September.

Main activities in Korea

Seoul National University has joined the GIF LFR PSSC by signing the MoU in November 2015. The LFR R&D efforts in Korea were university-based during the past 20 years.

Figure 3.4: **Seoul National University, Korea signs LFR PSSC MoU**

William D. Magwood, IV, Director-General, Nuclear Energy Agency, and Professor Il Soon Hwang, Seoul National University.

The Korean LFR programme has two main objectives:

- A technology development requirement for sustainable power generation using energy produced during nuclear waste transmutation.
- A new electricity generation unit development requirement to match the needs of economically competitive distributed power sources for both developed countries and developing nations that need massive and inexpensive electric power with an adequate margin against worst case scenarios encompassing internal and external events.

To meet the first goal, the Proliferation-resistant Environment-friendly Accident-tolerant Continuable-energy Economical Reactor (PEACER) development has recently been initiated to transmute long-lived wastes in spent nuclear fuel into short-lived low- and intermediate-level wastes. The Korean government has selected the SFR as the technology for long-lived waste transmutation. Hence LFR R&D for transmutation has turned its direction towards ADS-driven Th-based transmutation system designated as Thorium Optimised Radioisotope Incineration Arena (TORIA).

For the second goal, Korea has also started to develop the Proliferation-resistant, Accident-tolerant, Self-supported, Capsular and Assured Reactor (PASCAR) for 20-year operation without on-site refuelling. Recently the Korean government has been funding an international collaborative R&D to further develop PASCAR into an improved design called the Ubiquitous, Rugged, Accident-forgiving, Nonproliferating, and Ultra-lasting Sustainer (URANUS).

Proliferation-resistant Environment-friendly Accident-tolerant Continuable-energy Economical Reactor

PEACER is a Pb-Bi-cooled fast reactor being developed at the Nuclear Transmutation Energy Research Center (NUTRECK) of Seoul National University, designed for power production and waste transmutation. PEACER incorporates a pancake-type core with a U-Pu-Zr metallic fuel with a high thermal conductivity in square lattice cooled by forced circulation by a main coolant pump (MCP), and the Rankine cycle for power generation. As with other Pb-Bi-cooled fast reactor concepts, the operating coolant temperature is low spanning 300~400°C to achieve corrosion-resistant conditions and a longer reactor lifetime.

PEACER provides two reactor designs of different capacity. PEACER-550 has a 1 560 MWth core, following the basic integral fast reactor design. PEACER-300 is designed to produce 850 MWth. There is no intermediate heat transport system. The steam at the turbine inlet is superheated to 633.15 K and 8 MPa. The thermal efficiency is estimated to be 35.3%.

PEACER is equipped with an active reactivity control and shutdown system (motor driven) and a passive reactor shutdown system (gravity driven). The active reactivity control and shutdown system consists of 28 control assemblies that are used for power control, burn-up compensation and reactor shutdown. PEACER includes in-house pyroprocess units for spent nuclear fuel recycling under a multinational control, leaving behind low- and intermediate-level wastes to return to the country of origin.

Since 2014, TORIA has been studied as an innovative option to load its core with high fraction of minor actinides mixed with ThO_2 matrix with the assistance of proton cyclotrons. TORIA operating at k-eff of about 0.98, can burn transuranic (TRU) wastes that would be discharged from pyrochemical separation of spent nuclear fuels. Majority of separated TRU wastes are transmuted in multiple units of large-scale SFR in order to allow sustainability of Korea nuclear power fleet. The residual wastes further extracted from the wastes can be transmuted in one unit of TORIA that has less than 100 MW of nuclear power. Ultimate waste from SFR-TORIA symbiosis will be transformed into intermediate-level waste, requiring institutional control period of less than 300 years.

Ubiquitous, Rugged, Accident-forgiving, Nonproliferating, and Ultra-lasting Sustainer (URANUS)

Based on the PEACER design, a small proliferation-resistant transportable power capsules designated as PASCAR has been developed at NUTRECK by capitalising on outstanding natural circulation and chemically stability of lead-bismuth eutectic (LBE) coolant. The PASCAR design employs a pool-type capsule including a core of U-TRU-Zr-alloy fuel rods in open-square lattice and in-vessel steam generators with no pump while enriched uranium dioxide fuel can be used for the near-term applications. Recently the core design has been changed to use fresh enriched UO_2 fuel rods in hexagonal geometry. Like PASCAR design, URANUS is targeted for 20 years of operation without on-site refuelling at electric power up to 100 MW with a Rankine cycle efficiency of 35%. The natural circulation capability, fast load-follow capability, coolant chemistry management technique as well as steam generator tube leak-before-break features are considered to be promising solution to meet the demand for passive safety and security as well as competitive levelised cost of electricity.

URANUS R&D is focused on *i)* three-dimensional neutronic and thermohydraulic analysis code development; *ii)* corrosion-resistant functionally graded composite (FGC) materials production; and *iii)* an integral mock-up test of about t1/200 scale (about 500 kW) using electrical heaters. The mock-up, designated as PILLAR, has been designed and will be built and operated by May 2016.

Main activities in Euratom

Following the signature of the Fostering ALfred CONstruction (FALCON) Consortium Agreement in December 2013 by Ansaldo, ENEA (Italy) and ICN (Romania), the consortium was enlarged by the addition of the CV-ŘEŽ laboratory (Czech Republic) in December 2014. The consortium successfully involved a number of additional European partners through the signature of a number of Memorandum of Agreements (MoA) expanding throughout Europe as much as possible the interest in the development of lead technology. In the frame of the MoAs, all activities are performed on an in-kind basis by the parties. The present additional partners who signed the MoAs and are contributing to technical activities related to technology development and/or ALFRED implementation are: CRS4 (Italy), NRG (Netherlands), SRS (Italy), IIT (Italy), and SYMLOG (France). Further contacts are ongoing with other organisations interested in joining the FALCON and/or signing an MoA.

The main activities related to ALFRED design development in 2015 have been concentrated on the following actions: *i)* for the primary side design a new design configuration is under development; *ii)* newly developed design of steam generators is under CFD evaluation; *iii)* Primary pumps are a subject of sensitivity studies and different possible solutions are analysed; *iv)* a new decay heat removal system has been integrated in the pool; and *v)* optimisation studies of core and fuel assemblies are under way.

Organisational activities of FALCON included in 2015 the following actions: *i)* proposal of development of a distributed research infrastructure for lead technology development (i-CRADLE proposal); *ii)* development of a proposal to be financed by structural funds for the construction in Romania of the largest facility for lead technology (vessel of three metres in diameter and nine metres in height) for the testing of full size components; and finally *iii)* actions are ongoing to promote the ALFRED project as a major project for Romania.

As for technology efforts, ENEA progressed in the development of dedicated instrumentation, namely acoustic sensors for steam bubble detection, oxygen probes for coolant purity assessment and neutron flux detectors. As for materials science, the protective surface development and the low-swelling creep resistant steels were dealt with. Suitable self-passivating alumina-forming surfaces (FeCrAl) and ceramic Al-based coatings were further developed and qualified. A new heat of 20% cold-worked double stabilised austenitic steel (DS 4) was produced by the Vacuum Induction Melting (VIM) technique, and complete mechanical characterisation at 650°C was successfully performed. In the area of core design, the neutron characterisation of the new ALFRED core configuration was carried on as well as the feasibility study for in-vessel storage of ALFRED fuel elements. The fine-tuning and qualification of thermal-hydraulic numerical codes progressed.

Figure 3.5: **Hot rolling DS4 steel**

With respect to MYRRHA, at the beginning of 2015 the FEED contract, awarded in October 2013 to a consortium formed by AREVA, ANSALDO, EMPRESARIOS AGRUPADOS and Grontmij, was suspended. The reason of suspension is that a deep review of the primary system configuration of MYRRHA was needed. SCK•CEN pursued studies throughout the year related to different possible options for the primary side configuration. Major options were selected, and dedicated studies on critical components will be continued in 2016 prior to the relaunching of the FEED contract. SCK•CEN is already carrying out basic studies on lead-bismuth technology development related to corrosion mechanisms, thermal-hydraulics and Polonium behaviour. Interaction with the Belgian Safety Authority resulted in the introduction of a double tube wall steam generator. Different possible solutions are under evaluation at SCK•CEN.

Euratom launched in September 2015 a call for project proposals as a part of the Horizon 2020 Framework Programme for Research and Innovation. Dedicated topics under which proposals are sought in relation to generation IV reactors include those related to safety characteristics and materials research. European organisations are considering several proposals to be submitted, which include activities on lead technology, development of generation IV SMRs, safety system development, and material development and qualification. Through its direct actions conducted by the Joint Research Centre of the European Commission, Euratom also supported the development of the heavy liquid metal experimental facility to conduct pre-normative tests of candidate structural materials for LFRs in temperatures up to 650°C. Commissioning of the facility is expected in 2016.

Main activities in China (observer)

In China, Chinese Academy of Sciences (CAS) had launched a project to develop ADS and lead-based fast reactors technology since 2011. China LEAd-based Reactor (CLEAR) is selected as the reference reactor for ADS and fast reactor system and is being developed by the Institute of Nuclear Energy Safety Technology (INEST/FDS Team), CAS. The programme consists of three stages with the goal of developing 10 MWth lead-based research reactor (CLEAR-I), 100 MWth lead-based engineering demonstration reactor (CLEAR-II) and 1 000 MWth lead-based commercial prototype reactor (CLEAR-III) on each stage. To promote the CLEAR project successfully, INEST deeply involves in the reactor design, reactor safety assessment, design and analysis software development, lead-bismuth experiment loop, key technologies and components R&D activities.

Detailed conceptual design of CLEAR-I has been completed and the engineering design is underway, which has subcritical and critical dual-mode operation capability for validation of ADS transmutation system and LFR technologies. KYLIN series lead-bismuth eutectic (LBE) experimental loops have been constructed. And the R&D activities on structural material corrosion experiments, thermal-hydraulics tests and safety experiments are underway. The key components, including the control rod drive mechanism, refuelling system, fuel assembly, and simulator for principle verification etc., have been fabricated and tested. In order to validate and test the key components and integrated operating technology of lead-based reactor, the lead alloy-cooled non-nuclear reactor CLEAR-S and the lead-based zero power nuclear reactor CLEAR-0 are being constructed, the lead-based virtual reactor CLEAR-V are being developed as well.

In addition, series of innovative concepts for different purposes are being developed to enlarge the application perspective of lead-based reactors, which are not only for ADS and fast reactors but also for other innovative applications, such as CLEAR-SFB for spent fuel burning, CLEAR-Th for thorium utilisation, CLEAR-H for hydrogen production, etc.

Main activities in the United States (observer)

Work on LFR concepts and technology in the United States has been carried out since 1997. In addition to reactor design efforts, past activities included work on lead corrosion and thermal-hydraulic testing at a number of organisations and laboratories, and the development and testing of advanced materials suitable for use in lead or LBE environments. While current LFR activities in the United States are very limited, past and ongoing efforts at national laboratories, universities and the industrial sector demonstrate continued interest in LFR technology.

With regard to design concepts, of particular relevance is the past development of the Small, Secure Transportable Autonomous Reactor (SSTAR), carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other organisations over an extended period of time. SSTAR is an SMR that can supply 20 MWe/45 MWt with a reactor system that is transportable. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life (15-30 years) with cassette refuelling; and an innovative supercritical CO_2 (S-CO_2) Brayton cycle power conversion system. This concept represents one of the three reference designs of the GIF LFR PSSC.

Additional university-related design activities include past work at the University of California on the Encapsulated Nuclear Heat Source (ENHS) and more recent efforts at the University of Alaska and Texas A&M University to design a passively operated lead arctic reactor (POLAR).

In the US industrial sector, ongoing LFR reactor initiatives include the Gen4 Module (G4M) by Gen4 Energy, a new LFR reactor concept identified as LFR-AS (Amphora Shaped) by Hydromine, Inc., and a recently-announced initiative by Westinghouse Corporation to design and commercialise a new advanced LFR system.

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3.3 Molten salt reactor (MSR)

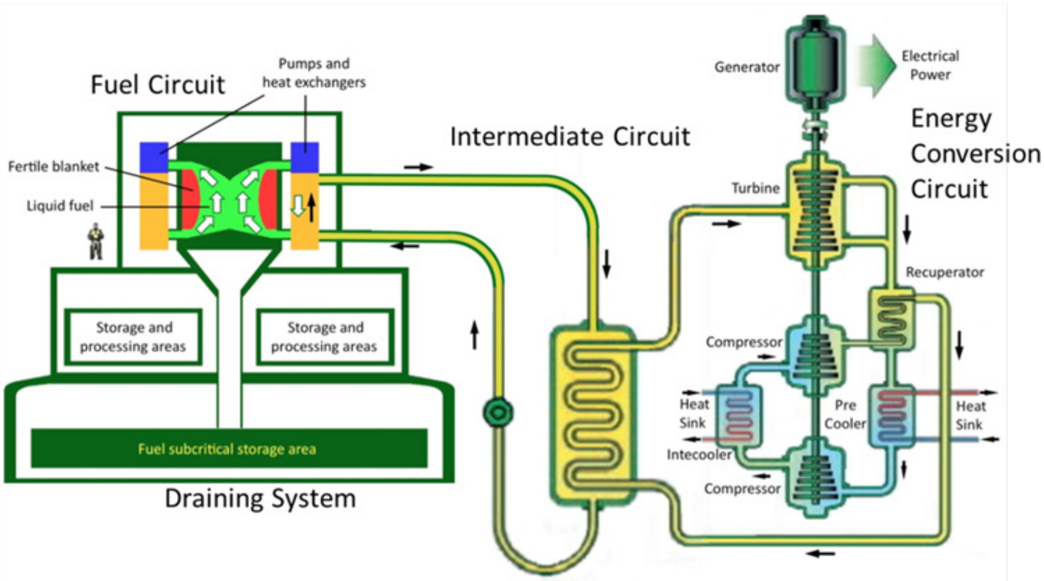
Main characteristics of the system

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 coated particle fuelled core similar to that employed in VHTRs. In order to distinguish the reactor types, the solid fuel variant is typically referred to as a fluoride salt-cooled high-temperature reactor (FHR).

Molten salt fast reactor

In the beginning, MSRs were mainly considered as thermal-neutron-spectrum graphite-moderated reactors. Since 2005, liquid-fuelled MSR R&D has focused on fast spectrum MSR options (see Figure 3.6) 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).

Figure 3.6: **Fast MSR power plant**



Compared to solid-fuelled reactors, fast MSR systems have lower fissile inventories, no radiation damage constraints on attainable fuel burn-up, no reactivity reserve, strongly negative reactivity coefficients, no requirement to fabricate and handle solid fuel, and a homogeneous isotopic fuel composition in the reactor.

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 FHR-Demonstration Reactor (FHR-DR). The FHR-DR is a design concept proposed in the United States for an FHR small-scale technology demonstration reactor. 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.

GIF activities

Within the GIF, research is performed on the first subclass, the molten salt fast reactor concept, under an MoU signed by Euratom, France, Russia and Switzerland in 2015. Two fast spectrum MSR concepts are being studied, large power units based on a liquid circulating fuel: the molten salt fast reactor (MSFR) concept initially developed at CNRS, France and the molten salt actinide recycler and transmuter (MOSART) concept under development in Russia. Simulation studies and conceptual design activities are ongoing in order to verify that fast spectrum MSR systems satisfy the goals of generation IV reactors in terms of sustainability (closed fuel cycle, breeder system), non-proliferation (integrated fuel cycle, multi-recycling of actinides), safety (no reactivity reserve, strongly negative feedback coefficient) and waste management (actinide burning capabilities).

The United States and China, observers in the PSSC of the MSR, are currently working on FHR concepts. Non-GIF member countries as well as multiple private companies in North America and Europe are aslo developing MSR concepts. The work presented here is limited to that performed with government support in GIF member states.

In order to establish interface with Gen IV, the Consultancy Meeting (CM) on “Molten Salt Reactors: Status and possible role for IAEA to facilitate Technology Development” was organised by IAEA in November 2015. The scope of the CM was to present and share important information on the interest and status of technology developments in the area of MSR designs. Based on this information on the general technology status in the world and specific input from the participants of the CM, the NPTD section plans to prepare a larger and all-inclusive Technical Meeting (TM) on MSR Technology in September 2016. This TM will invite all the important role players and interested parties from all member states to come and present the status of their MSR technology and to explore the need for future closer collaboration among member states in the framework of the IAEA. The CM prepared a draft programme for the 2016 meeting as well as created a proposition of member states, institutions and experts to be invited to the meeting.

R&D objectives

Partners of the MSR PSSC are involved in the Euratom-funded Safety Assessment of the Molten Salt Fast Reactor (SAMOFAR) project. This SAMOFAR project is one of the major research and innovation projects in the Horizon 2020 Euratom research programme with a total budget of around EUR 5 million. It started on 1 August 2015 for a period of four years. The grand objective of SAMOFAR is to prove the innovative safety concepts of the MSFR by advanced experimental and numerical techniques.

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;
- perform safety optimisations and studies for fast spectrum MSR;
- 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).

Main activities and outcomes

Transient calculations of the MSFR

Recent calculation results were obtained for a reactor configuration called “reference MSFR” and studied in the frame of the Evaluation and Viability of Liquid Fuel Fast Reactor Systems (EVOL) Euratom project of the Framework Program 7. This is not to be taken as an optimised reactor but as a basis for interdisciplinary studies.

The reference MSFR is a 3 GWth reactor with a total fuel salt volume of 18 m^3, operated at a max fuel salt temperature of 750°C. The system includes three circuits: the fuel circuit, the intermediate circuit and the power conversion circuit. The fuel circuit, defined as the circuit containing the fuel salt during power generation, includes the core cavity, the inlet and outlet pipes, a gas injection system, salt-bubble separators, pumps and fuel heat exchangers.

As shown in Figure 3.7, the fuel salt flows from the bottom to the top of the core cavity (note the absence of in-core solid matter). The total fuel salt volume is distributed half in the core and half in the external part of the fuel circuit, with a circulation time of 3-4 seconds. In preliminary designs developed in relation to neutronics calculations, the core of the MSFR is a single compact cylinder (2.25 m high x 2.25 m diameter) where the nuclear reactions occur within the liquid fluoride salt acting both as fuel and as coolant. Recently, thermal-hydraulic studies performed in the frame of the EVOL project have shown that a torus shaped core (see Figure 3.7) improves thermal flow.

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. 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.

Both contributions to the feedback coefficient: density coefficient (or void, related to the salt thermal expansion) and Doppler coefficient are largely negative, leading to a total feedback coefficient of -8 pcm/K. This is a significant advantage for both the operation and the safety of the reactor as discussed below. The characteristics of the reference MSFR configuration are summarised in Table 3.2.

Figure 3.7: **Schematic conceptual MSFR design, with the fluoride-based fuel salt in green and the fertile blanket salt in red**

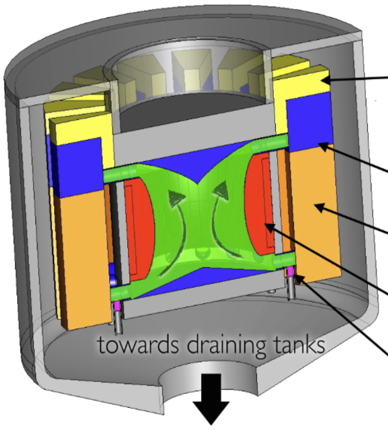


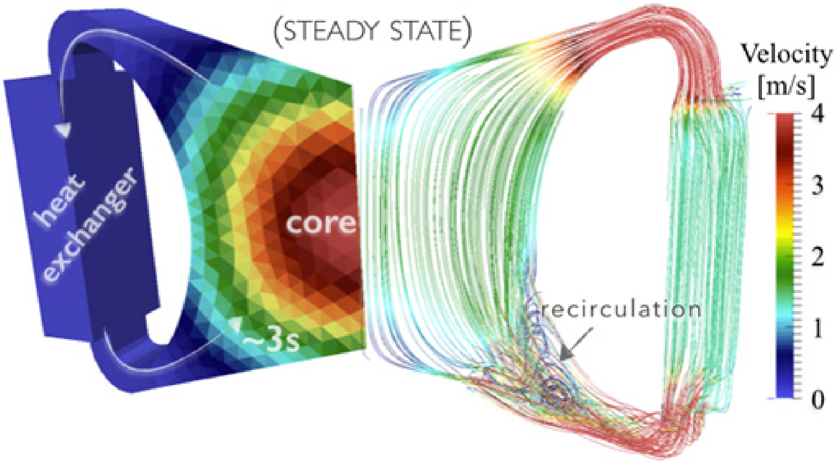
Table 3.2: **Characteristics of the reference MSFR**

Thermal/electric power	3 000 MWth/1 300 MWe
Fuel salt temperature rise in the core (°C)	100
Fuel molten salt – Initial composition	$LiF-ThF_4-(^{233}U$ or $^{enr}U)F_4$ or $LiF-ThF_4-(Pu-MA)F_3$ with 77.5 mol% LiF
Fuel salt melting point (°C)	565
Mean fuel salt temperature (°C)	700
Fuel salt density (g/cm^3)	4.1
Fuel salt dilation coefficient (g.cm^{-3}/°C)	8.82 10^{-4}
Fertile blanket salt – Initial composition (mol%)	$LiF-ThF_4$ (77.5%-22.5%)
Breeding ratio (steady state)	1.1
Total feedback coefficient (pcm/°C)	-8
Core dimensions (m)	Radius: 1.1275 Height: 2.255
Fuel salt volume (m^3)	18
Total fuel salt cycle in the fuel circuit	3.9 s

The MSFR, as a liquid-fuelled reactor, calls for a new definition of its operating procedures. The negative feedback coefficient provides intrinsic reactor stability. The reactor may be driven by the heat extracted, allowing a very promising flexibility for grid load-following for example. Unlike with solid-fuelled reactors, the negative feedback coefficient acts very rapidly since the heat is produced directly in the coolant, the fuel salt itself being cooled in the heat exchangers. This definition and assessment of MSFR operation procedures requires dedicated tools to simulate the reactor's behaviour during normal (e.g. load-following) or incidental (e.g. over-cooling) transients. The reactor modelisation requires specific treatments to take into account the phenomena associated to the liquid fuel circulation.

Classical calculation codes cannot be employed directly because of the specificity of the core cavity's geometry, and because of the precursor motion. The latter and the MSFR thermal feedback effects imply a strong coupling between the neutronics and the thermal-hydraulics during reactor transient calculations. Dedicated tools are thus currently being developed. Coupled to a CFD calculation code, different neutronics models are used, as detailed below: the Transient Fission Matrix (TFM) approach, the diffusion model, or the direct coupling with a Monte Carlo (MC) approach for reference calculations with a reduced computational time. The velocity distribution (Figure 3.8 right) with the stream lines highlights the complex flow pattern in the core of the MSFR, requiring a CFD calculation to capture the vortex and the recirculation at the core inlet. The latter, along with the density distribution, has a significant impact on the neutronic behaviour through the induced variations in the neutron macroscopic cross-sections. Recent studies highlighted the large impact of CFD modelling hypotheses on the MSFR analysis and the need to adopt accurate turbulence models and realistic three-dimensional geometries. In this view, the OpenFOAM® multiphysics toolkit allowed an efficient simulation of steady-state and transient cases on detailed full core 3D geometries.

Figure 3.8: **Distribution of power and velocity in the MSFR at steady state**

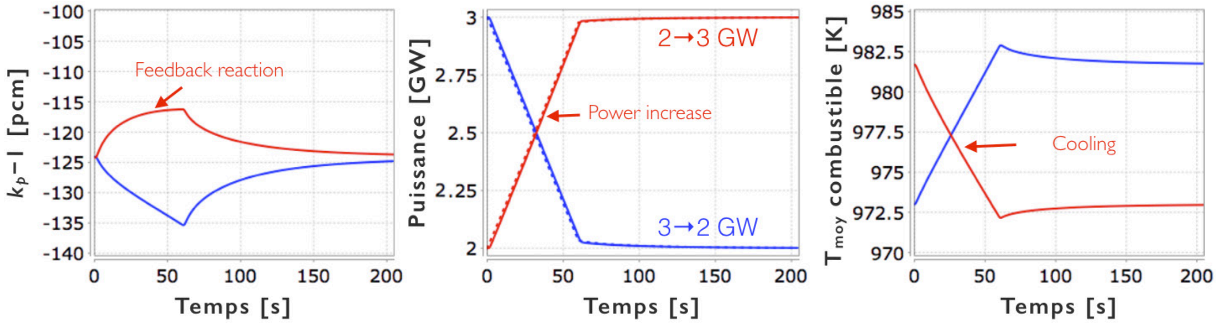


Source: Laureau, 2015c.

Some simplified tools were developed for the modelling of the MSFR neutronics among which tools based on the diffusion approximation of the neutron transport equation. Other tools adopted the finite element, the finite-difference or the finite-volume discretisation of the coupled equations of the CFD/neutronics problem. All these tools proved useful as fast-running options, during the initial MSFR design optimisation phase, in identifying the specifics of the reactor physics of circulating-fuel systems confronted to thermal feedbacks on the neutronics. The TFM approach has been developed specifically as a neutronic model able to take into account the precursor motion associated phenomena and to perform coupled transient calculations with an accuracy close to that of Monte Carlo calculations for the neutronics while maintaining a low computational cost. This approach is based on a pre-calculation of the neutronic reactor response prior to the transient calculation. The results of the SERPENT Monte Carlo code calculations are condensed in fission matrices, keeping the time information. These fission matrices are interpolated to take into account local Doppler and density thermal feedback effects due to temperature variations in the system. With this approach, an estimation of the neutron flux variation for any temperature and precursor distribution in the reactor can be obtained very quickly.

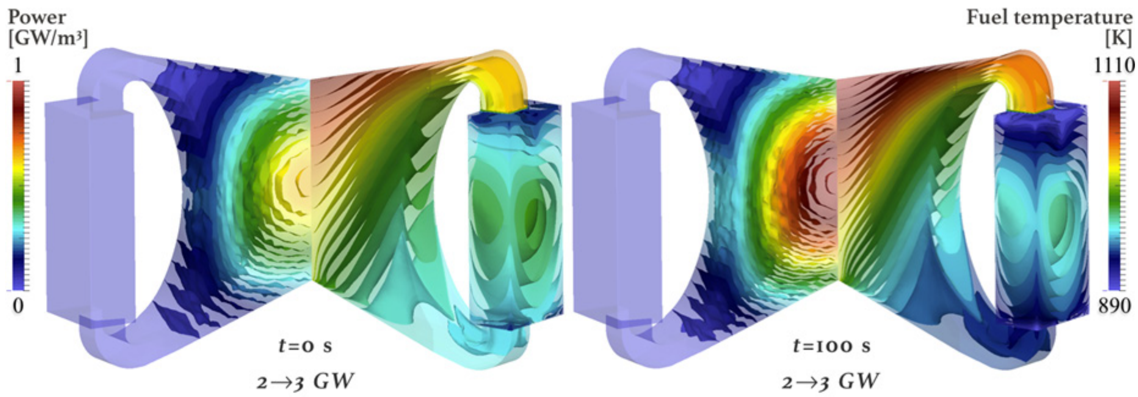
The results obtained with this method applied to a load-following transient are shown in Figures 3.9 and 3.10. The initial condition corresponds to a critical reactor with 2 GWth power (red curves). At the beginning of the simulation, the temperature of the intermediate circuit is reduced to increase the power extracted (in dashed line) up to 3 GWth in one minute. During this first minute and after the load variation, the feedback effect maintains the neutronic power level at the extracted one, the power stabilises to its new level and the reactivity progressively returns to its initial value with a time constant corresponding to the balancing of the delayed neutron precursor population. This illustrates the good behaviour of the reactor on load-following transients for the neutron kinetic and thermal-hydraulics of the fuel circuit point of view. Further studies relative to the heat exchangers are still required to assess this reactor ability to realise such transients.

Figure 3.9: **Load-following transient of the MSFR from an extracted power of 2 GWth to 3 GWth in 1 second computed with the TFM-OpenFOAM® coupled code: evolution of the margin to prompt criticality (left), the power (middle) and the mean fuel salt temperature (right)**



Source: Laureau, 2015a.

Figure 3.10: **Load-following transient of the MSFR from an extracted power of 2 GWth to 3 GWth in 1 second computed with the TFM-OpenFOAM® coupled code: power and fuel temperature distributions at the beginning (left) and at the end (right) of the transient**

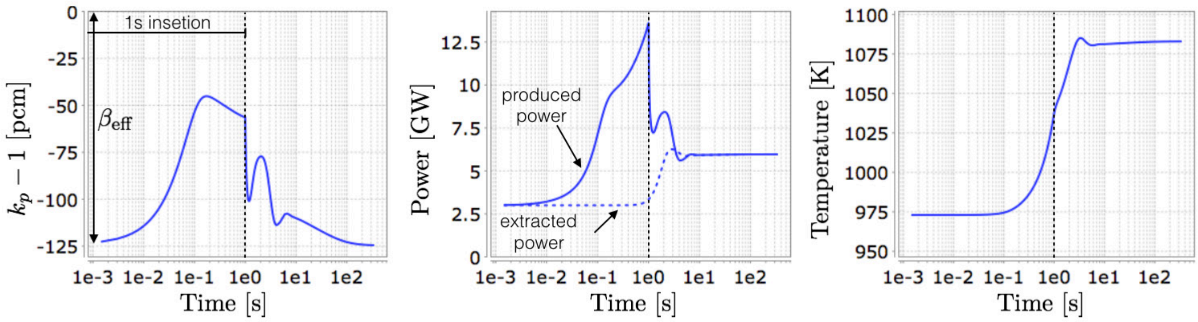


Source: Laureau, 2015a.

As illustrated in Figures 3.11 and 3.12, a reactivity insertion of 1 000 pcm in 1 second has been studied, corresponding to the maximum reactivity margins (ISRN, 2015) with a time constant characteristic of the salt transport between the recirculation loops and the core.

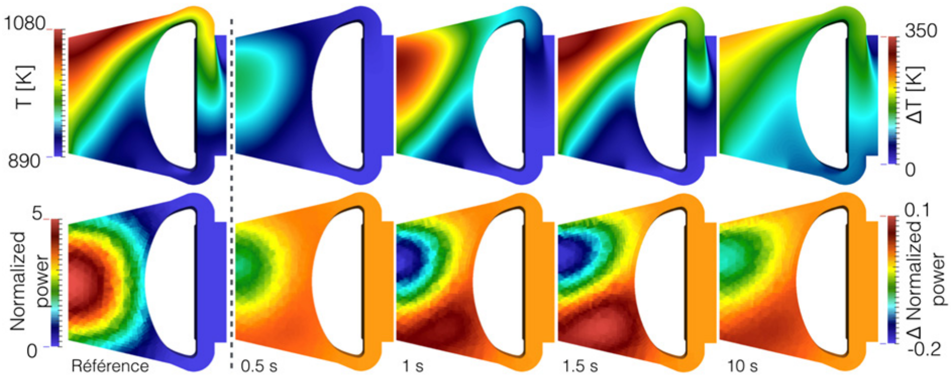
As a conclusion, an adequate and reliable tool coupling neutronics and thermal-hydraulics is now available for the simulations of the MSFR, allowing precise and fast (around one day per full 3D transient calculation) studies. The next steps for these safety and design assessments of the MSFR will take place under the framework of the Horizon2020 European Commission project Safety Assessment of Molten Salt Fast Reactors (SAMOFAR) which started in the second half of 2015 up to 2019.

Figure 3.11: **Reactivity insertion of 1 000 pcm in 1 second in the MSFR, computed with the TFM-OpenFOAM® coupled code: evolution of the margin to prompt criticality (left), the power (middle) and the mean fuel salt temperature (right)**



Source: Laureau, 2015a.

Figure 3.12: **Reactivity insertion of 1 000 pcm in 1 second in the MSFR, computed with the TFM-OpenFOAM® coupled code. Top line: initial fuel salt temperature distribution T(t=0) (left) and its variation ΔT(t) = T(t)– T(0) (right). Bottom line: initial normalised power (left) and its variation due to the flux redistribution in the reactor induced by the temperature redistribution**



Source: Laureau, 2015a.

JRC-ITU contribution to the GIF annual report 2015

Optimisations of methods for synthesis of actinides fluorides for thermodynamic and electrochemical studies in molten salt media continued in 2015. Using the HF gas line (Figure 3.13) connected to a dedicated glove box equipped with a horizontal fluorination reactor very pure UF_4 and ThF_4 were synthetised from their stoichiometric dioxides (see conversion of UO_2 to UF_4 in Figure 3.14) and synthesis of several grams quantities is now possible using just a single fluorination step. The thus established method has been used to synthesise two molten salt fuel compositions for the irradiation under the SALIENT project at NRG Petten. The exact compositions of the two samples were following:

- composition 1: ^7LiF – ThF_4 (78.0-22.0 mol%);
- composition 2: ^7LiF – BeF_2 – UF_4 (71.7-16.0-12.3 mol%).

Both samples have been prepared by direct mixing of their end-members which have been prior hand synthesised (case of ^7LiF, ThF_4 and UF_4) or purified in case of commercially obtained BeF_2. Their proof of purity was made by XRD diffraction and by melting point determination using a conventional differential scanning calorimetry (DSC). In total nine graphite crucibles have been filled, four of composition 1 and five of composition 2 and in some of the crucibles specific alloy has been inserted to address the corrosion performance under irradiation. The samples have been delivered to NRG Petten and are planned to be irradiated starting in 2016 for the time period of one to two years.

Figure 3.13: **Experimental equipment installed in ITU: from left to right – an argon glove box, HF gas installation, electrolyser and fluorination reactor**

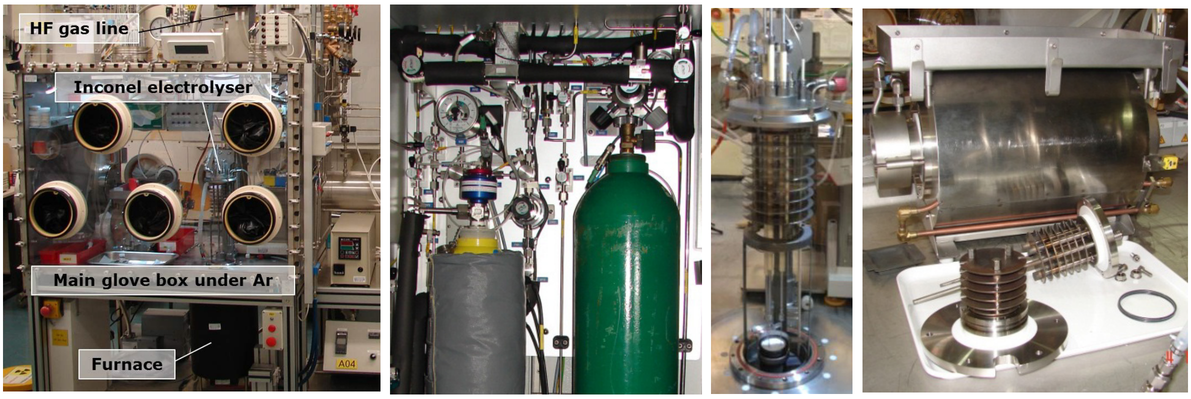
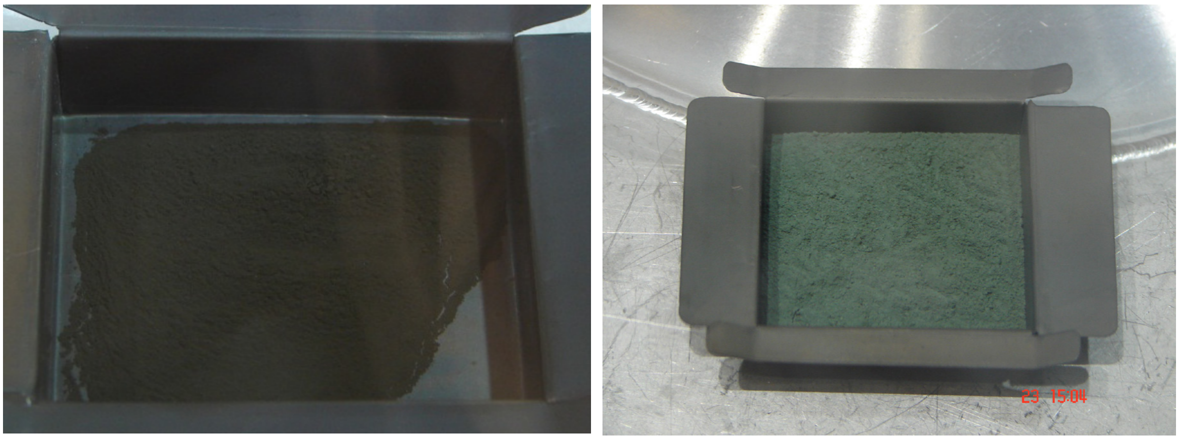


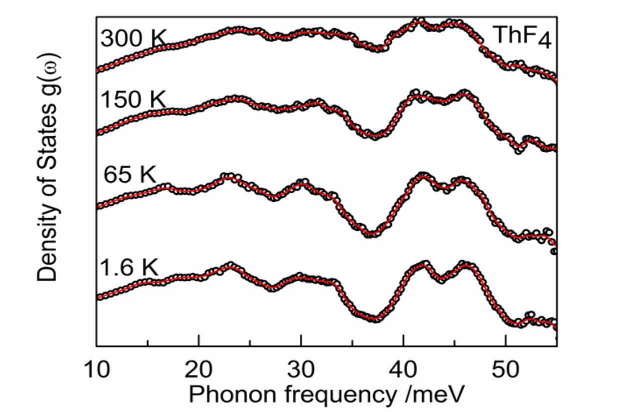
Figure 3.14: **Initial material (left) and final product (right) from UF_4 synthesis by HF fluorination of UO_2 within the SALIENT project**



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 focusing of novel phase equilibrium data and heat capacity data of binary salts. Using the Knudsen cell mass spectrometry the caesium and iodine release from LiF-ThF_4 and from FLiNaK solvents has been measured confirming retention capacity of both caesium and iodine fission products once dissolved in molten fluoride media.

Using a neutron scattering technique a direct observation of the lattice dynamics of ThF_4, UF_4 and UF_3 at very low temperature has been provided for the first time (Figure 3.15). Density of states of these compounds was measured and calculations are ongoing.

Figure 3.15: Measured experimental density of states of ThF_4 as a function of temperature



SAMOFAR European project (A Paradigm Shift in Nuclear Reactor Safety with the Molten Salt Fast Reactor)

Based on the outcome of the previous studies like the EVOL European project, the MSFR is now ready to move forward to technology readiness (TR) level 3 by providing the experimental and numerical proof of concept of the key technologies contributing to the Safety and Reliability of the MSFR.

To this end, SAMOFAR will focus on the safety assessment of the MSFR and will make significant progress beyond the state-of-the-art by:

- Developing a new integral safety approach specific for liquid fuel reactors.
- Extending the database on safety-related properties of the fuel salt by advanced experimental setups, which need to be developed in the project as well.
- Demonstrating experimentally and numerically the proof of concept of two key safety features of the MSFR important for fail-safe decay heat removal: the freeze plug concept to drain the fuel salt in fail-safe storage tanks, and the natural circulation dynamics of the internally heated fuel salt in the primary vessel and in the drain tanks.
- Providing a thorough assessment of transients including uncertainty quantification by the most advanced multiphysics computation tools for liquid fuel reactors.
- Validating experimentally the reductive extraction processes between LiF-ThF_4 and Bi-Li to separate the minor actinides and the lanthanides from the fuel salt.
- Designing the various stages in the chemical plant and evaluating the nuclide inventories at each stage, followed by a thorough assessment of the criticality and shielding requirements.
- Studying the interaction between the (simulated) fuel salt and structural materials at high temperature, including the benefits of zinc-oxide liners.

MOSART activities

Rosatom supported MSR activities continue to be limited to the subclass of 2 400 MWth MOSART design without and with U-Th support. It includes the following R&D needs:

- A preliminary assessment of the MOSART safety performance for the fuel circuit and fuel salt processing unit.
- Experimental study on fuel/coolant salts physical and chemical properties required for safety analysis.

- Experimental verification of the melt chemistry control and compatibility of container material with fuel/coolant salts.

No new significant projects on MSR designs development under the agreement with Rosatom were carried out in 2015.

The NEA Expert Group on Integral Experiments for Minor Actinide Management (EGIEMAM-II) is performing in 2015-2016 a benchmark on sensitivity and uncertainty analyses to evaluate the impact of the MA nuclear data uncertainty on selected integral parameters in the 2 400 MWth MOSART critical reactor with fertile-free fuel (see Figure 3.16) under development in Russia. The main parameters of the core model are summarised in Table 3.3. Additional information on the above-mentioned benchmark is available on the NEA EGIEMAM-II web page.

Figure 3.16: **Layout of the 2 400 MWth MOSART critical reactor with fertile-free fuel**

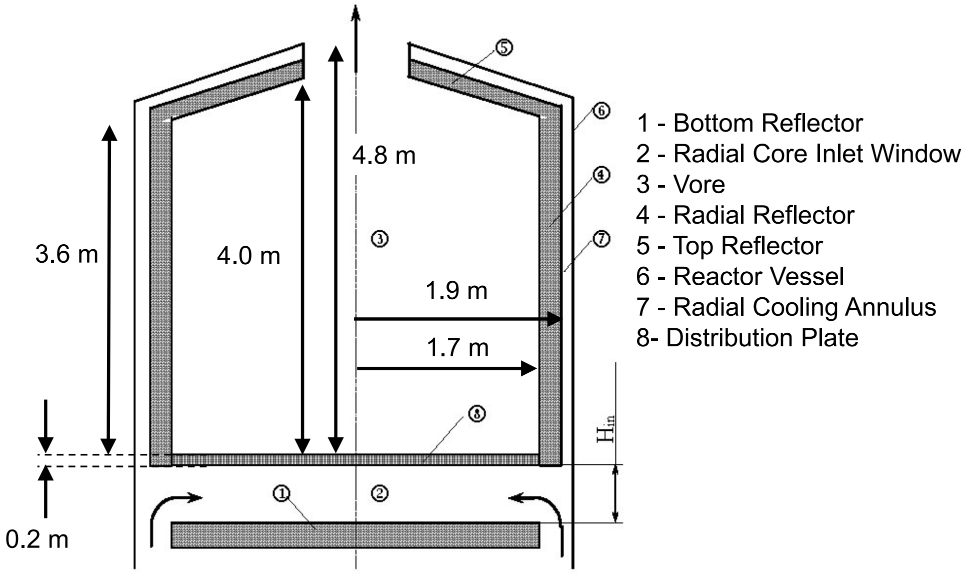


Table 3.3: **Main parameters of the MOSART TRU burner model at nominal condition**

Power [MWth]/[MWe]	2 400/1 100	
Fuel solvent system, mol.%	^{58}NaF-^{15}LiF-$^{27}BeF_2$	
Fuel (PuF_3 and MAF_3)	UOX/MOX spent fuel (45 MWd/kg reprocessed after 30 years)	
Solubility of trifluorides at minimal temperature in primary circuit (mol.%)	~2	
Core volume [m^3]	~32	
Fuel salt average T [K]	900	
Reflector operative T [K]	Graphite	950
	HN80MTY alloy	1 073
Fuel salt flow rate [kg/s]	10 000	
Fuel salt inlet/outlet temperature [K]	873/983	
Core circulation time, s	3.94	
Out-core circulation time, s	6.99	

During year 2015, the following important experimental R&D results concerning MOSART design development were obtained in Russia.

Determination of actinide elements and salt – soluble fission products

In order to ensure the safe and efficient MOSART operation, it will be necessary to maintain adequate surveillance of the various reactor streams. Ideally, all such analyses would be performed automatically with transducers located in the salt streams, since analysis of discrete samples in hot cells is subject to unavoidable delays and is expensive.

Determinations which appear to be of most significance include the redox condition of the fuel, corrosion product ions, oxide, bismuth, hydrogen and tritium. The accurate determination of total uranium, protactinium, TRU and soluble fission products in the fuel salt would also be quite useful.

During the operation of MOSART diverse TRU and electropositive fission products will present in fluoride streams of the fuel and processing systems sufficient for measurement or detection by different techniques.

In previous studies individual and joint solubilities for actinides and lanthanides fluorides in the LiF-NaF-KF and LiF-BeF_2-ThF_4 melts were measured by the method of isothermal saturation with subsequent sampling and chemical analysis. The technique developed provides reliable determination of equilibrium in the system under consideration and measurement with relative error less than 10%.

In year 2015 concentrations of Pr^{3+} ion have been estimated during in-line measurements by reflectance spectroscopy and electroanalytical methods in molten alkali metal fluorides in the temperature range from 773 K up to 1 023 K.

Table 3.4: **Determination of the PrF_3 solubility in LiF-NaF-KF eutectics by different methods**

Temperature, K	PrF_3, mol. %		
	Sampling	Electrochemical method	Reflectance spectroscopy
773	8.9	10.1	13.3
823	13.4	13.7	17.7
873	19.0	18.9	22.2
923	26.6	34.2	26.7
973	36.2	40.8	31.2
1 023	45.3	48.3	35.6
Equation	lgS = 3.8731-2 261.2/T	lgS = 3.43-1 889/T	lgS = 2.87-1 347/T

Electroanalytical studies performed are based on normal differential pulse and cyclic voltammetry. The concentration of Pr^{3+} and Am^{3+} ions in the melt was determined by atomic emission spectrometer on the optical Optima 4200 DV inductively coupled plasma. It was found that the time to establish equilibrium between solid and liquid phases is 3-5 hours at 873 K.

Electronic spectroscopy provides information on the valence forms of rare earth ions, their co-ordination numbers, and solubility of rare earth compounds in molten media. For electronic transitions in the series LiF → NaF → KF → CsF, a long wavelength shift is observed (nephelauxetic effect). The intensity of the hypersensitive transition 3H4 → 3P2 ($\Delta J = 2$) depends on the covalence of the bonds in PrF_{36} – complex species and on their symmetry. The intensity of non-hyper sensitive transitions depends only on the symmetry of complex groups.

Comparison of the experimental data on the PrF_3 solubility in LiF-NaF-KF eutectics by sampling and different in-line methods is given in Table 3.4. In order to ensure reliable determination of actinide elements and salt-soluble fission products in the stream this work should be continued in near future at different ions concentrations truly below its solubility in the fuel/coolant salts.

Equilibrium distribution of samarium and europium between fluoride salt melts and liquid bismuth

The extraction of samarium and europium from the melt of a molar composition $73LiF–27BeF_2$ into liquid bismuth with additions of lithium as a reducing agent at a temperature of 600–610°C was studied. The equilibrium distribution coefficients of samarium and europium were measured.

It was shown that D(Sm, Eu) = 2Log D(Li) + K'(Sm, Eu), where K'(Sm) ≈ 3.65, and K'(Eu) ≈ 3.15 (see Table 3.5). If we take into account the effect of the mole fraction of LiF in the salt melt, the experimental distribution coefficients of samarium and europium correlate with the data of Ferris and Grimes for the molten $66LiF–34BeF_2$ (mol. %) mixture.

The influence of the LiF/BeF_2 ratio on the metal distribution coefficients is explained by the significant dependence of the activity coefficients of certain types of lanthanide and actinide fluoride salts on the molar concentration of LiF in the $LiF–BeF_2$ melt as a result of their strong interaction with the solvent and by the formation of complex ions in the salt phase due to the strong polarising ability of tri- and tetravalent elements. The reduction of the mole fraction of LiF in the $LiF–BeF_2$ melt makes it possible to considerably increase the efficiency of its purification from lanthanides in the system.

Table 3.5: **Dependence of the distribution coefficients of samarium D(Sm) and europium D(Eu) on the lithium distribution coefficient D(Li) for salt melts of different compositions at a fixed temperature**

Ref.	Molten salt, mol.%	n	T, °C	LgD(M) = nLgD(Li) + K'
18, 19	$66LiF-34BeF_2$	2	600	LgD(Eu) = 2LgD(Li) + 3.861
20	$66LiF-34BeF_2$ $66LiF-34BeF_2$	2 2	600	LgD(Sm) ≈ 2LgD(Li) + 4.29 LgD(Eu) ≈ 2LgD(Li) + 3.56
18, 19	$69.2LiF-19.4BeF_2-11.4ThF_4$	2	600	LgD(Eu) = 2LgD(Li) + 3.739
18, 19	$75LiF–13BeF_2-12ThF_4$	2	600	LgD(Eu) = 2LgD(Li) + 3.650
18, 19	$80.5LiF-6.1BeF_2-13.4ThF_4$	3	700	LgD(Sm) = 3LgD(Li) + 5.342
17	$73LiF-27BeF_2$	2 2	609	LgD(Sm) = 2LgD(Li)+ 3.65 LgD(Eu) = 2LgD(Li)+3.15

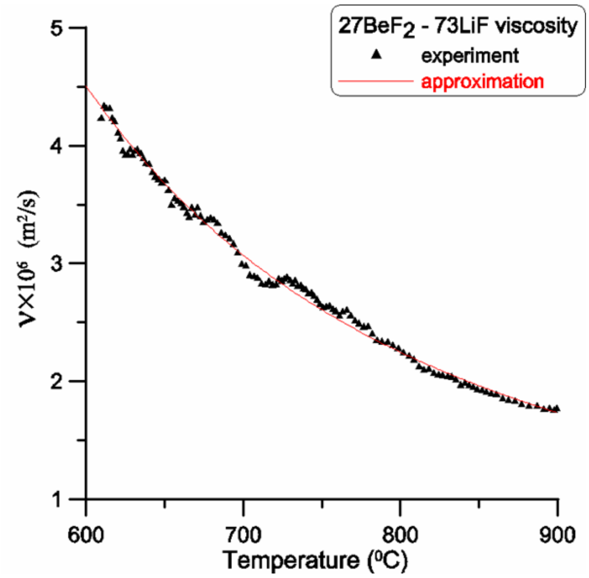
Viscosity and liquidus temperature for the molten salt mixture

As applied to MOSART design operating without Th-U support the viscosity of the molten $73LiF-27BeF_2$ salt mixture has been measured at the temperature ranging from liquidus up to 900°C by the method of torsional oscillations attenuation of the cylinder with the melt under study.

The dependence of kinematic viscosity (υ in 10^{-6} m^2/s) vs. temperature (T in °C) for molten $73LiF-27BeF_2$ salt mixture is given in Figure 3.17. 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 addition on viscosity was also studied. The presence of CeF_3 in the molten salt mixture decreased its viscosity at the cold leg of the measured temperature range.

Figure 3.17: **Kinematic viscosity of the molten 73LiF-27BeF₂ salt mixture**



Materials compatibility and salt chemistry control

Recent study with molten 73LiF-27BeF$_2$ salt mixture (mole%) fuelled by 2 mole% of UF$_4$ and containing additives of metallic Te included 250 hours tests with exposure of Ni-based alloys specimens at temperatures up to 800°C without mechanical loading. 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. Compositions of the Ni-Mo alloys used in this testing are given in the Table 3.6.

Table 3.6: **Ni-Mo alloys under testing (in mass%)**

Alloy	N	HN80MTY	06	12	16	22	29	30	32	34	36
Cr	7.5	6.8	5.1	7.0	7.0	7.5	7.1	7.1	7.1	5.0	7.1
Mo	16.3	13.2	12.3	12.3	12.3	13.2	11.8	12.2	12.1	12.1	12.1
Ti	0.26	0.93	0.63	—	1.82	1.71	0.56	0.56	0.57	0.95	0.94
Fe	3.97	0.15	<0.33	<0.33	<0.33	0.18	<0.33	<0.33	<0.33	<0.33	<0.33
Mn	0.52	0.013	<0.1	<0.1	<0.1	0.013	<0.1	<0.1	<0.1	<0.1	<0.1
Nb	—	0.01	—	0.96	—	0.98	1.0	1.0	1.0	—	—
Re	—	—	—	—	—	—	—	1.08	—	—	—
Y	—	—	—	—	—	0.01	—	—	0.001	—	—
Si	0.5	0.040	≤0.05	≤0.05	≤0.05	0.053	≤0.05	≤0.05	≤0.05	≤0.05	≤0.05
Al	0.26	1.12	2.39	—	—	0.015	—	—	—	1.5	1.6
W	0.06	0.072	—	—	2.20	—	—	—	—	—	—
C	0.05	0.025	0.006	0.005	0.005	0.003	0.007	0.021	0.004	0.006	0.016

After alloy N exposure without stress at 760°C in LiF-BeF_2-UF_4 melt with [U(IV)/]/[(III)] = 60 a significant Te intergranular corrosion (IGC) was observed. For the fuel salt with [U(IV)/]/[(III)] ratio = 90 at 800°C the tellurium IGC for the HN80MTY alloy (the k parameter) was by about ten times lower as compared to reference alloy N. The studies have shown that the IGC in Ni-Mo alloys 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 minimise drastically the Te intergranular corrosion. New findings in developments of Ni-Mo alloys for MSR with fuel salt temperatures up to 750-800°C finally shift the emphasis from alloys modified with titanium and rare earths to those modified with niobium and aluminium.

FHR-related activities

The US government supported 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. Two large university programmes are currenlty underway. One is a partnership between the Massachusetts Institute of Technology (MIT), the University of California at Berkeley, the University of Wisconsin, and the University of New Mexico. The other is a partnership between the Georgia Insitute of Technology (GaTech), the Ohio State University (OSU), and Texas A&M University. Both project teams also include industry and international partners. The university projects are focused on resloving the technology issues necessary for FHRs to be deployed. Results from the previous MIT led university project were published as a series of MIT technical reports.

During 2015, the United States also began evaluating which reactor technology would next be built as either a test or demonstration reactor. An FHR technology demonstration reactor is being considered as one of the candidate technologies. National laboratory led efforts in 2015 have concentrated on evaluating the requirements and characteristics for such an FHR-DR.

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. Significant work under the co-operative agreement began in 2015. Under the agreement ORNL is commissioning a forced convection liquid salt loop, investigating the principles of liquid salt pump design, developing a molten salt flowmeter calibration test stand, adapting the SCALE reactor physics modelling toolset for FHRs, developing safety and licensing training materials for the Chinese nuclear safety authorities, and working with SINAP to evaluate the safety and performance characteristics of SINAP's initial candidate solid-fuelled MSR. SINAP is also sponsoring the University of California to provide training on its surrogate material liquid salt thermal-hydraulic test facility and its pebble bed core X-ray examination facility.

Development of 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. Also an ANS working group to develop a standard on the design safety of liquid-fuelled MSRs was organised in late 2015.

Additional information on MSR technologies is available on ORNL MSR web pages: www.ornl.gov/msr. During October of 2015 ORNL hosted an MSR technology workshop that featured a commemoration of the 50th anniversary of the Molten Salt Reactor Experiment (MSRE). The workshop notably included participation from the nuclear power industry, eight propective molten salt reactor vendor companies, nuclear safety authoriities, universities, and national laboratories from around the world.

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3.4 Supercritical-water-cooled reactor (SCWR)

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, then by a Euratom partnership, more recently by China, 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 PMBs within the SCWR System: *i)* System Integration and Assessment (provisional); *ii)* Materials and Chemistry; *iii)* Thermal-hydraulics and Safety; and *iv)* Fuel Qualification Testing (provisional). China signed the SCWR System Arrangement in 2014. The projects plans for the Thermal-Hydraulics and Safety (TH&S) project as well as for the Materials and Chemistry (M&C) project have been updated and include the planned contributions of each member.

R&D objectives

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

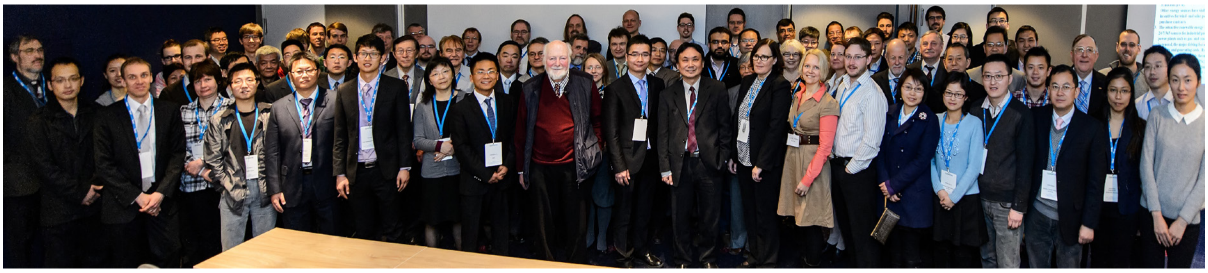
- System integration and assessment: Definition of a reference design, based on the pressure tube and pressure vessel concepts, that meets the generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance.
- Thermal-hydraulics 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.

Main activities and outcomes

System integration and assessment

The 7th International Symposium on Super-critical Water-cooled Reactors (ISSCWR-7) was held in Helsinki, Finland on 15-18 March 2015. It provided a forum for discussion on advancements and issues, to share information on technical achievements and establish future collaboration on Research and Development for supercritical water-cooled reactors (SCWR). More than 90 participants from 14 different countries representing research organisations, universities and industry were present at this event and 92 presentations were given.

Figure 3.18: **ISSCWR-7 participants**



The Canada SCWR concept has been accessed by the international experts on 5-6 October 2015 in Ottawa, Canada. The experts confirmed the achievements performed so far on the Canadian SCWR as an important contribution to the international collaboration. Several suggestions have been recommended for further improvement in the future programme. The Canada SCWR concept is an evolution of the pressure tube/channel-type of reactor with key features such as modular configuration, separation of coolant from moderator, and the use of heavy water as the moderator. Many simplifications have been introduced to reduce cost and improve accessibility for maintenance. The development focuses mainly on the reactor core, fuel,

fuel channel, out-reactor components, reactor building, reactor control system, safety system, fuel-handling system, and spent fuel bay. The balance-of-plant will be based on the state-of-the-art of fossil-fired SCW power plants at the time of construction; current high-pressure turbines can withstand a steam pressure of 25 MPa and a temperature of 625°C. The Canadian SCWR fuel assembly concept is a combination of new concepts and materials and existing, proven technologies. The insulator, flow exchanger and use of stainless steel cladding represent the former while collapsible cladding, CANLUB and high-density thoria-based fuel pellets the latter. Sufficient information was available to develop the fuel assembly concept with high confidence that initial performance will be as expected. A safety analysis of the Canadian SCWR concept has been performed. It covers key accident scenarios that could be encountered during the operation. The large-break loss-of-coolant accident remains the limiting scenario with a maximum predicted cladding temperature of 1 075°C. The development of the Canadian SCWR concept focuses on addressing the GIF technology goals (i.e. enhanced economics, safety and reliability, proliferation resistance and physical protection, and sustainability). Methodologies developed under the GIF have been applied in the evaluation. The safety characteristics of the Canadian SCWR concept have been improved with the introduction of the passive moderator cooling system and the insulated fuel assembly concept. The cost for the Canadian SCWR concept is comparable to that of the advanced boiling water reactor (ABWR). The Canadian SCWR concept shows sustainability advantages for clean air objectives, fuel utilisation, and reductions in nuclear waste.

China has completed the Chinese SCWR concept (CSR1000), which is a light water-cooled and moderated, pressure-vessel-type reactor. The reference reactor core concept consists of 157 fuel assemblies providing about 2 300 MWth power (and 1 000 MWe at 43.5% efficiency). It is developed to operate at the pressure 25 MPa, an average coolant outlet temperature of 500°C and the coolant flow rate is 1 189 kg/s. In order to simplify structural design and obtain more uniform moderation, the standard fuel assembly cluster is composed of 4 square sub-assemblies, and each of them consists of 56 fuel rods and a square water rod in the centre surrounded by a square channel box, and the cruciform control rods similar to that of BWRs are used. A 9×9 square arrangement for fuel rods in each subassembly is adopted, while central moderator box takes up 5×5 fuel rod cells. Various components in the core (e.g. internals, pressure vessel, etc.) have been developed. The reactor plant layout including active and passive safety system, etc., has been established. A safety analysis of key postulated accident scenarios, such as loss of flow rate accident and loss-of-coolant accident, has been completed. The Chinese SCWR concept has been assessed on the GIF technology goals on safety, economics and sustainability. A review of the Chinese SCWR concept has been scheduled with international experts in March 2018.

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. Furthermore, these data were applied in assessing the prediction capability of the analytical tools (such as subchannel code and computational fluid dynamic tools).

During the conceptual development phase, it is premature to perform heat transfer experiments using a full-scale fuel assembly replica. Heat transfer characteristics of the fuel assembly concept of the SCWR have been established using the subchannel code. To improve the confidence of the predictions, the Canadian ASSERT subchannel code has been assessed against experimental data of Wang et al. (2014) obtained with a 4-rod (2×2) bundle with no spacing device (i.e. bare bundle). Figure 3.19 compares ASSERT predictions of the circumferential wall-temperature distribution around the heated tubes of the 4-rod bundle against measurements at three different ranges of bulk fluid temperature. The ASSERT predictions follow the experimental circumferential wall-temperature variations. At local supercritical and subcritical temperatures, the ASSERT code overpredicts the wall temperatures at the corner region (around 180°) but underpredicts those at the subchannel regions. The overprediction is larger at the supercritical temperatures (about 7°C) than at the subcritical temperatures (about 2°C). At the pseudo-critical temperature, the ASSERT code underpredicts the wall temperatures at the corner region but predicts quite well those at the centre subchannel.

Predictions of heat transfer in subchannels of SCWR fuel assemblies are based on tube-data-based heat transfer correlations. Assessments of heat transfer correlations against an extensive database over on heat transfer in tubes at supercritical pressures revealed deficiencies in predicting the heat transfer coefficients accurately over the range of bulk fluid temperatures of interest to SCWR fuel assembly analyses. A look-up table of heat transfer coefficients in tubes has been developed. It covers a wide range of flow conditions bounding those of interest to SCWR fuel assembly analyses, particularly with pressures varying from subcritical to supercritical values. Experimental heat transfer coefficients were incorporated into the table improving further the prediction accuracy. Table 3.7 compares prediction accuracies of correlations of Swenson et al. (1965), Mokry et al. (2008) and Gupta et al. (2010), and the look-up table (Zahlan et al., 2015) against 12 293 data points obtained with tubes at three different heat transfer regions.

Figure 3.19: **Comparisons of experimental and subchannel code predicted circumferential wall-temperature distributions around heated rods of the 4-rod bundle without spacers**

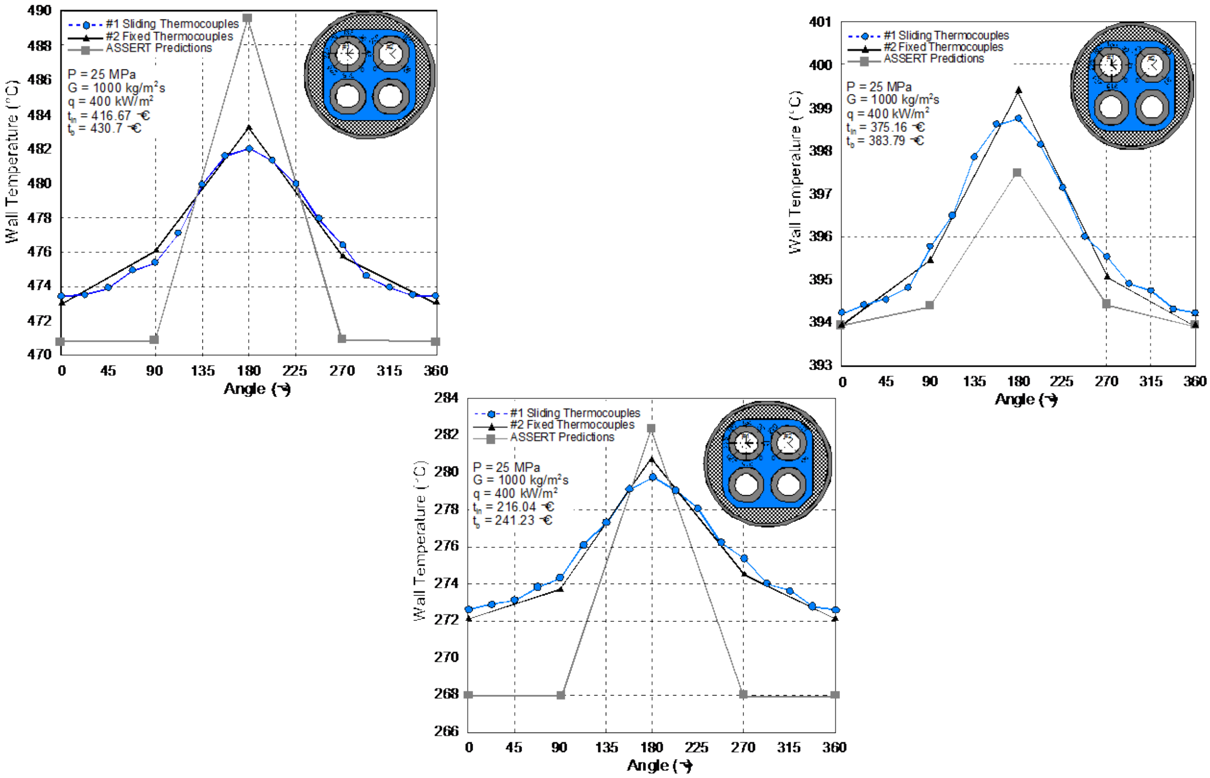


Table 3.7: **Comparisons of prediction accuracy in wall temperatures for correlations and look-up table**

Prediction method	Liquid-like region (743 data points)		Gas-like region (3 075 data points)		Close to pseudo-critical point (8 475 data points)	
	Avg. error (%)	RMS error (%)	Avg. error (%)	RMS error (%)	Avg. error (%)	RMS error (%)
Swenson et al. (1965)	-7.2	22.0	-14.3	19.6	4.4	21.3
Mokry et al. (2008)	-8.0	20.6	-7.3	15.2	1.8	15.8
Gupta et al. (2010)	-24.0	29.1	-10.4	18.5	0.5	16.8
Water-data-based look-up table (Zahlan et al., 2015)	-0.8	16.3	-0.1	11.2	0.2	15.0

Experiments are going to be performed at both DeLight (Figure 3.20) and SCMix (Figure 3.21) test facilities, which use supercritical (SC) Freon as working fluids, in the Technical University of Delft (DUT). In DeLight facility heat transfer at SC Freon will be investigated with the main emphasis on the investigation of mechanism of heat transfer deterioration. The main purpose of SCMix experiments is the mixing behaviour of supercritical fluids under strong density variation and buoyancy effect.

Figure 3.20: **DeLight facility with SC Freon in DUT**

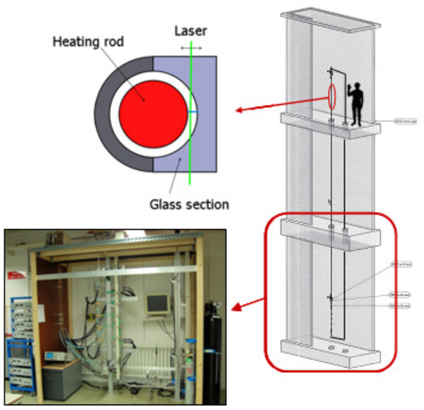
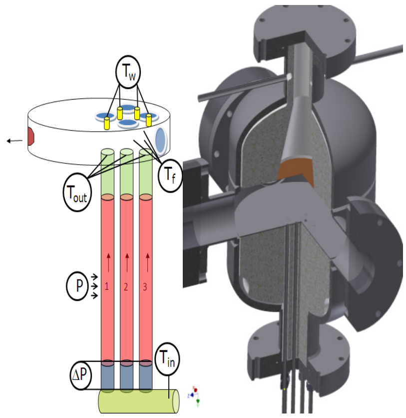


Figure 3.21: **SCMix facility with SC Freon in DUT**



Several, mainly technical, reasons have caused a severe delay in the experiments at DUT. The numerical scoping analyses showed clearly the extreme requirements for the measurement window. This caused an additional delay. Seeding required for the laser Doppler anemometry (LDA) and particle image velocimetry (PIV) measurements quickly disappears. Different injection methods and frequencies have been assessed. As much as possible, recirculation zones are being avoided in the loop. The extremely small area of interest close to the wall of the heated rod required development of a ray-tracing programme to determine the measurement position with sufficient accuracy.

Velocity profile measurements have been performed for an unheated rod at different subcritical and supercritical temperatures which will already provide valuable information to model developers. Accurate measurement of parameters as defined for future code validation will be carried out after the termination of the Thermal-hydraulics of Innovative Nuclear Systems (THINS) project and the experimental results from DUT can be used later for further experimental validation of the numerical approaches outside of the scope of the THINS project.

In the frame of the THINS project, CFD simulation of flow and heat transfer of supercritical fluids was carried out by two groups, i.e. KTH and PSI/Pisa University. Under collaboration between the Pisa University and PSI, several in the open literature available 4-equation turbulence models were taken into consideration, i.e. Abe et al. (1995), Hwang-Lin (1999), Deng et al. (2000) and Zhang et al. (2010), and were implemented into the PSI in-house code THEMAT. Numerical results were compared with some selected experimental data obtained in supercritical CO_2 and supercritical water. The results show that the capability and accuracy of the selected turbulence models depend on test parameters and no general conclusions can be drawn. However, it is pointed out that significant future efforts are required to improve the turbulence modelling of heat transfer to supercritical fluids.

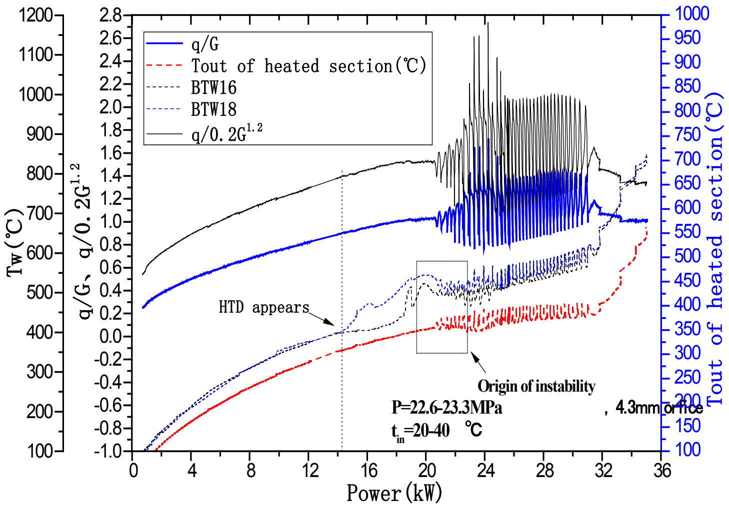
The SCWR thermal hydraulics research in China includes four major aspects: heat transfer and flow tests of SCW in tubes, annular channel and simple rod bundles; safety performance-related tests including natural circulation, critical flow, CHF near critical pressure, flow stability in parallel channels; assessment and applicability of CFD codes; research on scaling method of different supercritical fluids. Based on these years of research, experimental techniques and analysis methods of SCW T-H have been mastered, a T-H database of SC fluid has been established, and some thermal-hydraulic characteristics of SC fluid have been obtained.

In the next five years, more research has been planned on thermal-hydraulics of SCWR technology in China. The crucial techniques including flow stability, T-H performance of rod bundles, basic hydraulic performance, multi-scale modelling analysis, critical flow, etc., will be focused on towards SCWR industrial application. Various projects will be supported by Chinese government in the strategic planning of national SCWR development, such as R&D on SCWR technology (phase II) from the Ministry of Industry and Information of China, Fundamental Research from National Natural Science Foundation of China.

A new benchmark exercise has been lunched to evaluate simulation tools on supercritical water flowing through a 2×2 rod bundle. The experimental data will be provided by Nuclear Power Institute of China (NPIC) and Canadian Nuclear Laboratories (CNL). The participants will first do blind-calculation based on the operating conditions and geometry parameters and then compared with the experimental data afterwards.

Figure 3.22: **Test sections of tube, annular and 2×2 rod bundles**

Figure 3.23: **Natural Circulation of SCW**



Materials and chemistry

M&C PMB members and colleagues presented their latest results at the 7th International Symposium on Supercritical Water Reactor, held in Helsinki, Finland, 15-18 March 2015. More than 30 talks related to M&C activities were given during this symposium. In addition, presentations from researchers outside the nuclear field were given in order to initiate discussion on cross-cutting issues. R&D has been focusing on selection and qualification of candidate alloys for all key components in the SCWR, including general corrosion and stress corrosion cracking (SCC) tests, development work on oxide modelling and water chemistry strategies as well as water radiolysis studies.

A major activity of the M&C PMB has been the organisation of a 2nd Round Robin corrosion exercise involving 12 partners from the EU, Canada and China. This exercise will compare the results of corrosion tests in different test facilities, the main objective being identifying the origins of differences observed in the results of the first Round Robin exercise e.g. coupon preparation, differences in flow or exchange rates. For the 2nd Round Robin exercise JRC-IET provided pre-polished test coupons to eliminate differences in sample preparation between institutions, and more test parameters (flow rate, heat-up rate) were fixed by the PMB members to reduce differences between test facilities. The test plan for this exercise was finished at the end of 2015. Tests will start in early 2016 and be completed by the end of 2016. Preliminary results will be reported at the beginning of 2017.

At VTT, the Academy of Finland project IDEA (Interactive modelling of fuel cladding degradation mechanisms, 2012-2016) has focused on assessing general corrosion mechanisms of potential candidate materials. Modification of high-temperature steels is a promising option to enhance a material's resistance against degradation in supercritical water (SCW). Possible modification methods to achieve the required corrosion resistance are surface coatings, thermomechanical treatment or modifications in the chemical composition. Autoclave tests at 700°C/25 MPa were initiated in late 2015 to assess the general corrosion resistance of high-performance materials with high Ni content including 800H and Sanicro 25, different alumina-forming alloys, and materials used in conventional supercritical fossil-fired power plants such as alloy 263 with different chemical modifications, and alloy 617. In this study, the oxidation behaviour of will be evaluated by measurements of weight change and oxide film thickness. Evaluation of the results is underway. In addition to evaluating the role of microstructure, a specific thermomechanical treatment was implemented to process Alloy 800HT in collaboration with the University of Saskatchewan. It was found that thermomechanical processing not only improved oxidation resistance, it also alleviated oxide scale spallation. The results demonstrated that in comparison with texture, grain size has stronger effect on the oxidation resistance of Alloy 800HT in SCW at 600°C. Coatings are finding applications in the nuclear field, both for existing reactors as well as for innovative reactors like the SCWR. A large variety of industrially available coatings exist, and the following coatings were investigated in SCW at 700°C/25 MPa: CrN on alloy 316L, NiCrAlY on alloy 214, CrAl on alloy 304 and different variations of CrAlNi on alloy 304. This autoclave test started in late 2015 and will be finished in 2016. Coating technology is mature, well advanced and relatively cheap, but it is evident that application of coatings in an NPP will require significant safety evaluations.

CIEMAT (Spain) activities in 2015 mainly focused on SSC tests on austenitic stainless steel 316L. In this work alloy 316L was tested in deaerated SCW at 400°C/25 MPa and 30 MPa and 500°C/25 MPa to determine how variations in water properties influence its SCC behaviour and to better understand the mechanisms involved in SCC in this environment. In addition, a selected oxide layer formed at 400°C/30 MPa/<10 ppb O_2 was analysed to gain insight into these processes. It was found that alloy 316L showed susceptibility to SCC in SCW. The susceptibility seemed to increase with temperature and pressure, although the effect of pressure seemed to prevail over the effect of temperature. In the oxidation studies it was found that the oxygen concentration did not have a measurable effect on sample weight gain, in good agreement with results reported by other authors. It was found that chromium was only incorporated into the external oxide layer when the oxygen concentration was low. A possible explanation is the transformation of Cr3+ into Cr6+, which was also observed for the same alloy tested under BWR conditions (288°C with 8 ppm of O_2).

At JRC-IET Petten, SCWR research was conducted within the institutional project “Integrity and Ageing of present Light Water Reactors” (IntAG-LWR) with a focus on corrosion and SCC resistance of candidate materials in SCW, as well as the development of electrochemical potential (ECP) sensors capable of working in an SCWR environment.

General corrosion tests focused on assessment of the effect of surface finish and water chemistry on two candidate materials, 310S and alloy 800H. Besides the standard surface finish, samples were treated using two different surface finish techniques: shot-peening (performed in VTT) and sand-blasting (JRC-IET). Corrosion exposures with a target of at least 5 000 h have been conducted at 550°C/25 MPa with 2 000 ppb of dissolved oxygen. Results obtained from specimens taken out after 600 and 1 200 h indicate beneficial effect of surface cold work due to sand-blasting or shot-peening.

In the past, 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, the effect of ageing was evaluated in follow-up tests in 2015. First, SSRT specimens were exposed in the furnace at 750 and 850°C under an Ar overpressure for 200 h. Subsequent tests in 550°C/25 MPa SCW with 2 000 ppb of dissolved oxygen did not show any significant decrease of SCC resistance for all three materials.

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 using pneumatic bellows-based loading devices were performed within the internal project. An iron/iron oxide electrode was developed by the IFE OECD Halden Reactor Project for *in situ* corrosion monitoring up to 700°C in SCW. The first two prototypes were installed in the JRC-IET SCW autoclave. The first tests focused on 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 in 2014. In 2015, a dedicated electrochemical cell which included a Fe/Fe_3O_4 reference electrode, a Pt-basket counter electrode and a 316L cylinder working electrode was installed in one of the SCW autoclaves. Electrochemical Impedance spectroscopy (EIS) measurements have been performed to investigate: *i)* effect of temperature, in particular when crossing from sub- to supercritical water; *ii)* effect of exposure time by making EIS measurements at 500°C/25 MPa SCW for more than 2 000 h (Figure 3.24); and *iii)* effect of pressure by making EIS measurement while gradually decreasing the autoclave pressure from 25 to 10 MPa.

The activities of Research Centre Řež (Czech Republic) in 2015 focused on development and testing of a leak-tight joint based on conductor-insulator for instrumentation in an SCW environment. The joint stayed leak-tight for 500 h at 600°C and 25 MPa and stayed leak-tight. The joint design will be used as basis for electrochemical sensor development (Figure 3.24).

Figure 3.24: **Effect of time on the resistance and capacitance of the oxide layer on 316L at 500°C/25 MPa SCW calculated from EIS spectra over more than 2 000 h**

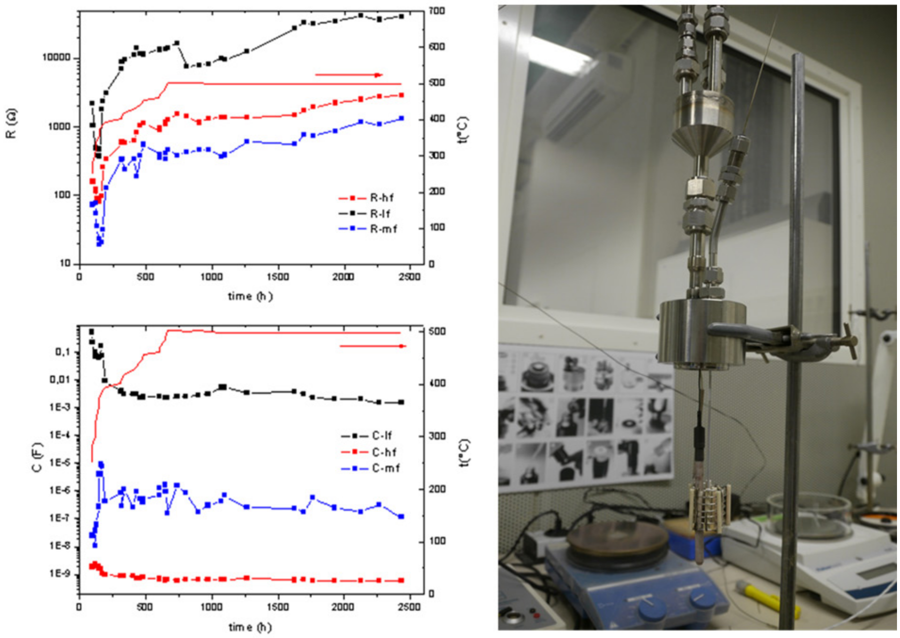
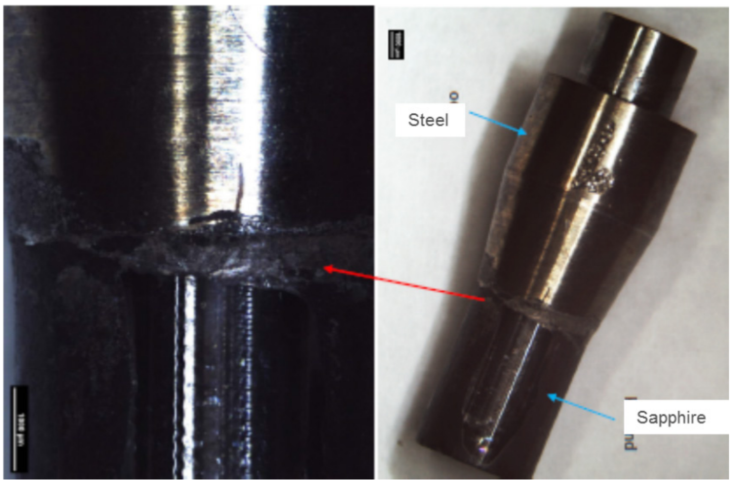


Figure 3.25: **Joint after exposition**



In the project ARMAT, material screening tests for SCW power plants were performed. One of the main goals of the project is testing of ceramic (corundum, sapphire-based) and metallic materials in SCW. In the second part of the project the influence of the $CuAl_2O_4$ content of the quality of turbine blade casting and the stability of components in SCW will tested. The first set of specimens (Figure 3.26) was fabricated and will be tested in 2016.

Figure 3.26: **Casted blade 3% of $CuAl_2O_4$ (both left) and 6% $CuAl_2O_4$ (right)**

Figure 3.27: **3D visualisation of the in-pile supercritical water loop installation**

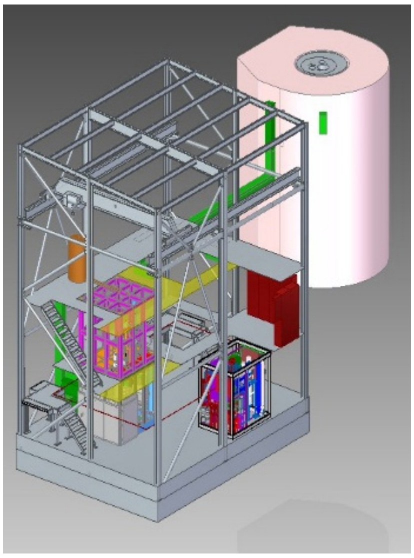
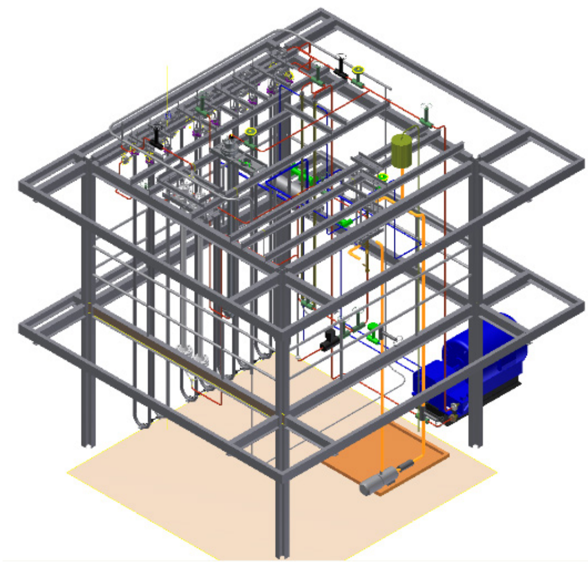


Figure 3.28: **Supercritical CO_2 loop**



The project SUSEN is a large investment, with the goal of building new experimental Gen IV and fusion facilities. In 2015, the following activities were performed:

- In-pile supercritical water loop in Řež: EUR 1 500 000 investment, installation to reactor planned for mid-2017. In 2015, the first part of the licensing application was prepared, new equipment (i.e. H_2, O_2 analysers) purchased, and piping and cooling circuits and other infrastructure for connections on reactor hall were prepared.
- Design, analyses and construction of the SCO_2 – loop; the installation is ongoing, commissioning is expected for mid-2016.
- Design and analyses of the ultracritical water loop (UCWL) facility, basic and detail designs were finalised, installation is ongoing, and commissioning is expected for December 2016.

In 2015, the Canadian materials and chemistry programme focused on further evaluation of five candidate fuel cladding alloys (347 SS, 310 SS, alloy 800H, alloy 625 and alloy 214) in preparation for assessments of the Canadian SCWR concept by a panel of Canadian experts in February 2015 and by a panel of international experts in October 2015. The methodology used to assess the materials is described in Guzonas et al. (2015).

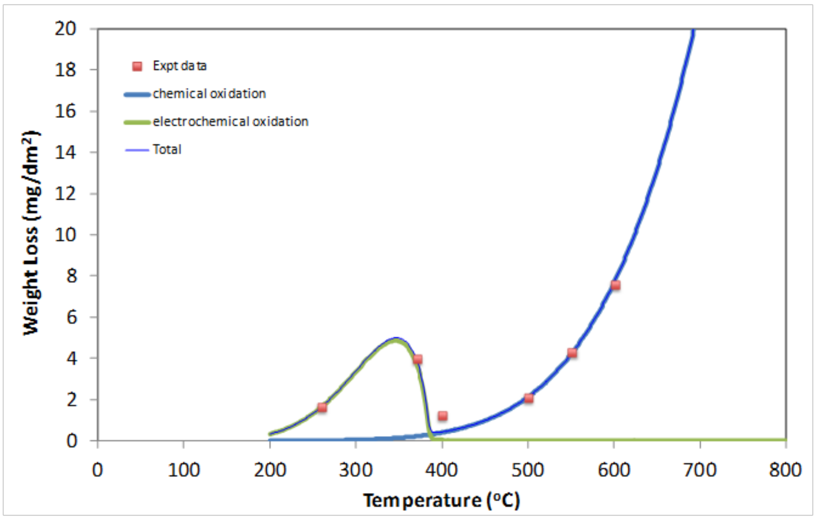
Oxidation testing of capsule samples of SS310S and SS316L at 500°C in SCW is underway for exposure times of up to 50 000 h; 50 000 h exceed the proposed in-service life of the Canadian SCWR fuel cladding, and these data will be extremely important in demonstrating the long-term performance of these materials. It was noted that there was a remarkable similarity between the nanoporosity in the samples used in SCW cracking studies and that found in natural hydrothermal systems (geological samples). The need to consider hydrothermal chemistry effects when considering the mechanism of corrosion in SCW has been highlighted before. Recent molecular dynamics simulations have shown that the density of water molecules at an $Fe(OH)_2$ surface is higher at the surface than in the bulk. This means that the alloy surface will be covered mostly by adsorbed water. The concentrations of reactants such as oxygen at the surface can be higher or lower than those in the bulk fluid.

Measurements at Trent University further highlighted the role of hydrothermal chemistry effects. Concentrations of dissolved metals, O_2 and H_2 at the exit of an alloy 800H tube exposed to SCW containing 20 ppm oxygen were monitored. Initially both water and O_2 act as oxidants, resulting in significant H_2 release and release of Fe, Al, Ni and Mn into the SCW at very low concentrations. No Cr release is observed during this time, attributed to the presence of Mn in the Cr oxide. No O_2 is observed at the test section exit during this time. Eventually the film becomes thick enough to limit O_2 access to the surface and O_2 appears at the test section outlet. The data suggest restructuring of the oxide layers at this time. Mn dissolution from the oxide eventually results in Cr oxidation and dissolution, and formation of an outer layer composed of both magnetite and hematite. At steady state, Cr and Al were the major elements released, and no iron release was observed. The effect of flow rate on release was small relative to the effect of temperature for Fe, Ni and Mn, but significant for Cr and Al. The data show that formation of an equilibrium film requires several hundred hours, the time required increasing with increasing temperature.

The effect of water pressure (0.1, 8 and 29 MPa) on oxidation of alloy A-286 at 625°C for 1 000 h was assessed by transmission electron microscopy. Exposure at 29 MPa resulted in an oxide layer about 1 µm thick containing Fe_2O_3, a spinel phase and Cr_2O_3. Isolated internal oxidation, up to 10 µm, as well as recrystallisation of substrate material occurred. Exposure at 0.1 MPa resulted in little oxidation. The surface oxide was ~200 nm thick, comprised of a top layer of Cr_2O_3, a thin sub-layer of SiO_2, and limited grain boundary oxidation as $(Cr, Fe, Ti)_3O_4$ and TiO_2. Exposure at 8 MPa resulted in high external and internal oxidation, with about 20 µm of Fe_2O_3 on the external surface, followed by a partially oxidised zone (up to 20 µm).

Examination of several years' worth of corrosion and SCW exposure data for alloy 625 and alloy 800 is underway at the University of New Brunswick. It was found that after an initial period of relatively rapid oxidation and weight gain, nickel-based alloys typically show weight loss at all temperatures examined while the stainless steels exhibit weight gains. A procedure was developed to determine the kinetic parameters required to predict the long-term corrosion behaviour of these alloys; these data were used during the selection of the Canadian SCWR fuel cladding alloy (Figure 3.29).

Figure 3.29: **Experimental and modelled corrosion rates of Alloy 625**

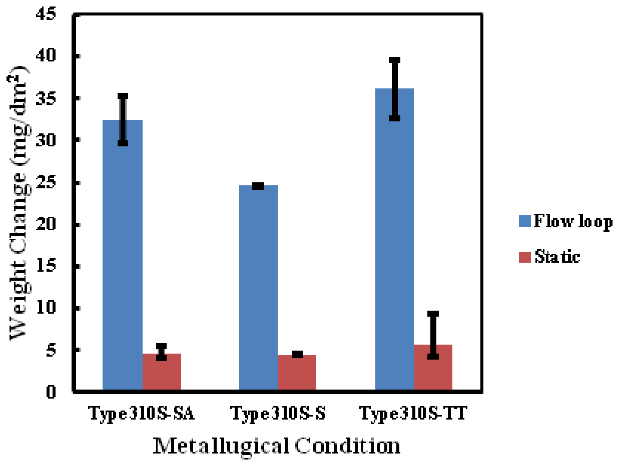


Source: Choudhry, 2015.

Corrosion performance of UNS S31008 and UNS N08810 was tested in an autoclave at 625°C and 25 MPa. Weight change and microscopy characterisations indicated that these alloys exhibited good general corrosion resistance in SCW although a noticeable increase in corrosion was observed after the 4th thermal cycle exposure. The SCC susceptibility of the alloys was studied using slow strain rate testing in an SCW loop. The results indicated that there is a threshold strain level below which SCC could not be developed on these steels. The structure and composition of the oxide scales formed on Alloy 800HT in SCW after ~500 h were studied by TEM to better understand the metallurgical factors influencing the protectiveness of the oxide layers. It was proposed that formation of a discontinuous Fe_3O_4 outer nodular layer is largely controlled by the underlying microstructure, particularly the presence of ε-martensite plates. Reducing or eliminating the small volume fraction of ε-martensite from the starting microstructure of Alloy 800HT may be a method to optimise the corrosion resistance.

The microstructural instability of the cladding alloy and its effects on corrosion and SCC were investigated. The effect of microstructure instability on the corrosion resistance in SCW was measured using a static autoclave (stagnant, deaerated) and a flow loop (flowing, 8 ppm dissolved oxygen). The microstructure instability resulting from high temperature had little influence on weight gain (Figure 3.30), albeit after only relatively short exposure times. While a detrimental effect may occur if a continuous network of intermetallic phase precipitates formed on the short-circuit grain boundary diffusion paths, such formation is unlikely over the in-service life of the fuel cladding (~30 000 h) based on published predictions of intermetallic phase precipitate volume fractions after prolonged exposure times. Figure 3.30 shows that microstructure instability effects are insignificant compared to the combined effects of flow and dissolved oxygen content. While precipitates have no immediate effects on corrosion, the brittle sigma phase formed from high-temperature exposure of stainless steels could be a concern for SCC initiation through an internal oxidation mechanism, and additional work is underway to assess this.

Figure 3.30: **Weight change data for pre-treated type 310S stainless steel after exposure in 25 MPa SCW at 550°C**



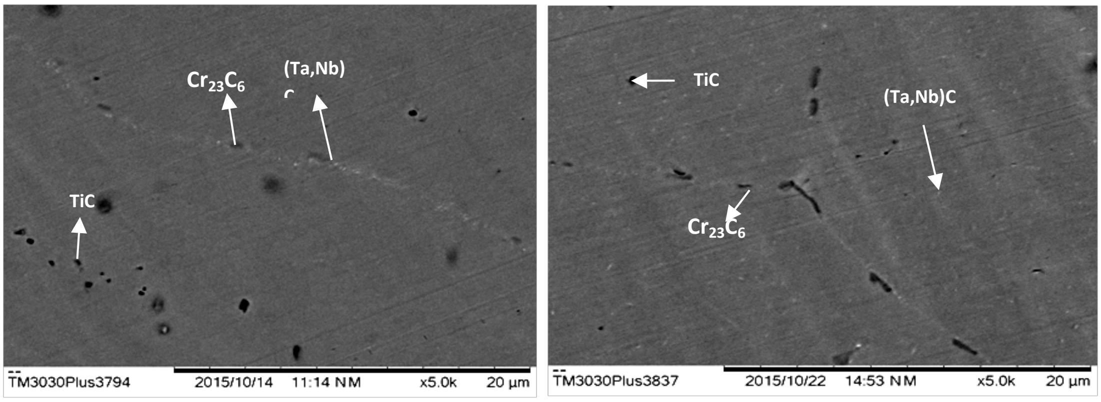
Work has been carried out with the University of Saskatchewan to develop an integrated approach to fuel cladding material selection that combined corrosion assessment with thermal-hydraulics and neutronics calculations. The focus was on assessing the significance of nickel content in the alloy on fuel cladding selection. The effect of Fe4+ irradiation damage and subsequent annealing on the hardness of AISI 310 and Alloy 800H was studied at Western University and an irradiation hardening – thermal softening model developed. The effect of accumulated helium on the hardening of AISI 310 and Alloy 800H was also presented. Studies of irradiation effects on SCC using proton irradiation are ongoing, focusing on effects of irradiation dose on the threshold strain level for crack initiation in SCW.

Work continued developing chemistry control strategy for the Canadian SCWR concept. Measurements of the solubilities of key oxides involved in the stability of corrosion films in SCW (MoO_2, MoO_3, Cr_2O_3) is underway at St. Francis Xavier University. In oxygenated water a tetrahedral molybdate complex is dominant up to 500°C and 145 MPa regardless of Mo concentration. Work at the University of Guelph highlighted two chemistry issues: i) the importance of oxidised uranium species in studies of fission product transport; and ii) the possible effects of solutes on the critical point in crevices or other confined spaces.

Work continued on understanding the effects of water radiolysis on corrosion in SCW. Experiments at Western University showed that, unlike in experiments carried out for carbon steel in previous lower-temperature studies, adding oxygen did not simulate the effects of radiation on corrosion. Although the tests were of short duration, it was noted that the inner layer forms over the first few days of exposure and therefore factors such as radiolysis that alter the formation of this film may have long-term consequences. This is consistent with the results of the Trent University tests reported above. At Mt. Allison University the rate constants required for modelling of water radiolysis in SCW were re-evaluated. It was concluded that, based on simple calculations, the concentration of O_2 produced by radiolysis of pure water at 400°C should be 0.006 times less than O_2 from radiolysis at room temperature, and will increase at temperatures above 400°C to just less than the value at room temperature.

At Nuclear Power Institute of China, composition design of fuel cladding and internal component materials has been completed in order to improve the structure stability of 310S in long-term operation condition. On the basis of 310S (25Cr-20Ni-0.08C wt.%), the cluster-plus-glue-atom model is introduced to design new alloy by multi-element co-alloying (Nb, Ti, Zr, Ta, W) of 310S. This cluster model dissociates the solid solution structure into a cluster part and a glue atom part: the cluster is the nearest-neighbour polyhedron and glue atoms are located in-between the clusters. There are three types of atom sites: the cluster centre, the cluster shell, and the glue sites. The place where the atom occupies in these three sites is determined by the interactions (enthalpies of mixing H) between the solute elements and the base solvent one. A solute showing a negative ΔH tends preferentially to occupy the cluster centre, and that showing a positive ΔH occupies the glue site. The cluster model could be expressed with a composition formula of [cluster](glue atoms)x (x is the glue atom number for matching one cluster). It is found that the stable FCC solid solutions generally correspond to the cluster formula of [CN12 cluster](glue atoms)1~6, where the cluster is a cub octahedron with a co-ordination number of 12. The basic Fe-Ni-Cr ternary composition of 310S(Fe55.0Cr24.7Ni22.3 wt.%) is determined as the cluster formula [Cr-($Fe_{10}Ni_2$)](Cr_4Ni_2), where Cr represents Cr-similar elements (Cr,Nb,Ta,Ti,Zr) and Ni represents Ni-similar ones (Ni,Mn,C). A new alloy Fe-24.6Cr-22.2Ni-1.01Mo-0.09Nb-0.09Ti-0.17Ta-0.05C is designed from this formula by Nb, Ti and Ta co-alloying. The alloy ingots were prepared by vacuum arc melting processing. These ingots were then solid-solutioned at 1 200°C for 1h, stabilised at 950°C for 0.5 h, and aged at 800°C for 24 h. The experimental results indicate that after stabilisation treatment, a large amount of (Nb,Ta)C nanoparticles with a size of 50-70 nm are distributed on the grain boundaries of the matrix, besides minor TiC and $Cr_{23}C_6$. After ageing treatment, the MC nanoparticles dispersed in the inner-grains uniformly, with no change of the particle size; a few $Cr_{23}C_6$ particles precipitate on grain boundaries, with a size of about 1 m. As shown in the SEM micrograph of the matrix (Figure 3.31), the addition of Nb, Ti and Ta can form fine TiC, (Nb,Ta)C particles and the coarse $Cr_{23}C_6$ particles are suppressed. Research was carried out to investigate the effect of water chemistry on the SCC behaviour. The data were collected by SCC tests in candidate chemistry regimes to explore beneficial chemistry methods.

Figure 3.31: **SEM image of the Fe-24.6Cr-22.2Ni-1.01Mo-0.09Nb-0.09Ti-0.17Ta-0.05C alloy**



(a) After stabilisation treatment

(b) After stabilisation treatment and ageing treatment

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 high-performance light water reactor (HPLWR) evaporator conditions have ended on 31 December 2014 and in mid-2015, respectively. The majority of the objectives foreseen by the projects have been achieved; the main results obtained are as follows:

- The facility for the fuel qualification test in supercritical water has been designed. The facility will be operated in the research reactor LVR-15 in Řež, Czech Republic. The fuel qualification test shall be performed at evaporator conditions of the HPLWR, where the heat flux is the highest and therefore, the heat transfer deterioration is challenged; the nominal temperature and pressure around the fuel assembly are 384°C and 25 MPa, respectively. The test shall be performed on a small-scale fuel assembly with four fuel rods and with dimensions and design (fuel rod diameter, fuel rod pitch, wire wrap spacer, square assembly box) identical to the HPLWR fuel assembly. The active height of the fuel assembly is limited by the LVR-15 core height to 600 mm.

- The facility is composed of a pressure tube containing the test section with the fuel assembly located inside one cell of the reactor core grid and the remaining systems assembled outside of the reactor in an adjacent building.
- The design of the test section has been analysed at steady-state conditions to assure that the surface temperature of the fuel rods does not exceed the maximum allowed temperature of 550°C.
- Safety systems have been designed for the facility to prevent fuel failure during the test. Safety analyses have been performed for design-basis accidents as well as for an accident beyond the design basis to estimate all possible source terms.
- Candidate materials for the fuel cladding have been selected at the beginning of the project among austenitic stainless steels. The steels were tested for general corrosion and stress corrosion cracking resistance. From the obtained results, the stainless steel 316L has been chosen for the fuel cladding.
- The final outcome of the project was presented at the 7th International Symposium on SCWR (ISSCWR7) from 15-18 March 2015 in Helsinki.

Besides the collaborative projects mentioned above, China is also planning to do two kinds of in-pile irradiation test, in order to qualify the material and fuel of Chinese supercritical water reactor. One or two kinds of candidate internal structure materials and fuel cladding materials respectively are planned to conduct the irradiation test during 2016~2019 with a newly designed high-temperature material irradiation rig. The materials' irradiation data from these tests and the corresponding results of post-irradiation examination (PIE) would be an essential part of the project to promote the R&D of SCWR. An in-pile fuel assembly irradiation test loop with supercritical water in the research reactor is in the plan which is able to simulate the typical operation conditions of the SCWR fuel assembly and conduct the irradiation test to qualify its performance. The project is divided to four stages. The first stage is conceptual design (2016~2019).

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3.5 Sodium-cooled fast reactor (SFR)

Main characteristics of the system

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

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 Phénix 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 (CEFR).

- 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 3.32.
- An intermediate-to-large size (300 to 1 500 MWe) pool-type reactor with oxide or metal fuel as shown in Figure 3.33 and Figure 3.34.
- 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 3.35.

The two primary fuel recycle technology options are *i)* advanced aqueous and *ii)* 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 3.32: **Japanese sodium-cooled fast reactor (loop-configuration SFR)**

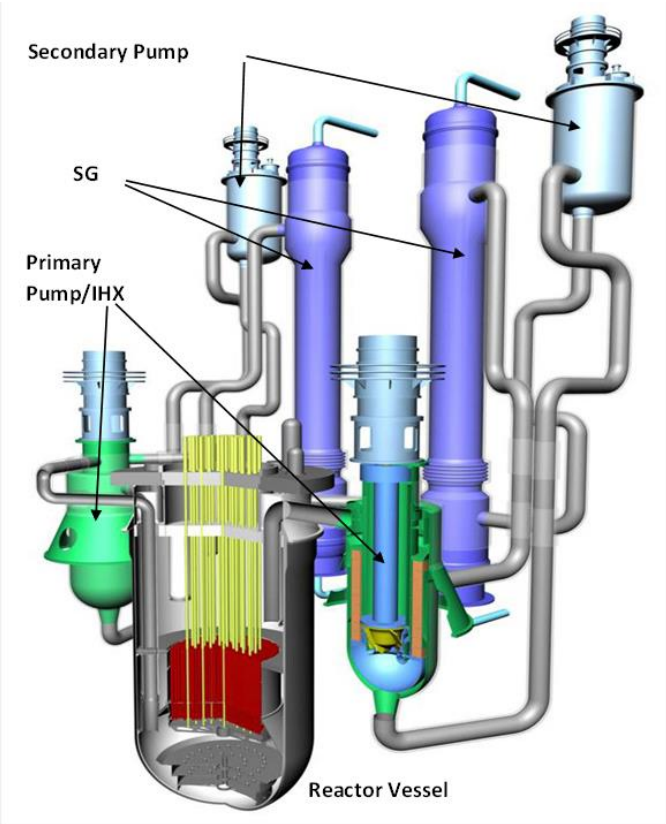


Figure 3.33: **Example sodium fast reactor (pool-configuration SFR)**

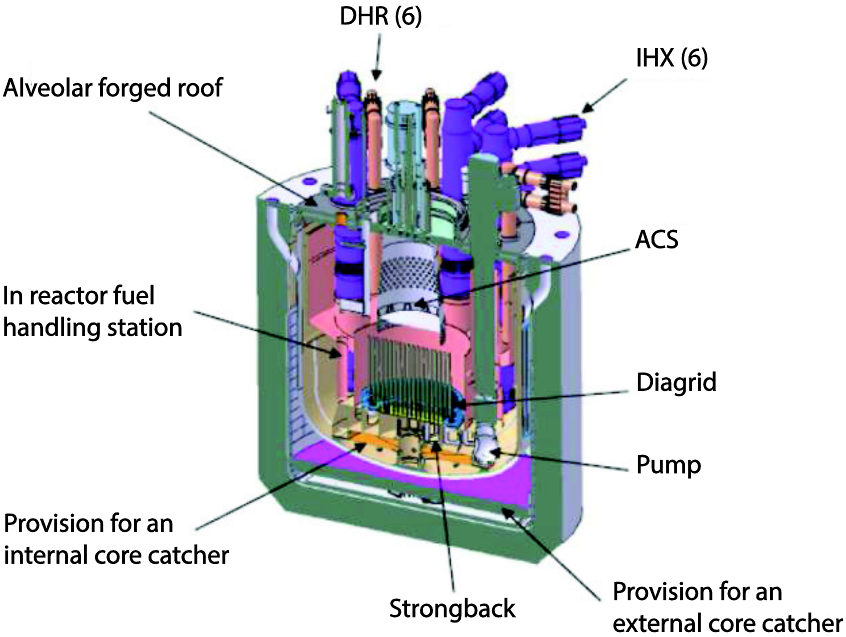


Figure 3.34: **Korea advanced liquid metal reactor (pool-configuration SFR)**

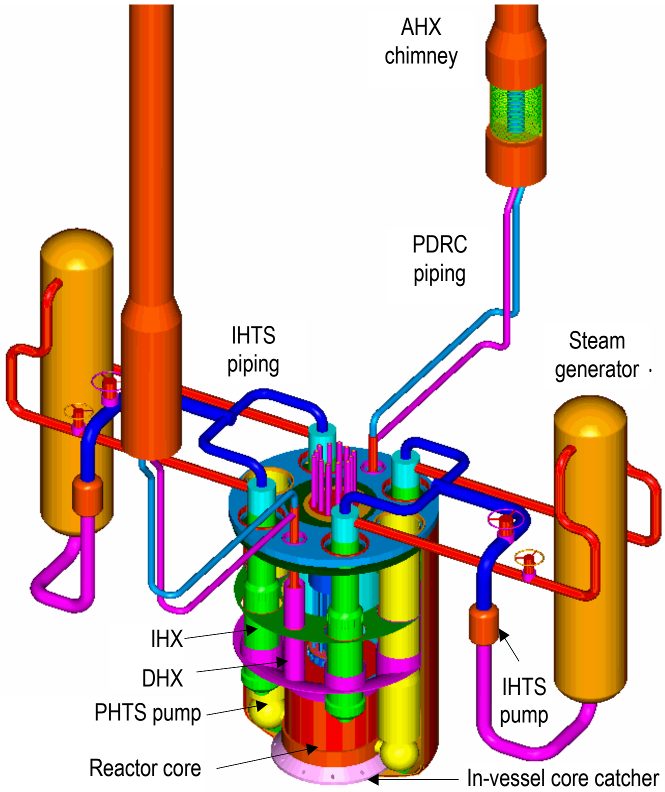
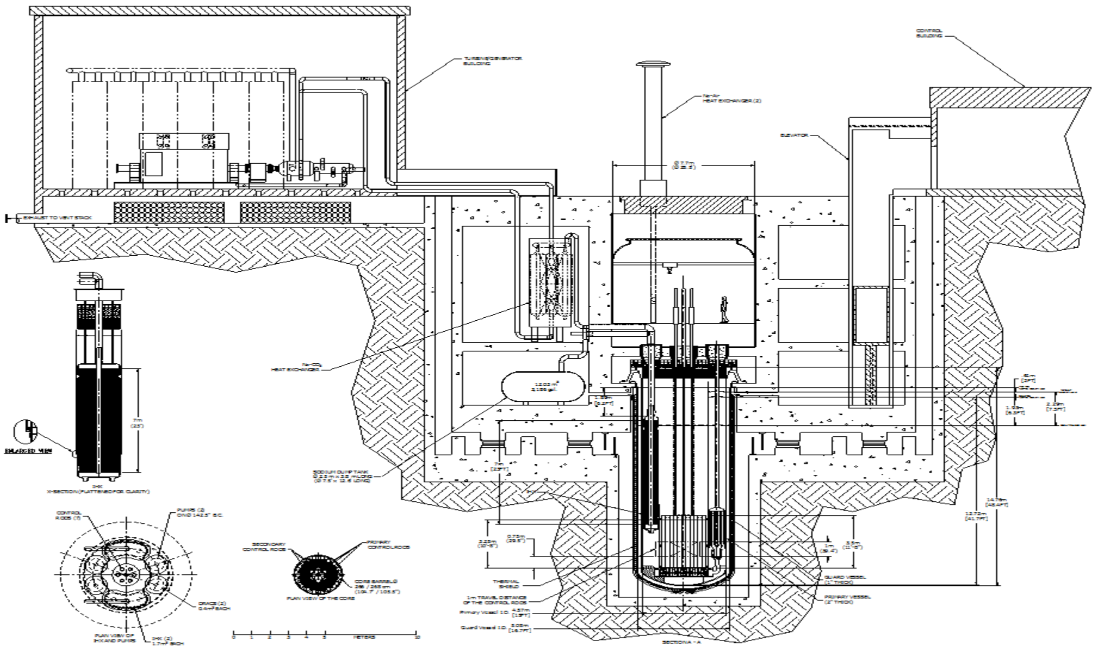


Figure 3.35: **AFR-100 (small modular SFR configuration)**



Status of co-operation

The 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, Science, ICT and Future Planning, Korea.
- China National Nuclear Corporation, China.
- Rosatom, Russia.

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 Project Arrangement for Advanced Fuel was amended in October 2015 to include contributions of China and Russia. The new CD&BOP Project Arrangement (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 Russia. The Project Arrangement for System Integration and Arrangement (SIA) was signed by all members in 2014.

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, Russia, 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, Phénix, 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 burn-up MA-bearing fuels as well as claddings and wrappers withstanding high neutron doses and temperatures. It includes: research on remote fuel fabrication techniques for fuels that contain minor actinides and possibly traces of fission products as well as performances under irradiation of fuels, claddings and wrappers. Candidates under consideration are: oxide, metal, nitride and carbide for fuels, alternate fast reactor fuel forms and targets for special applications (e.g. high temperature), and ferritic/martensitic and oxide dispersion-strengthened (ODS) steels for core materials.

Component Design and Balance-of-Plant project

Research on component design and balance-of-plant covers experimental and analytical evaluation of advanced in-service inspection and repair technologies including leak-before-break assessment for advanced materials, advanced steam generators and development of alternative energy conversion systems, e.g. using Brayton cycles. The incorporation of such technologies, if shown to be viable and shown to perform, would reduce the levelised cost of electricity generation significantly. The primary R&D activities related to the development of advanced balance-of-plant (BOP) systems are intended to improve the capital and operating costs of an advanced SFR. The main activities in energy conversion system include: *i)* development of advanced, high reliability steam generators and related instrumentation; and *ii)* 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 and trade study contributions:
  - **2014:** Annual contributions of self-assessment results for SFR design tracks.
  - **2014**: Annual contributions of trade studies to assess key performance features of generation IV SFR concepts.
  - **2014:** Solicit economics assessment using the Economics Methodology Working Group (EMWG) methodology.
  - **2015:** Solicit safety assessment using RSWG methodology.
  - **2017:** Solicit proliferation assessment using Proliferation Resistance and Physical Protection (PR&PP) 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, Phénix, 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:

- **2007-2017:** Preparation for the limited MA-bearing fuel irradiation test.
- **2007-2017:** Preparation for the licensing of the pin-scale curium-bearing fuel irradiation test.
- **2007-2017:** Programme planning of the bundle-scale MA-bearing fuel irradiation demonstration.

Main activities and outcomes

System Integration and Assessment (SIA) project

The integration and assessment activities are conducted directly as part of the Signatory's responsibilities for preparation and consultation at the SIA PMB meetings. The second official PMB meeting was held 8-9 April in Ispra, Italy. The third PMB meeting was held 13-14 October in Obninsk, Russia. At each PMB meeting:

- the list of major System options and design tracks was updated;
- the comprehensive list of R&D needs was reviewed;
- the recent R&D results of each SFR Technical Project were reviewed to assure consistency with generation IV system options and R&D needs.

At the October PMB Meeting, the United States proposed AFR-100 as a design track for the small, modular system option. Based on the generation IV assessment, the PMB approved AFR-100 as a design track, replacing the previous small modular fast reactor (SMFR) concept. New design track contributions are expected from several project members in the near future.

Trade study contributions in 2015 included a preliminary study on scenarios for the example sodium fast reactor (ESFR) deployment (Euratom), studies on fuel option and core configurations for CFR1200 (CIAE), and generalised trade studies on comparison of homogeneous and heterogeneous transmutation of minor actinides (CEA), and impact of core outlet temperature for a variety or energy conversion technology options (DOE). A multi-year self-assessment by JAEA will apply the GIF safety design criteria and guidelines, as developed by the GIF Risk and Safety Working Group, to the Japanese sodium-cooled fast reactor (JSFR) design. The first year assessment of "lessons learnt from Fukushima" organised the safety features against the Fukushima events, and assessed the safety measures.

Safety and Operation project

Since 2012, R&D activities were implemented, as mentioned above, within the framework of three 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 main developments carried out by all SO PA members in these three areas in 2015 summarised as follows.

WP SO 1: Methods, models and codes

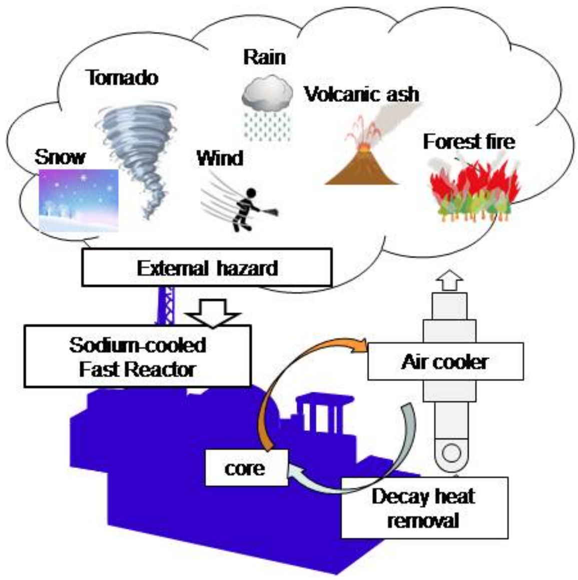
A specific method has been developed by French specialists for 3D neutronic modelling SFR core with modified geometry under seismic conditions. This approach, based on local core deformation corresponding to single assembly displacement and first order perturbation theory, permits to evaluate maximal positive and negative reactivity insertion due to core deformation induced by a postulated mechanical energy supplied to the SFR core. The given method was applied for the Phénix core analysis, but it can be easily scaled-up for large-sized core safety analysis.

Japanese specialists continue development of L-1 probabilistic risk assessment (PRA) methodology against external hazards that can influence on SFR safety. This activity is carried out within the framework of a four-year research project since 2012, that is aimed at development of PRA and margin assessment methodologies of decay heat removal function against such external hazards as snow, tornado, wind, rainfall, volcanic eruption and forest fire (Figure 3.36). In 2015, studies were dedicated to analysis of impact of strong wind on implementation of decay heat removal function. It was developed strong wind PRA methodology based on application of Weibull and Gumbel distributions for hazard curves and use of Saffir-Simpson hurricane scale for hazard categories. Results obtained by using wind PRA methodology showed 6x10-9/year of CDF.

Another area of activities of Japanese specialists within WP SO 1 is related to development of evaluation method for sodium-concrete reaction in case of sodium leak. This study included analyses of possible sodium-concrete reactions and development of appropriate reaction model, implementation of Na-SiO_2 reaction experiment, XRD analysis of solid products, creation of phase diagram and evaluation of rate constant. It was obtained that reaction of concrete aggregate with sodium is similar to that of Na-SiO_2 reaction. Temperature of Na-SiO_2 reaction was identified at around 800 K. XRD analysis and phase diagram showed that overall reaction is likely to occur.

In 2015, Korean specialists continued SAS4A code model development to perform severe accident analysis for PGSFR. The DEFORM-5A model, PINACLE-M model and thermal physical property models have been developed for metal fuel and integrated with the SAS4A code. The SAS4A code has been adjusted to provide calculation of accident with inlet FSA cross-section blockage and appropriate calculation has been carried out. Comparison of results obtained by previous and modified versions of SAS4A code showed appreciable difference in configuration of molten fuel cavity arising during accident with inlet FSA cross-section blockage.

Figure 3.36: **Risk assessment methodology of decay heat removal function against external hazards**



Within WP SO 1, mechanistic source term (MST) was developed for a metal-fuel, pool-type SFR in ANL. Deliverable of 2015 includes description of the history of regulatory source term analyses in the United States, including the use of TID-14844 and NUREG-1465/R.G. 1.183 for LWRs, along with past SFR licensing efforts and the NRC's encouragement of mechanistic source terms for advanced reactor licensing. The current project seeks to identify and characterise sources of radionuclides within a metal-fuel, pool-type SFR system and identify and characterise barriers to release and transport phenomena. Gaps in the current state-of-knowledge were identified on the base of fuel melting experiments, radionuclide transport experiments and past accidents occurred in US SFRs (meltdown of 13 FSAs in Sodium Reactor Experiment (SRE) in 1959, meltdown of 2 FSAs and damage of 2 FSAs in Fermi-1 in 1966, failure and melting of experimental fuel capsule in EBR-II in 1967).

WP SO 2: Experimental programmes and operational experiences

CIAE accumulated commissioning and operation experience of main heat transport systems in CEFR. It was reviewed commissioning and operation in 2014, including operational history and results of transient tests, such as 40% Pn load throw-off test, 40% Pn turbine fault test, 40% Pn one-loop cut-off operation test, 40% Pn outage of external grid test, 50% Pn overpower test, 75% Pn load thrown-off test. It was described experience feedback related to modification PLC control system, argon pressure control system in reactor vessel, and bypass system.

Within WP SO 2, Japanese specialists carried out two experiments on substantiation of a Primary Reactor Auxiliary Cooling System (PRACS) used for decay heat removal in JSFR under fully natural circulation (NC) mode condition. Thermal-hydraulic phenomena in heat exchanger region of PRACS were investigated in experiment 1. Experiment 2 was dedicated to modelling NC in PRACS loop after pump stop with small initial NC head in PRACS loop. These experiments were performed at the PLANDTL experimental facility. Obtained experimental results revealed that time duration till full NC development was not affected by initial NC head, but only dependent on pressure loss in PRACS loop.

Within WP SO 2, KAERI started work on design (FY2015-2016) and construction (FY2018) of Sodium Integral Effect Test Loop STELLA-2 (Figure 3.37), and implementation of integral experiments (FY2018-2019) for substantiation of PGSFR safety. Deliverable of 2015 includes description of potential scope and design strategy for STELLA-2, review of important simulated phenomena, applied scaling method, prerequisite requirements, schematic, solid modelling, and site information. STELLA-2 of height scale 1:5 has 18 ton of sodium inventory and 1 MW of core power electrical simulators. It will provide modelling the original temperature distribution in heat removal loops for various transients and accidents:

- total loss of flow accident (LOF);
- loss of feedwater accident or steam generator failure (LOHS);
- PHTS pump discharge pipe break;
- total loss of DHR, including potential in-vessel retention.

In 2015, Russian specialists submitted results of experiments on sodium boiling carried out at the AR 1 test facility (Figure 3.38). Experiments were performed at test section with seven-rod FSA model for both modes with forced and natural coolant circulation. Obtained experimental data showed quite stable sodium boiling during approximately five minutes without DNB for both forced and natural circulation modes. The experimental data will be used for verification of COREMELT code designated for analysis of severe accidents in SFR core.

Within WP SO 2, US specialists submitted results of activities related to construction of the Natural Convection Shutdown Heat Removal Test Facility (NSTF) at Argonne National Laboratory (Figure 3.39) and air-based testing programme. This large-scale test facility is designated for passive heat removal testing. Deliverable of 2015 contains overview of the NSTF objectives, NQA-1 programme, facility overview, including heated cavity, outlet plenum, fan loft design, facility configurations, testing parameters of interest, facility characterisation, baseline test procedure, baseline behaviour, performance of block risers, and air-based testing results. These results will be used for supporting code validation.

Figure 3.37: **Principal scheme of STELLA-2**

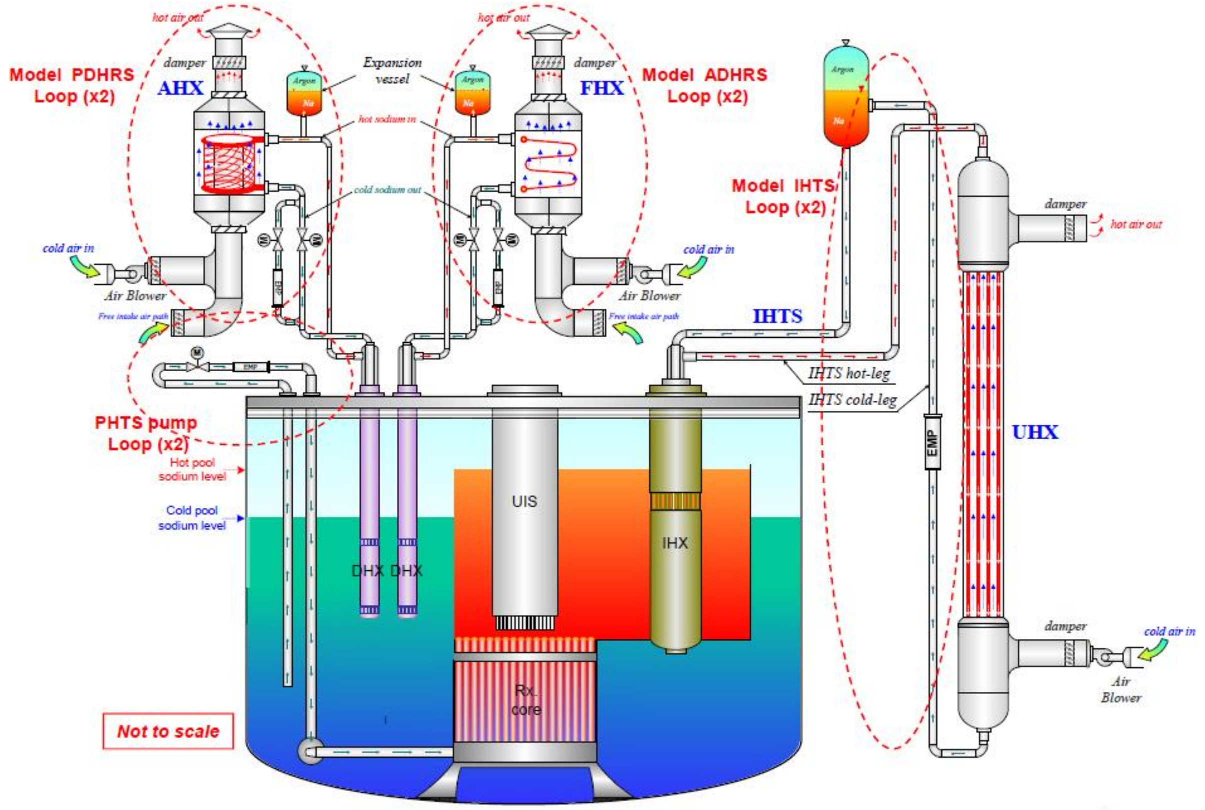


Figure 3.38: **Views of the AR-1 test section during sodium boiling experiments**

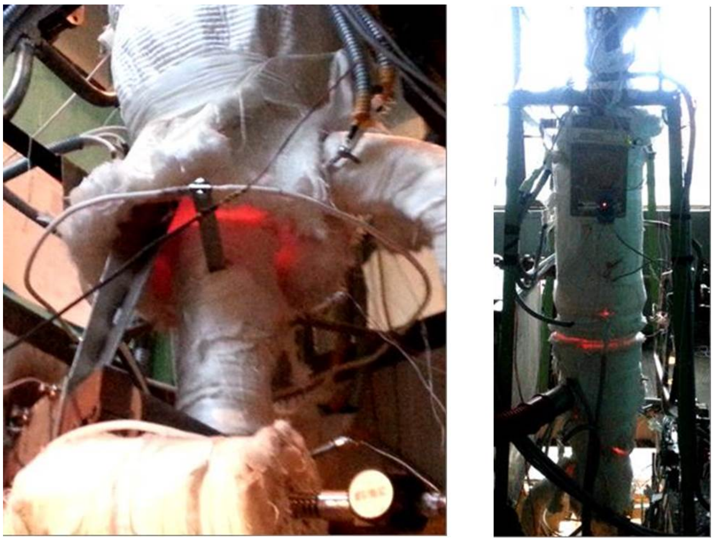


Figure 3.39: **View of the Natural Convection Shutdown Heat Removal Test Facility (NSTF)**

WP SO 3: Studies of innovative design and safety systems

A new methodology has been developed by French specialists to provide a multicriteria SFR core mechanical design optimisation. This methodology takes into account the effects of fuel subassembly design options, manufacturing tolerances, including geometrical uncertainties, and material laws (thermal dilatation, irradiation swelling and creep laws) on the static mechanical equilibrium of SFR core and gives less conservative and more realistic results. It was applied to optimisation of a CFV core (*coeur à faible coefficient de vidange*, or low void effect core) that permitted to make some recommendations for improving design of fuel sub-assemblies, in particular, high pads flexibility provides minimisation of handling forces during refuelling and spacer pads located close to the top of fuel pins promote pads effect.

Assessment of impact of use of MA-bearing fuel on the transient behaviour of the ESFR has been done by Euratom. Analyses were performed for beginning-of-life (BOL) and end-of-life (EOL) condition of the optimised ESFR core with minor actinides. The ESFR core has been optimised in order to reduce the sodium void reactivity (CONF2). Main modifications concern the axial core layout: higher sodium plenum, upper absorber layer, no upper axial blanket, shorter upper gas plenum, lower axial blanket. Two core options have been investigated with different Am contents homogeneously introduced in the core and in the blanket region:

- CONF2 with 2%wt. Am in the lower axial blanket and with 1.9%wt. Am homogenously loaded in core.
- CONF2-HOM4 with 4%wt. Am in the lower axial blanket and with 3.8%wt. Am homogenously loaded in core.

The impact of MA loading on transient behaviour of the ESFR core was studied by using the thermal-hydraulic system code SPECTRA based on point kinetics model with pre-calculated reactivity data. Both unprotected and protected transients were simulated, including coast-down of all secondary pumps, LIPOSO (leakage from diagrid to cold pool), doubling of core bypass flow, loss of feedwater to all steam generators, runaway of grouped control rods, coast-down of all primary pumps, station blackout.

The activities on creation of common projects within the SO PA were initiated. Four key safety topics have been defined for potential common projects:

- natural circulation in sodium systems (first priority):

  - design issues: thermal stratification, flow redistribution or reversal, freezing, thermal stress;
  - evaluation methods: phenomena identification and ranking table (PIRT), model selection, plant-scale validation, uncertainty quantification;
  - fundamental models: heat capacity, pressure loss, and property correlations; experimental measurement techniques.
- reactivity control system options:
  - hydraulic, fusible devices, curie-point, GEMs, ARC, etc.
- ex-vessel cooling system options:
  - RVACS (air NC), forced oil convection, modelling approaches, etc.
- sodium boiling experience:
  - timing and location, stability, codes and methods, experiments.

It was decided to adjust mechanism of implementation of the common projects on the base of the key safety topic related to investigation of the natural circulation in sodium systems and afterwards to spread this practice to other common projects. Progress achieved in implementation of the common projects will be discussed each year.

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 2015, fuel and material process development and property determination, irradiation test preparation and implementation, PIE as well as calculations of fuel behaviour under irradiation, have continued regarding oxide and metallic fuel-based systems. In particular, PIE and performance analysis for minor actinide oxide fuels irradiated up to various burn-ups in the ATR and OSIRIS reactors have continued. Analysis of minor actinide and rare earth containing metal fuels irradiated in the ATR continued. PIE results of U-Zr-type fuels from the 1st HANARO irradiation test are being collected and analysed as preparation work has continued for an irradiation test up to a medium burn-up in a 2nd HANARO irradiation test. The effect of the oxygen potential on the thermophysical properties of oxide fuels as well as the corrosion resistance of minor actinide-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 while preparation for evaluation of the materials irradiation tests advanced. Finally, approval of all SFR advanced fuels signatories was obtained such that China and Russia joined the advanced fuels arrangement.

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/United States, 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 alternative advanced energy conversion system to the conventional steam Rankine cycle system. Details of each study are stated as follows.

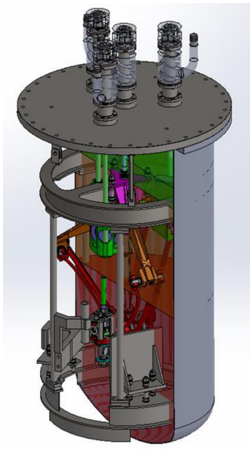
Inspection technologies

The first investigations of under sodium viewing (USV) using ultrasonic transducers were performed; in particular, the definition of the quantitative objectives and the dedicated transducers (specifications, single element, phased array, and characterisation). The first experimental results in simulant fluid (water) have been realised. In addition, the development of experimental tools to achieve in sodium testing started (conception and realisation of a robotic arm able to move the transducer in a hot liquid sodium environment [Figure 3.40]).

Studies on maintainability and reparability for JSFR were done with some analysis of the access route, inspection concepts for core support skirt and lower plenum, improved measures of the primary piping in RV, enlargement of the clearance between GV and RV, pump-integrated-type IHX, design improvement of PHX, primary main piping, modification of the total plant design, structural analysis of the reactor structure, and thermal-hydraulic analysis of pump-integrated-type IHX.

Concerning feasibility of test for ranging with waveguide ultrasonic sensor, tests for ranging with such sensor have been realised. It was demonstrated that targets located more than 1.1 metre from the sensor were well detected. This action was followed by performance enhancement of ranging waveguide sensor, development of ranging inspection software, design and construction of ranging.

Figure 3.40: **Robotic arm for under sodium viewing**



Repair technologies

In this field, the repair remote technologies using laser scouring techniques were investigated with particularly good results. on metallic wall wetted by a liquid. Moreover, the cleaning (Na scraping) feasibility by heating and machining feasibility by steel evaporation were confirmed in the studies.

Supercritical CO_2 Brayton cycle

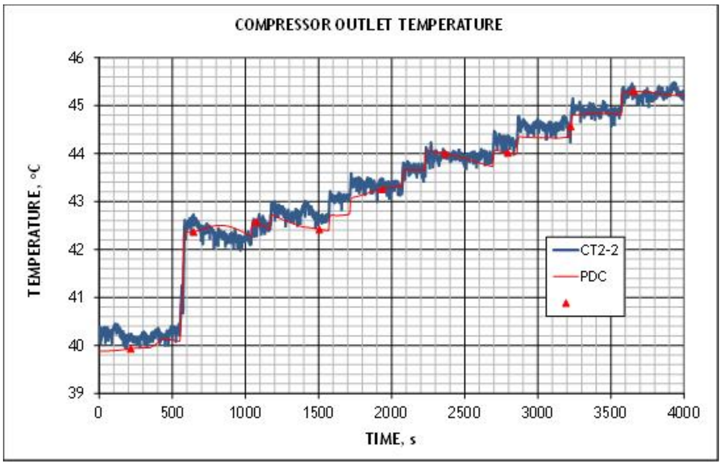
The CFD simulations of a SC-CO_2 compressor using the STAR-CCM+ code have been continued for different operating conditions. Comparison to some experimental data obtained on the TITECH compressor has been performed. Different compressor designs have been evaluated. To establish performance maps, several modelling approaches have been tested (ideal gas, Barber-Nichols Inc. and an adaptation of this new CFD approach). It was confirmed that the CFD simulation of the S-CO_2 compressor predicts temperatures and pressures in agreement with the experimental data (Figure 3.41). Based upon the results, it is recommended that the compressor inlet temperature measurements should be replaced by inlet density measurements for the next experimental development.

Concerning the thermodynamics study of the SC-CO_2 cycle, including analysis of different configurations, different layout for an SMR and SFR applications were evaluated. For both applications, an optimised layout has been proposed. Moreover, the study on cavitation and bubble dynamics in liquid CO_2 near the critical point was continued, including a SC-CO_2 cycle in the condensing mode, modelling of the bubble collapse, and simulations of the bubble collapse in liquid CO_2. It was demonstrated by the study that there is no risk of damage due to cavitation.

The development and application of the Plant Dynamics Code to advanced SFR concepts has also continued. The plant dynamics code (PDC) has been extended for application to the Sandia National Laboratory small-scale SC-CO_2 loop and the results were compared to the test data. Good agreement was obtained between the PDC and the experimental results. This contributes to confidence in code predictions for full-scale S-CO_2 Brayton cycle power converter designs. The Bechtel Marine Propulsion Corporation (BMPC) small-scale SC-CO_2 Integrated System Test (IST) was also modelled (including steady state, transient, IST control…) and good agreement was also obtained.

The development and the construction of the sodium-CO_2 interaction tests loop has been finalised in 2014 and preliminary plugging test data as wel as preliminary wastage test data were produced. The experimental programme has been interrupted due to technical issues on the loop. (unexpected pressure peaking during high-pressure CO_2 injection into liquid sodium leading to a large amount of sodium aerosol causing gas vent line clogging). The facility has been maintained and modified. In parallel, a computational code has been developed to evaluate system transients with a sodium-CO_2 interaction event. Investigation of fundamental reaction mechanisms through experimental study has been initiated in order to include features of mass diffusion effects as well as kinetics.

Figure 3.41: **Comparison of experimental data and PDC simulation results**



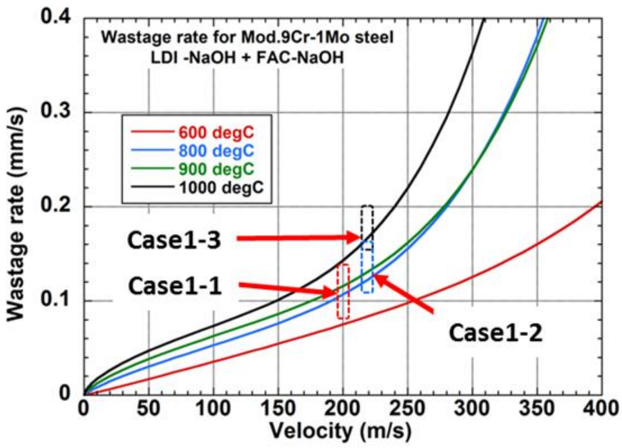
Steam generators

The experimental results obtained on thermal transient testing and strength evaluation of a tube-sheet made of Mod. 9 Cr-1 Mo steel were reported. Destructive examinations including results of liquid penetrant tests make cracks appear at the hole edges. Complementary observations using SEM (cracks and fracture surface examinations) were performed. The failure mode has been identified: creep-fatigue crack initiation and propagation. The place of crack initiation was localised, and it was established that the temperature histories during the thermal transients were caused by the direct stream of inflowing sodium approaching the upper surface of the tube-sheet. The thermal stress analysis and strength evaluation was numerically evaluated. Creep-fatigue lifetimes estimated by some methods agreed well with the observed crack distributions for outermost holes.

The codes to simulate tube failure propagation in an SFR steam generator have been developed based on a mechanistic methodology with development of mechanistic sodium-water reaction analysis codes and development of safety evaluation codes. In parallel, experiments and construction of a database (to utilise for safety assessment for new regulations/criteria) was also proposed.

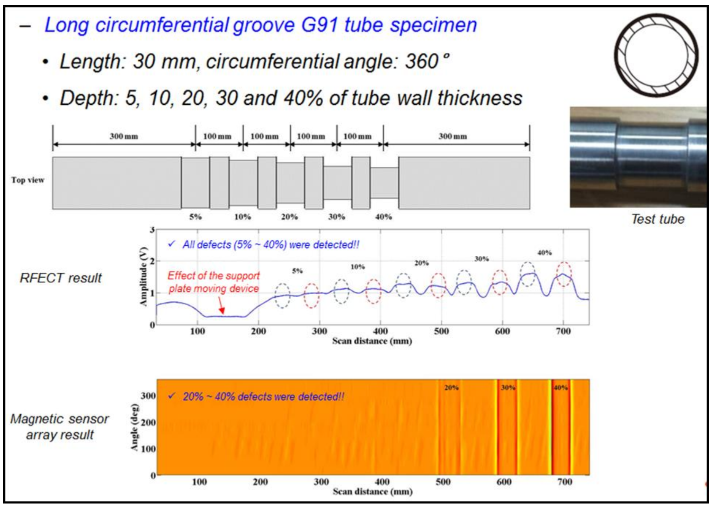
Concerning wastage evaluations, the construction of a new target-wastage correlations combining a composite oxidation-type corrosion with flow (COCF) due to sodium compounds with liquid droplet impingement erosion (LDI) was performed. The applicability of the new wastage correlations using sodium-water reaction tests was validated: the new wastage curve can predict appropriately the SWAT-3R test data (Figure 3.42).

Figure 3.42: **New wastage curve (COCF+LDI)**



The development of an inspection sensor for steam generator tubes progressed: it was proposed to use a combined steam generator tube inspection sensor (using remote field eddy current testing [RFECT] and magnetic sensor testing). A prototype was developed together with the associated signal processing unit and signal analysis software. A test facility for sensor performance in air was assembled. Preliminary performance tests of the steam generator tube inspection system were carried out (long distance signal transmission test, damage detection test) (Figure 3.43).

Figure 3.43: **Example of results obtained concerning defect detection on steam generator tubes by combined sensor**



Global Actinide Cycle International Demonstration project

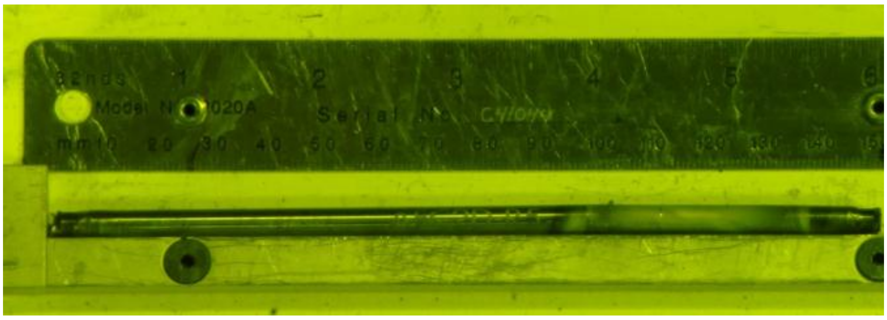
The Global Actinide Cycle International Demonstration project aims to show that SFR can effectively manage all actinide elements, including uranium, plutonium, and minor actinides (MAs: neptunium, americium and curium) by transmutation. The project includes fabrication and licensing of MA-bearing fuel, pin-scale irradiations, material property data preparation, irradiation behaviour modelling and PIEs, as well as transportation of MA raw materials and MA-bearing fuels. Bundle-scale demonstration will be included.

The irradiation behaviour of the MA-bearing fuel irradiatedin the Joyo reactor, such as americium migration, was analysed and investigated in detail based on the PIE results for irradiation behaviour modelling. To evaluate the pellet structural change of MA-bearing MOX fuel, pore migration model was improved by introducing MA-MOX vapour pressure calculation in consideration of O/M dependence. It was shown that the central void formation observed in Am-bearing MOX irradiated in Joyo was simulated well by this improvement.

The reference fuel (U, Pu)O_2 was the subject of the first measurements at CEA and ITU of melting temperature and thermal diffusivity for two O/M=1.94 and 1.98. The measurements of vapour pressure and heat capacity on these reference samples are planned early in 2016. R&D on (U,Pu,Am,Np)OX fabrication in CEA was presented and the organisation of the sample shipment from Cadarache to ITU is ongoing. Next year a portion of the properties measurements programme on minor actinide-bearing fuel will be completed at ITU.

In parallel, the post-irradiation examinations of the MA-bearing fuel $(U_{0.75},Pu_{0.2},Am_{0.03},Np_{0.02})O_{1.95}$ irradiated in the AFC-2D irradiation in ATR were achieved in INL with visual examination, neutron radiography, gamma spectroscopy, dimensional measurements. Some destructive examinations are in progress, mainly : fission gas analysis, metallography, microhardness and burn-up analysis.

Figure 3.44: **Visual exams at INL of AFC-2D Rodlet 5 $(U_{0.75},Pu_{0.2},Am_{0.03},Np_{0.02})O_{1.95}$ Irradiated at <25 at%**



3.6 Very-high-temperature reactor (VHTR)

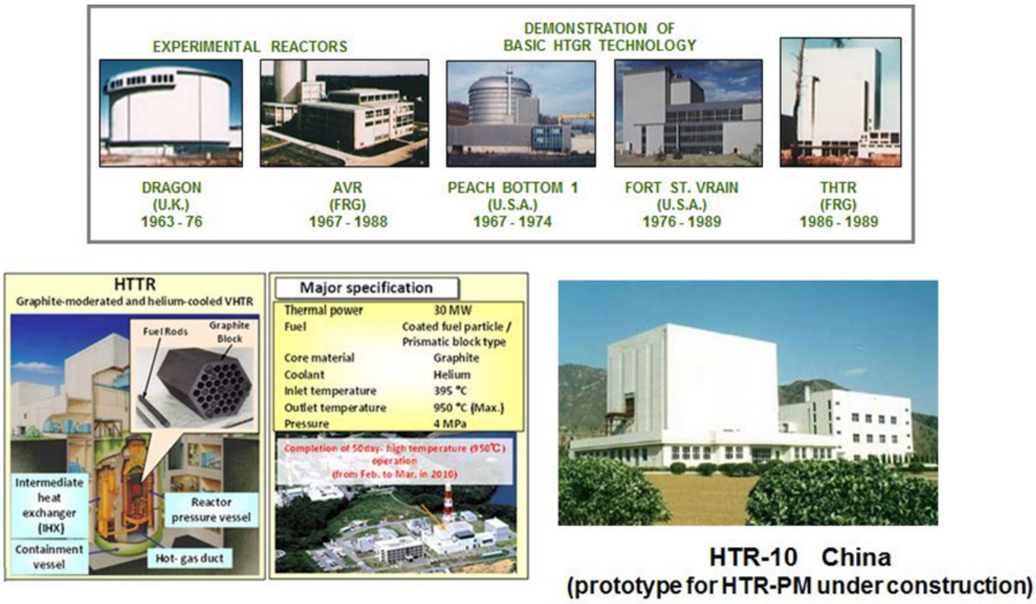
Main characteristics of the system

The very-high-temperature reactors are the descendants of the high-temperature reactors developed in the 1970s-1980s (Figure 3.45). They are characterised by a fully ceramic-coated particle fuel, the use of graphite as neutron moderators, and the inert gas helium as coolant, resulting in passive decay heat removal capability, and thus very attractive inherent safety performance. The VHTR is also very well adapted to large-scale non-electric applications for a variety of industrial uses, e.g. in the chemical industry or for bulk hydrogen production which can then be used for many other purposes. In this manner, nuclear energy can substitute for fossil hydrocarbons (gas, oil and coal) both as fuel and feedstock for the chemical industry (e.g. natural gas for fertiliser production can be replaced by nuclear process heat and nuclear generated hydrogen).

Use of helium as coolant and ceramics as fuel and moderator enables helium outlet temperatures up to 950°C with moderate need for innovative structural materials. Such temperatures have been tested over longer time periods in two reactors. Different market studies have shown that the market potential for nuclear process heat is very significant in most industrialised countries already for helium outlet temperature of approximately 750°C and further increases for higher temperature applications.

The technology for the VHTR has been demonstrated in former high-temperature gas-cooled reactors such as the US Peach Bottom and Fort Saint-Vrain plants as well as the German AVR and thorium high-temperature reactor (THTR) prototypes, followed more recently by the Japanese high-temperature test reactor (HTTR) and the Chinese HTR-10 test reactors. They represent the two baseline concepts for the VHTR core: the prismatic block-type and the pebble bed-type. The fuel cycle will initially be once-through with low-enriched uranium fuel and very-high-fuel burn-up, but the U-Pu and Th-U fuel cycles are also possible. Solutions need to be developed to enhance the management of the back end of the fuel cycle, and the potential for a closed fuel cycle should be established beyond lab-scale. Although various fuel designs can be considered, all concepts rely on tristructural isotropic (TRISO) coated particle fuel as the common denominator. This fuel consists of small particles of nuclear material (most often UO_2 or uranium oxycarbide [UCO]), surrounded by a porous carbon buffer, and coated with three layers: pyrocarbon/silicon carbide/pyrocarbon. As demonstrated in numerous tests, this coating represents an effective first barrier against fission product release under normal, accident and repository conditions.

Figure 3.45: **Evolution of HTGR development since the 1960s**



Two HTR reactors, AVR and HTTR, were already operated at temperatures up to 950°C for extended periods of time confirming that the technology can supply nuclear heat and electricity over a wide range of core outlet temperatures between 700 and 950°C, or more than 1 000°C in future. The currently available high-temperature alloys used for heat exchangers and metallic components bound the current upper temperature limit to 950°C.

The original target for the GIF VHTR was set at 1 000°C or above because one of the main drivers for the technology was large-scale bulk hydrogen production with the iodine-sulphur process. This process consumes heat at 850°C and thus, accounting for temperature cascades in heat exchangers, requires about 1 000°C at the reactor outlet. For such ambitious applications, innovative materials such as new super alloys, ceramics and compounds need to be developed and qualified.

In the meantime, studies in several of the GIF signatory countries have confirmed the existence and development potential of a significant market for lower-temperature applications, especially process steam below 600°C which requires rather conservative primary helium outlet temperatures of approximately 750°C. This steam can then be used either for electricity generation (commonalities with the most advanced coal-fired power plant machinery is possible) and/or for industrial applications where the steam is used either as feedstock for chemical reactions or as a heat carrier. Therefore, for power conversion, several nearer-term VHTR projects use steam cycle technology whereas direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles are longer term.

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. As examples, the HTTR will provide a platform for coupling advanced hydrogen production technologies with a nuclear heat source up to 950°C and the HTR-10 is currently running a test with melt-wire pebbles to confirm the temperature distribution profile.

Several plant vendors and national laboratories in China, the United States, Korea and Japan run projects which make the technology advance, e.g. HTR-PM, NGNP, NHDD, SC-HTGR and GTHTR300C. Specifically the construction of the HTR-PM demonstration plant (two pebble bed reactor modules of 250 MWth each with a common super heated steam turbine generating 211 MWe) is progressing quickly (Figure 3.46). The helium outlet temperature will be 750°C, which is well within the performance window of commercially available materials. It is planned to connect the HTR-PM demonstration plant to the grid in 2017, which will represent a major step towards demonstrating this generation IV technology.

Figure 3.46: **HTR-PM roofwork in September 2015**

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 (MAT) PA, which addresses graphite, metals, and ceramics and composites, was signed by implementing agents from Canada, Euratom, France, Japan, Korea, South Africa, Switzerland and the United States by 16 September 2009, and has been effective since 30 April 2010. China initiated the process for joining the materials PMB in 2010. South Africa's withdrawal has become effective from the material PA 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 were finalised and approved by the VHTR SSC in 2014. Details of the signature process for signing the amended PA were agreed upon in early 2015 and it is expected to be signed by the signatories in 2016.

The Hydrogen Production (HP) 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 October 2011. The further update of the project plan is expected in 2016.

The computational methods validation and benchmarks provisional project management board met twice in 2015, with participants from China, Korea, Switzerland, the United States and Euratom. The new project plan will be reconstructed and finalised in 2016.

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

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, cogeneration, 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 TRISO-coated particles which are the basic fuel concept for the VHTR. R&D aims to increase the understanding of the standard design (UO_2 kernels with SiC/PyC coating) and examine the use of uranium oxycarbide (UCO) kernels and possibly advanced coatings for enhanced burn-up capability, and minimal fission product release under operational and accidental conditions. This work involves fuel characterisation, post-irradiation examination, safety testing, fission product release evaluation, as well as assessment of chemical and thermomechanical 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 MAs 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. The HP project currently envisages three main processes for hydrogen production from water featuring significantly higher efficiencies compared to classical low-temperature electrolysis. These are the iodine-sulphur process (China, Japan and Korea,), the copper-chlorine process (Canada) and the high-temperature steam electrolysis (China, France, the United States and the EU). 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 are being 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 is also under investigation and design-associated risk analysis is being performed covering potential interactions between nuclear and non-nuclear systems. Processes are examined in terms of technical and economic feasibility either in dedicated or cogeneration mode. The aim is to reduce operating temperature requirements in order to make these processes compatible with other reactor systems and non-nuclear heat sources, such as concentrated solar power.
- For HP, two main processes for splitting water were originally considered: the sulphur/iodine thermochemical 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 thermochemical cycle and the hybrid sulphur 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. Thermochemical 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.

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.

Main activities and outcomes

Fuel and Fuel Cycle (FFC) project

The 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.

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 an advanced coating for enhanced burn-up capability, minimised fission product release, and increased resistance to core heat-up accidents (above 1 600°C). Fuel characterisation work, PIEs, safety testing, fission product release evaluation, as well as the measurement of chemical and thermomechanical material properties in representative conditions is feeding 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 irradiated graphite needs to be defined (direct disposal vs. decontamination and recycling). Although a once-through fuel 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 3.47.

Figure 3.47: **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

Status of ongoing FFC activities

During 2015, significant work was accomplished in the areas of irradiation and PIE, characterisation, safety testing, and back-end fuel cycle issues.

Irradiation and PIE

In the United States, post-irradiation examination of the advanced gas reactor (AGR)-2 irradiation that began in June 2010 was completed in October 2013. Results of the metrology are complete and dimensional changes in compacts are similar to that observed in AGR-1.

In the United States, PIE of AGR-1 is complete. The PIE of high-flux reactor (HFR) European Union (EU)-1 test containing Chinese and German fuel irradiated at typical pebble bed conditions is also nearing completion in 2015. PIE following accident furnace testing at 1 600 and 1 800°C indicates cracks in the buffer, IPyC and SiC layers similar to that observed in US AGR-1 TRISO fuel. Fission product distribution measurements indicate the presence of Cs, Sr, Pd and Ag at the inner SiC surface. Kr is found in the porosity of the kernel. PIE of Chinese pebbles is anticipated to begin in 2016.

In Korea, the post-irradiation examination of the first irradiation of TRISO fuel in the high-flux advanced neutron application reactor (HANARO) began in July 2013 and was completed in March 2014. Detailed post-test analysis of the service conditions was conducted in 2015. Peak burn-up was estimated to be ~4% FIMA. Non-destructive examination (gamma scanning, X-ray tomograph) was completed in 2014. Fission gas analysis of capsule contents occurred in 2015. The results were very low suggesting no particle failures.

Additional PIE is planned in 2016.

Fuel attributes and material properties

In the EU, the PYrocarbon 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 from France are complete. Plans have been established for Korean surrogate particles but are awaiting funding decisions in the parties.

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 vapour in the air.

The leach burn-leach round robin has begun. 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 has developed methods to crack and drill a small hole in the TRISO coating to create defective particles. These defective particles will be sent to China. China is currently awaiting approval to ship the samples to the United States 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.

The fuel performance accident benchmark for TRISO fuel performance codes has begun. The United States distributed a report that provided all of the input data needed for the benchmark and a schedule was established for the work. The first set of calculations have been completed and more planned in 2015. 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.

Safety testing

In the EU, accident safety testing of HFR-EU-1 pebbles was completed in 2015 with some delay because the furnace had to be decontaminated after high caesium release in a previous test. Accident safety testing of pebble HFR-EU-1 is planned in 2016.

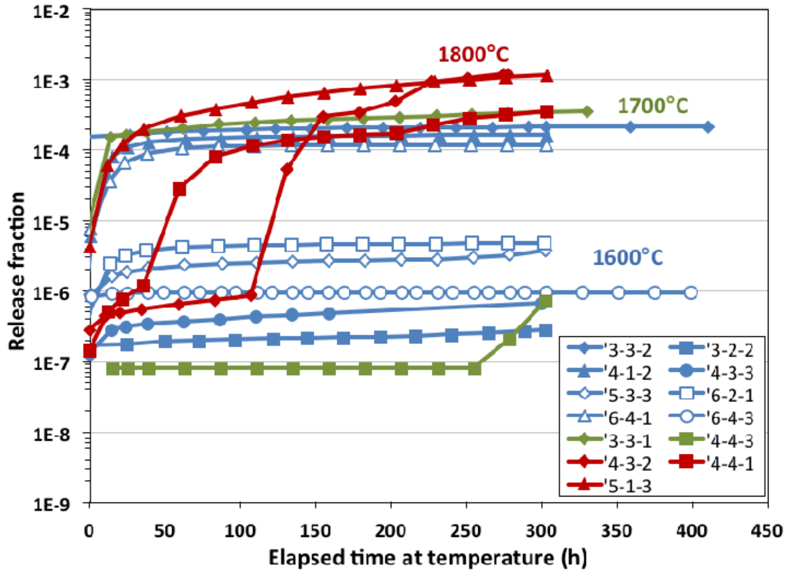
In Korea and China, the conceptual design of accident heating furnaces is underway but has been delayed because of technical and resource issues in each country. In China, conceptual designs of key pieces of PIE equipment necessary to analyse TRISO fuel have been completed. In Korea, the scope of the furnace, which was originally planned to cover gas reactor, fast reactor and light water reactor fuels, was recognised as too ambitious. Instead, a small furnace that can initially test simulants will be procured and installed in a laboratory to gain experience.

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 were irradiated over a spectrum of burn-up, 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 was gathered. PIE of the capsule began in 2015 with initial capsule gamma scanning and disassembly.

Both Korea and Japan are continuing out-of-pile oxidation experiments with several graphite materials and SiC TRISO-coated (dummy) fuel particles under air ingress accident conditions. 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 were studied in the Naturzug im Core mit Korrosion (NACOK) facility.

Figure 3.48: **[134]Cs release fraction under simulated accident conditions from 15 compacts irradiated in AGR-1: Excellent results from good coatings and UCO chemistry**



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 burn-up 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. From recent results obtained by the signatories it seems that ZrC is actually not a good choice and will not be pursued any longer as an alternative high-performance coating option.

Waste management and other fuel cycle options

This area covers three issues:

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

In the EU, three tasks were completed in 2015:

- corrosion of coatings under waste disposal conditions;
- model development for long-term performance of TRISO-coated particle fuel;
- safety case for waste management.

Certain results from the related European projects CARBOWASTE (completed) and Carbon-14 Source Term Project, or CAST (running) are of interest to the other parties, and permission to share deliverables with the FFC project is being sought by the EU member.

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 current five-year plan 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 exercise 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 VHTR FFC project has produced many positive results. The success has led to an ambitious second five-year plan (2012-2017). A few of the deliverables were not completed during the last work plan and have been transferred to the second five-year work plan. Most deliverables are on schedule.

Materials

Although the term of the original Materials Project Plan (PP) was completed in 2012, the Materials PA continued under a draft extension of the PP through 2015, reflecting expected changes to the signatories of the PA (the withdrawal of Canada and South Africa's PBMR and the addition of China). Contributions for the extension of the PP through 2015 were developed by the remaining six signatories (the European Union, France, Japan, Korea, Switzerland and the United States), to which the one from China was added. The extended and augmented contributions were compiled into a revised PP, approved by both the PMB and the VHTR SSC. The revised PA is expected to be signed in 2016.

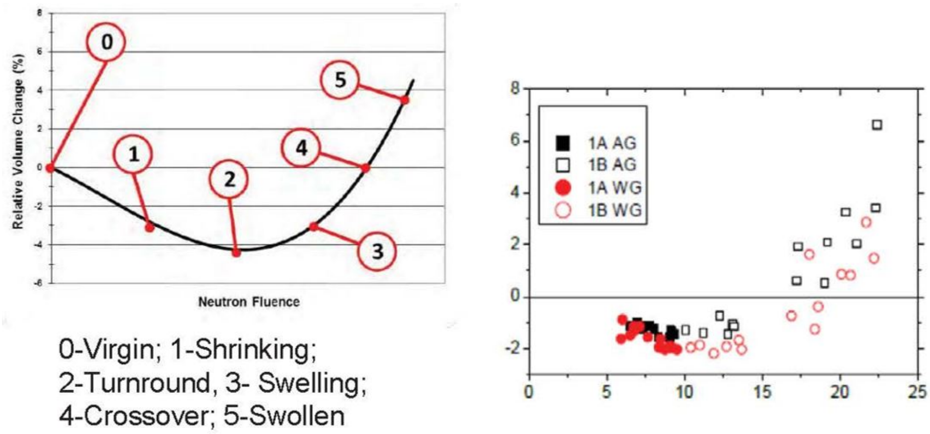
A thorough review was made of all the high-level deliverables (HLDs), as part of the extension of the PP. All HLDs scheduled for completion prior to the end of 2014 were completed, and those due by the end of 2015 are either on track for completion or will be extended as part of the expected further extension of the PP for an additional three years through the end of 2018, as agreed upon by the PMB. Revised contributions from all anticipated signatories for the extended PP are currently being developed.

By the end of 2015, over 340 technical reports describing contributions from all signatories will have 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 for graphite. Additionally, supporting laboratory data records for thousands of tests are also being uploaded into the handbook to allow for detailed evaluation of the experiments among individual researchers.

In 2015, 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. An example of irradiation studies on graphite from the work of the Joint Research Centre in the Europeon Union is provided in Figure 3.49. It shows results from the Innograph 1C experiment of relative volume change for a variety of graphites examined as a function of irradiation dose.

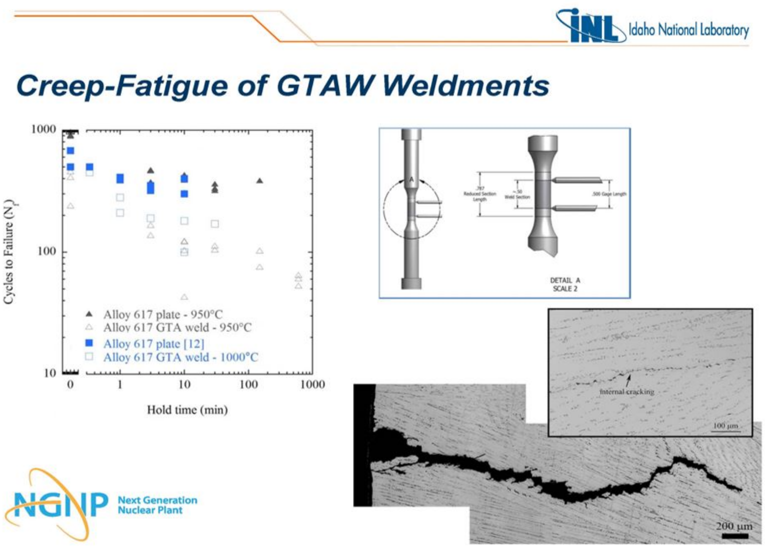
Figure 3.49: **Relative volume change for a variety of nuclear graphites as a function of irradiation dose (dpa)**



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 mock-up preparation, and subsequent testing.

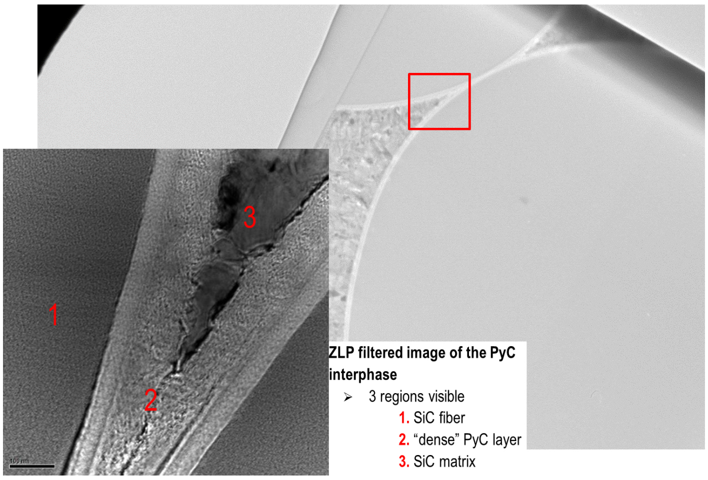
Significant studies on the thermophysical, mechancal, creep and creep-fatigue, and fracture properties of alloy 617 were performed as part of the development of the information required to include it in the ASME Code as an additional material for use in construction of high-temperature reactor components. This has resulted in the completion of the formal submission of the code case to the ASME to allow the use of alloy 617 to be used as a construction material for high-temperature reactors. An example of mechanical properties studies on alloy 617 from the work of the Idaho National Laboratory for the US Department of Energy Research illustrates creep-fatigue behaviour studies of weldments versus basemetal at 950-1 000°C in Figure 3.50.

Figure 3.50: **Effect of hold time of creep-fatigue behaviour of Alloy 617 weldments and basemetal at 950-1 000°C**



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 thermophysical and 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. An example of transmission electron microscopy studies performed at the Paul Scherrer Institute in Switzerland in the development of advanced SiC-SiC composites is shown in Figure 3.51.

Figure 3.51: **Bright field TEM image and zero loss filtered image (bottom left) of high-performance SiC/SiC composite (Hi-Nicalon fibres in a chemical vapour infiltrated SiC matrix, samples from SEP/SNECMA) (L. Fave, PSI and EPFL)**



Hydrogen production

The HP Project Arrangement has been signed by Canada, France, Japan, Korea, the United States and Euratom. China has been a candidate for joining the PMB. Hydrogen was the initial driver and still is one of the major potential applications for the VHTR and other Gen IV nuclear reactors, especially in countries where natural gas is expensive.

Active participation in the HP PMB has evolved considerably. While France was absent between 2010 and 2014, it is active again contributing results on high-temperature electrolysis. The US participation has recently been less active. The contributions from Asian countries (mainly Korea, Japan and China as a candidate) and from Canada have remained consistently strong.

The main activities overseen by the HP PMB deal with the thermochemical cycles (sulphur iodine [SI] cycle, copper-chlorine [Cu-Cl] cycle) and high-temperature steam electrolysis (HTSE).

Japan, Korea and China are strongly involved in SI developments and testing. Japan plans to connect a hydrogen production plant to the HTTR reactor as the process heat source.

Korea has engaged in an experimental programme on a Sulphur Iodine Integrated system producing 50 NLH2/h. After a series of separate tests for the Bunsen reactor section unit and a partial integration test in 2014, Korea succeeded in continuous operation of the integrated facility with almost constant hydrogen production rate. Korea is currently preparing 72-hour continuous operation with the same integrated facility before upscaling the installation for higher throughput at 1 000 NL/h. Korea's success in achieving good performance with the Bunsen reactor is a major achievement in the development of the SI process.

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 H_2 production rate of 60 NL/h for 86 hours. During the operation of this facility, major key parameters of three sections (Bunsen, SA, and HI) were monitored and found to be fairly steady, demonstrating a significant achievement. At Tsinghua University, developments in the area of HTSE has continued with experimental work on 1 to 10 cell stacks. Their work plan includes KW-class multi-stack modules and coupling with the HTR-10.

In Canada, CNL (formerly AECL), in collaboration with the University of Ontario Institute of Technology (UOIT), is developing the Cu-Cl Cycle. CNL is focusing on the electrolysis step while UOIT is developing the other steps and the overall integration of all the steps of the cycle. Development of the electrolysis step includes suitable membranes, designing an optimal cell, investigation of cell operation at higher temperatures (~80°C) and pressures (~7 atm). The latest work had also focused on minimising any copper crossover from the anolyte to the catholyte. Based on the understanding of earlier tests, a new double membrane cell (DMC) electrolyser design for CuCl/HCl electrolysis was proposed, which mitigates copper species crossover. The experiment with the DMC showed that it can maintain copper concentrations in the cathode at low levels, whereas in case of a single membrane cell (SMC) the copper concentration constantly increased with time. This DMC was shown to perform well for 1 600 h. A conceptual integrated-Cu-Cl cycle design diagram has been achieved. Significant advancements at UOIT have also been achieved with the other steps, namely water removal from aqueous Cu(II) chloride, hydrolysis reaction that produces the copper oxy chloride and its decomposition.

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 and INL (United States). Experimental work is continuing in Canada to improve the performance of the electrolysers at somewhat lower temperatures, thereby possibly extending the lifetime of such components.

Experimental work has also begun in Canada to study ways of mitigating the transfer of tritium produced in the core of the VHTR to the hydrogen production process. Tritium permeation through various materials is being characterised as part of this effort.

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-electrolyse CO_2/H_2O, fuel cell). The world 1st solid oxide electrolyser cell (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 Nm^3/h H_2. Good performance was demonstrated with stacks containing 3, 10 or 25 cells. Significant improvement in the durability of low-weight stack was also achieved in their latest demonstrations. Current focus is on pressurised operation up to 30 bars since hydrogen production at pressure can offer significant cost savings for hydrogen supply requirements. Comparison of operating points of alkaline electrolysers, proton exchange membrane (PEM) electrolysers 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. With further improvements in durability of these electrolysers, hydrogen produced can be cheaper than with PEM or alkaline electrolysis, especially when integrated with high-temperature reactors. Work at CEA with CO_2/steam co-electrolysis has also shown good performance of their electrolysers even with 45 (vol)% CO_2 at the inlet, a remarkable demonstration.

Euratom contributes with a 3-kWe-sized pressurised HTSE system which will be coupled to a concentrated solar power source. The system is being designed and manufactured, and co-electrolysis of H_2O and CO_2 at the stack level will be demonstrated also under pressure.

Figure 3.52: **Laboratory scale facility for sulphur-iodine hydrogen production at INET**

Computational methods validation and benchmarks

The CMVB Project Management Board was restarted in 2014. On the 11th provisional PMB meeting, which was held in Weihai, China just before the HTR-2014 conference, the work packages (WP) of the draft PP were identified, and specific member countries were assigned to lead each WP. From then on, provisional members focused on these tasks and the detailed content of work packages of the draft PP.

On 20 April 2015, the 12th provisional PMB was held in the Institute of Nuclear and New Energy Technology (INET), Beijing, China. Three provisional PMB members (China, Euratom and Korea) attended (Japan and the United States were not able to attend). The current status of the CMVB research activities among member countries was presented at the meeting. All tasks and work packages in the draft PP, which had been revised up to the 11th CMVB meeting, were reviewed and updated. Input was received from all participants, including from US and Japanese members. Five work packages were developed, each with task descriptions, schedules, contributors and leaders:

- phenomena identification and ranking table (PIRT) methodology (led by the EU);
- computational fluid dynamics (CFD) (led by China);
- reactor core physics and nuclear data (led by the United States);
- chemistry and transport (led by China);
- reactor and plant dynamics (led by China).

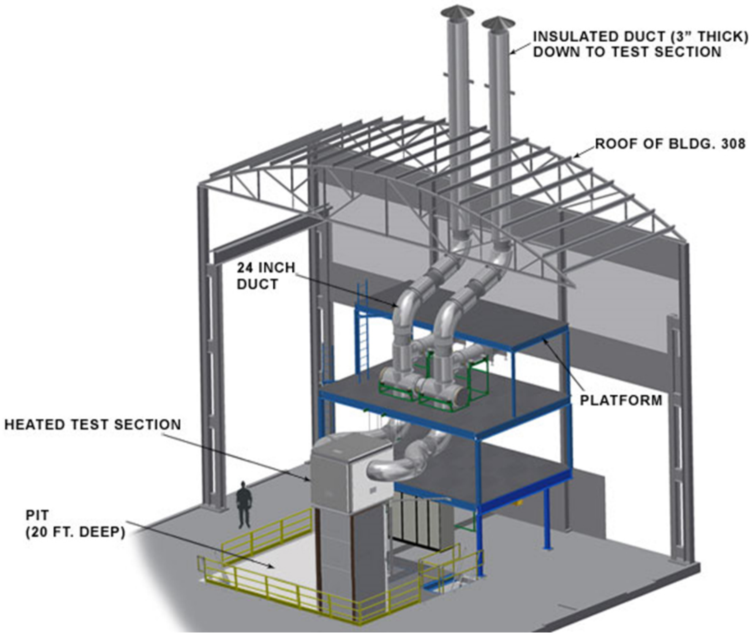
After discussion of the content, KAERI hosted the 13th CMVB PMB meeting in Daejeon, Korea on 2-4 December 2015. Participants from four provisional member countries attended (Euratom was excused). The draft PP containing the members' contributions, as well as the PA, will be finalised for signature in 2016 once the Framework Agreement Extension is signed by all parties.

Past, current, and new test facilities and projects have been proposed as potential resources to carry out the CMVB code development and benchmarking activities. In China, the construction of 16 separate engineering test facilities is almost completed and some of them have already provided essential data for HTR-PM development and code validation. The HTR-10 was restarted to test the major components and system operation. A melt-wire experiment to measure the in-core temperature will be implemented in 2016. The Advanced High-Temperature Reactors for Cogeneration of Heat and Electricity R&D (ARCHER) project (Euratom), focused on HTR demonstration-oriented technology R&D, and was completed in January 2015. Results have been offered to this project. Korea has focused its R&D on improvement and validation of VHTR passive safety features such as the hybrid air-cooled reactor cavity cooling system (RCCS) with water jacket. In the United States, NGNP supported the development of several code systems to characterise and simulate some phenomena. To perform the experimental validation, some test facilities (the High Temperature Test Facility [HTTF], the Natural Convection Shutdown Heat Removal Test Facility [NSTF], MIR, etc.) have been constructed. Data from NSTF experiments is available for validation of air-cooled RCCS models while HTTF experiments are expected to begin in 2016. All these research activities carried out in test facilities and reactors play an important role for verification and validation of computer codes and calculation methods, which will benefit the CMVB work.

Figure 3.53: **Engineering test facility of main helium circulator (ETF-HC) at INET's HTR-PM Laboratory**

Figure 3.54: **Hybrid reactor cavity cooling system (RCCS) test facility at KAERI**

Figure 3.55: **Refurbished Natural Circulation Shutdown Heat Removal Test Facility (NSTF) at ANL to generate data revealing the performance of this passive safety system**



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Chapter 4. Methodology working group reports

The three Generation IV International Forum (GIF) methodology working groups – the Economic Modeling Working Group (EMWG), the Proliferation Resistance and Physical Protection Working Group (PRPPWG) and the Risk and Safety Working Group (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) and its update (GIF, 2014) in terms of economics, proliferation resistance and physical protection, and safety.

4.1 Economic assessment methodology

According to its mandate, EMWG developed methodology for the economic assessment of generation IV systems for the two economic goals stated in the Generation IV Technology Roadmap. The methodology consists of cost estimating guidelines and a software model G4ECONS version 2.0 released for use in 2007. G4ECONS calculates two figures of merit required to assess the generation IV system economic relative to the current generation power reactors:

- levelised unit cost of electricity (LUEC) and other energy products expressed as USD/MWh;
- total capital investment cost (TCIC) expressed as USD/KWe.

The cost estimating guidelines and the G4ECONS are available from the GIF secretariat at NEA, as explained on the GIF website. The EMWG methodology has been extensively used both within and outside the GIF community, by the universities, consulting companies and the IAEA. Several publications demonstrate the use of the EMWG methodology for generation II, generation III and generation IV systems, including cogeneration applications.

The activities of the EMWG were focused on three key areas during 2015:

- collaboration with IAEA on benchmarking of G4ECONS with IAEA's Nuclear Economics Support Tool (NEST);
- development of the next version of G4ECONS;
- seeking collaborations with the system steering committees (SSCs) on the use of EMWG methodology.

G4ECONS was benchmarked against NEST for a generation IV supercritical-water-cooled reactor (SCWR) system, namely, the European high-performance light water reactor (HPLWR) (KIT, 2012). The HPLWR is a 1 000 MWe reactor operating at 500^{0}C with a thermodynamic efficiency of 43.5%. Average capacity factor was assumed to be 91% over the operating life of 40 years. The HPLWR uses a once-through fuel cycle with 8% enriched uranium fuel. The results of this benchmarking study were presented (Sadhankar, 2015) at the GIF Symposium in Chiba, Japan in May 2015. There was a close match of the two figures of merit, namely, the LUEC and the TCIC calculated by G4ECONS and three different versions of NEST.

Further benchmarking activities are underway in collaboration with IAEA for fast reactor systems using closed fuel cycles. Two sets of fast reactor systems, namely a break-even fast reactor (BR = 1) and a burner fast reactor (BR <1), were selected from the IAEA report of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) GAINS project

(IAEA, 2013). Fuel cycle unit costs were based on the data available in the INL Advanced Fuel Cost Basis report (INL, 2009) and the capital and operations costs were assumed to be similar to those for the newly built Gen III+ designs. Initial economic analyses of a 870 MWe break-even SFR-type fast reactor with 12% Pu fuel, 41.4% thermodynamic efficiency, 85% capacity factor and 140 equivalent full power day cycles showed promising results. In this exercise three NEST models/versions were used, namely, v2s3, v4s2, and v4s3. The v2s3 is the model based on Harvard University study (Bunn et al., 2003) for the case of break-even (equilibrium) fast reactors system. The v4s2 is a combination of model v2s3 with the approach developed in the INPRO methodology published in 2008 (IAEA, 2014). Model v4s3 is an extension of model v4s2 for the case of fast reactors operating with conversion rates other than 1 (breeders or burners).

Figure 4.1 shows the similarity of the overall unit levelised costs (LUEC), the levelised unit capital costs (LUCC) and the levelised unit operation and maintenance costs (LUOM) calculated by G4ECONS and the three NEST models. Small differences in the levelised fuel cost (LUFC) are attributed to the minor differences in the calculation of front-end and back-end fuel costs. Figure 4.2 shows similar TCIC calculated by G4ECONS and the three NEST models. Additional benchmarking will be performed for a burner fast reactor in a closed fuel cycle.

Figure 4.1**: Comparison levelised unit costs calculated by G4ECONS and NEST for the equilibrium fast reactor**

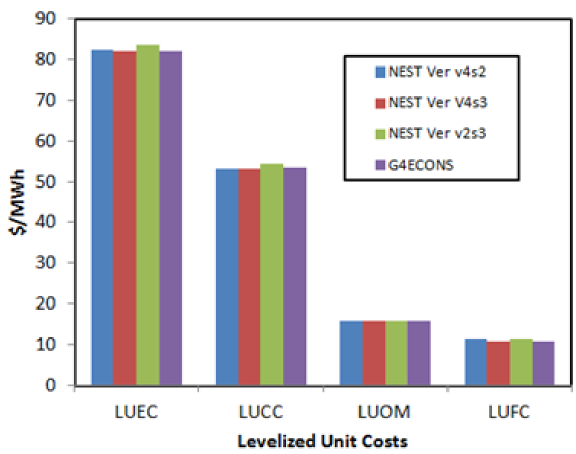
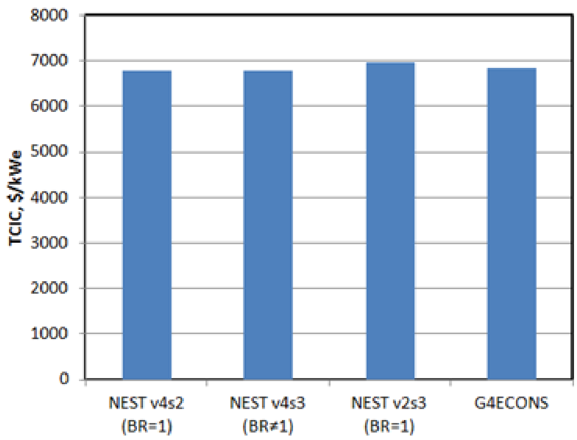


Figure 4.2**: Comparison of total capital investment costs calculated by G4ECONS and NEST for the equilibrium fast reactor**



Future benchmarking activities will include the comparison of hydrogen and thermal energy costs calculated by G4ECONS and other tools.

EMWG presented its methodology and extent of its use at the IAEA Technical Meeting on the Economic Analysis of High-Temperature Gas-Cooled Reactors and the Small and Medium Sized Reactors in August 2015.

The next version of the G4ECONS was released for alpha-testing within the EMWG. This next version, to be named version 3.0, is also Excel-based and has improved user interface. The input data entry is simplified using multiple data entry sheets. The version 3.0 also includes built-in uncertainty analysis capability, which will be useful to assess the impact of simultaneous uncertainties in several cost inputs. Feedback from alpha-testing will inform further improvements to the version 3.0 before releasing it for beta-testing.

The EMWG maintains contacts with the SSCs through participation of the representatives in the Experts Group (EG) and the Policy Group meetings. Following the successful joint meeting of the EMWG and the SCWR SSC in 2014, an EMWG member participated in the VHTR SSC meeting in May 2015 and presented the earlier work done on economics of the GT-MHR and hydrogen production. VHTR SSC members expressed interest in pursuing collaboration on the economic assessment of the hydrogen production. EMWG will continue to seek collaborations with the SSCs on the use of the EMWG methodology and solicit feedback for improving the methodology.

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4.2 Proliferation resistance and physical protection assessment (PR&PP) methodology

The PRPPWG was created to establish a framework for assessing generation IV nuclear systems against the proliferation resistance and physical protection goals of GIF. The PR&PP methodology developed by the group 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).

Other major outcomes from the group are available to the GIF community and more broadly through the GIF public website, including the Example Sodium Fast Reactor (ESFR) Case Study Report (GIF/PRPPWG/2009/002), the compendium report on PR&PP characteristics of each of the six GIF nuclear energy systems prepared with SSCs (GIF/PRPPWG/2011/002) and a set of frequently asked questions (FAQs) about the PR&PP methodology and applications (GIF/PRPPWG/2013/002), the compendium of materials presented at the PR&PP Methodology Workshop held in November 2015 (GIF/PRPPWG/2015/003).

In 2015, the document on FAQs was adapted and formatted to create a tri-fold leaflet which was distributed in various international symposia, workshops and conferences, including ICONE 23 (Chiba, Japan, 17-21 May 2015) and the ANS Winter Meeting (8-12 November 2015, Washington, DC, United States), as well as during the two the Experts Group/Policy Group (EG/PG) meetings and the workshop organised by the group at Berkeley University in connection with its 26th meeting, held in November.

Recognising that enhancements of the PR&PP methodology could be undertaken only after having benefitted from feedback from its applications in concrete case studies, the group focused its activities on communication to enhance the visibility of its outcomes and to encourage the use of its approach and tools within and outside GIF. Collaboration with other GIF bodies – in particular the RSWG and with other international endeavours on advanced nuclear systems, such as the IAEA/INPRO project were pursued actively. The group was represented in the two EG/PG meetings held in 2015 in Chiba, Japan and in Saint Petersburg, Russia.

The bibliography of the group, issued for the first time in mid-2014, is available on the GIF public website. It is maintained, updated and reissued annually. It provides a comprehensive list of publications in scientific journals and papers presented at major international conferences, covering all aspects of the PR&PP methodology and its applications within and outside GIF (www.gen-4.org/gif/jcms/c_71068/prpp-bibliography).

During the panel on GIF methodology working groups held during ICONE 23, the representative of the group provided an overview on its current activities and objectives for the coming years. A paper authored by the group was presented at the Global 2015 Conference held in Paris, France, on 21-24 September 2015. The paper summarises the status of the PR&PP methodology, illustrates its applications in various case studies and highlights challenges facing the group to strengthen its visibility and promote further uses of the approach by different stakeholders. One of the co-chairs of the group was invited to participate in the 50th Anniversary Workshop of the Molten Salt Reactor held at Oak Ridge National Laboratory, on 14-15 October 2015 and a Panel Session on Proliferation Risk and Sustainability of the ANS Winter Meeting held on 8-12 November 2015 in Washington DC, United States.

The 26th meeting of the group was held at UC Berkeley, California, United States, on 5-6 November 2015. It was associated with a meeting of the RSWG and a joint session of the two groups was organised to exchange information and discuss opportunities for further collaboration. The main outcome from the discussions is a decision to investigate further potential synergies and conflicts between safety, security and safeguards aiming at developing a white paper which could help in enhancing the methodologies developed by the two groups for the benefit of GIF SSCs and other reactor research or design teams. This joint activity will be initiated in 2016.

In connection with the 26th meeting, a workshop on 4 November 2015 was organised to introduce the PR&PP methodology in front of an audience of students and academics. The viewgraphs used during the workshop were compiled in a compendium which is posted on the GIF public website. The feedback from participants provided guidance on ways and means to enhance the accessibility and user friendliness of the approach, as well as to improve the flow of presentations, interactive sessions and discussions during the workshop. As mentioned above, the workshop agenda and presentation slides are available on the GIF open website (www.gen-4.org/gif/jcms/c_79016/prppwg-workshop-materials).

The lessons learnt from the workshops held yearly by the group constitute a robust set of guidance for future activities in the field of education and training. During the 26th meeting contacts were initiated with the newly created GIF Task Force on Education and Training and it is planned to strengthen this co-operation aiming at enhancing the materials available for workshops on the PR&PP methodology and promoting its dissemination through various media.

Representatives of the PRPPWG in GIF Experts and Policy Group meetings held in 2015 reported on the main activities being carried out and drew the attention of the GIF governance on the need for strengthening the awareness of SSCs on the PR&PP methodology. They stressed the relevance of using the approach proposed by the group for self-assessment by researchers and designers of the PR and PP characteristics and performance of their systems at an early stage of their development.

During the meetings of the SFR SSC and System Integration and Assessment (SIA) project held at JRC Ispra, Italy, in April 2015, one of the Euratom members of the PRPPWG was invited to make a presentation on the PR&PP methodology. He highlighted the main features of the approach and tools developed by the group and their usefulness in the context of self-assessments which might be undertaken under the auspices of the SIA project. Furthermore, he reminded the representatives of the SSC and SIA project that members of PRPPWG could provide assistance upon request to teams willing to undertake a PR&PP evaluation study.

The evolution of the international safeguards context is a key element for the evaluation of the proliferation resistance of an innovative nuclear system. Accordingly, the group maintains close contacts and regular exchange of information with the IAEA Department of Safeguards, for example through participation of members of the group in IAEA meetings, consultancies and conferences. In 2015, developments regarding the IAEA State Level concept to safeguards were followed with attention by the group in view of their potential impacts on the need for adaptation of the PR&PP methodology.

In the field of co-operation with other international endeavours, the group maintained regular exchange of information with the IAEA's INPRO project. It was represented at the interface meeting between INPRO and GIF held in March 2015 at the IAEA Headquarters in Vienna, Austria, where fruitful discussions were conducted on opportunities for future collaboration. A representative of INPRO participated in the 26th meeting of the group where he provided an overview on ongoing activities within the overall project, focusing on the most relevant outcomes from the INPRO Proliferation Resistance and Safeguardability Assessment (PROSA) tools project.

In sum, the PRPPWG has been actively engaged in outreach activities within and outside of GIF and seeks to increase it interactions with the GIF systems designers.

4.3 Risk and safety assessment methodology

The primary objective of the Risk and Safety Working Group is to provide an effective and harmonised approach to the safety assessment of generation IV systems in collaboration with and in support of all six SSCs. The RSWG proposes safety principles, objectives, and attributes based on Gen IV safety goals to guide research and development (R&D) plans. The RSWG also provides consultative support to SSCs and other Gen IV entities and undertakes appropriate interactions with regulators, IAEA, and other stakeholders. The RSWG has developed a safety assessment methodology consolidated in three main documents: the Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems (BSA), the Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems, and the Guidance Document for ISAM (GDI).

In May 2015, the third GIF Symposium was held in Makuhari Messe, Japan. At the symposium, a summary of the RSWG activities were presented in a talk on the guidance document for ISAM application expected to aid the SSCs to improve the safety of their respective systems, and on practical example of ISAM implementation in the reactor design process. The feedback received from the symposium participants was extremely valuable dealing in particular with the designers' difficulties in the full analysis of severe accidents and in the exhaustive identification of initiating events during the safety assessment.

The RSWG efforts in 2015 focused on the finalisation of the risk and safety white papers for the generation IV systems with the submission of the SFR document to the GIF Experts Group for approval and the final review of the VHTR document. In addition, the SCWR document is being reviewed by the RSWG members and discussions have been initiated with the new gas-cooled fast reactor (GFR) SSC to finalise their contribution. The risk and safety white papers are a joint work of the RSWG and each System Steering Committee (SSC) to present high-level information about the safety assessment of their system from the perspective of the applicability and helpfulness of the ISAM methodology.

Also in 2015, an important focus for the activity of the RSWG was related to the co-ordination of the safety design reviews of the six GIF reactor concepts after a decade since the start of the GIF in early 2000 to provide a snapshot of the main safety advantages and to identify the major safety challenges and the R&D needs to resolve those challenges. In June 2015, a joint workshop between RSWG members and the six SSC chairmen and representatives was held in Petten (the Netherlands) to prepare the safety assessment document according to the schedule and to discuss SSCs proposals. Three safety assessment documents from SFR, lead-cooled fast reactor [LFR] and VHTR concepts have been submitted by the SSCs and reviewed by the RSWG before their final approval by the GIF Expert Group. The RSWG is working in close contact with the other SSCs for the completion of their contribution.

In application of the lessons learnt from the accidents at TEPCO's Fukushima Daiichi nuclear power plant, the RSWG has started an internal review on the use of ISAM methodology to evaluate how those lessons can best shape our approach to assessing and ensuring the safety of generation IV systems. The objective is to analyse the ISAM methodology in reference to the Fukushima accident in order to identify any modifications needed in the methods and their application. The benefits of the application of such methods are to anticipate the challenges for Gen IV systems during the extreme external hazards and common cause failures.

Also in 2015, the RSWG worked in close collaboration with the GIF Safety Design Criteria Task Force (SDC-TF) contributing to the 2nd phase activity for development of the safety design guidelines (SDG). The RSWG supports the SDC-TF in the interaction between the GIF community and the international organisations and national regulators. In particular, the group continues to advice on the comments received by external organisations on the SFR SDC Phase 1 report and provide recommendations on the safety approach and safety assessment for the Gen IV reactor system.

In line with its advisory role to the PG and EG on interactions with the nuclear safety regulatory community, international organisations and relevant stakeholders, the RSWG maintains its own interfaces with the IAEA, INPRO, and MDEP. The RSWG also maintains internal contacts with the other methodology working groups and in particular with the PRPPWG. In November 2015 a joint meeting with the PRPPWG was held at the University of California Berkeley, United States, with the objective to exchange information and explore potential collaboration between the two groups in support to GIF SCCs. The joint meeting focused on the interface between safety, security and safeguards based on the proposal by the RSWG of a safety and security interface assessment methodology centred on the Objective Provision Tree (OPT) approach. Given the importance of addressing those issues a subgroup of members from the two working groups was created to evaluate the proposed draft methodology and develop a white paper on the safety, security and safeguards interface.

In 2015, the RSWG has newly experienced important changes in leadership with the departure of one of its co-chairs. This has presented some challenges in terms of continuity with the decade-long work performed within the group but also opportunities for new directions and thinking on how to accomplish the main objective of the group which remains the promotion of a consistent approach to safety, risk and regulatory issues among generation IV systems.

References

Ammirabile, L. and R. Nakai (2015), "Risk & safety working group", 3rd GIF Symposium, 19 May 2015, Makuhari Messe, Japan.

Chapter 5. Task force reports

5.1 Task Force on Safety Design Criteria

In 2015, the SFR Task Force (TF) pursued the integration of the international reviews on the safety design criteria (SDC) Phase I report and also proceeded to develop the safety design guidelines (SDG).

Following approval of the Generation IV International Forum (GIF) Policy Group in May 2013, the SDC Phase I report was circulated to international organisations (i.e. IAEA, MDEP, NEA Committee on Nuclear Regulatory Activities [CNRA]) and regulatory bodies of the states with active SFR development programmes under the GIF (i.e. China, EC, France, Japan, Korea, Russia, United States) for an external review and feedback. The formal reviews were conducted by the IAEA, US NRC, China's National Nuclear Safety Administration (NNSA), and France's IRSN. The review results include not only general comments (e.g. safety approach as the generation IV [Gen IV] reactor systems, relationship between safety and security, availability of a best-estimate method with considering sufficient uncertainty, independence of levels of defence-in-depth, and single/multiple failures and plant conditions) but also detailed specific recommendations for individual criteria related to technical characteristics of SFRs (e.g. a sodium-fire and its consequences on system steering committees [SSCs] important to safety, sodium-water reaction, and parameters important to accident analyses). The TF held three meetings in May, June and October 2015, and continued in-depth analyses of the external reviews. The TF is currently summarising the responses to the feedback and recommendations by these international reviews and preparing an update to the SDC Phase I report. As an additional interaction with international organisations, the NEA held a joint CNRA/Committee on the Safety of Nuclear Installations (CSNI) Ad hoc Group on the Safety of Advanced Reactors (GSAR) meeting in September 2015, and the GIF was invited to provide the status on GIF SDC Phase I report and SDG development as contributiions to the international efforts on regulation of the advanced reactors. It is anticipated that in the coming years (2016-2017) further feedback from the GSAR will be provided to improve both the SFR SDC and SDG based on regulatory insights.

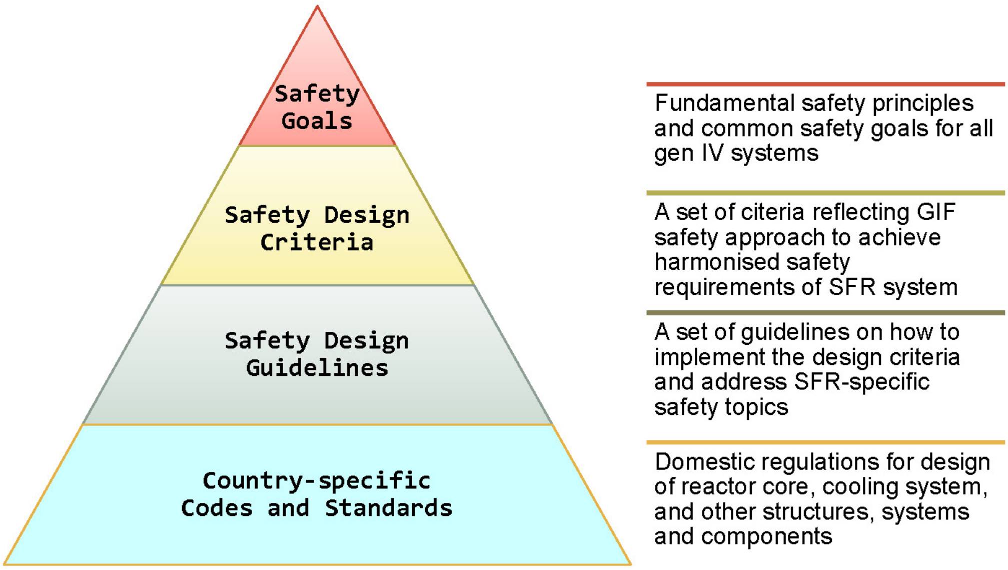
The international feedback on the SDC Phase I report for the Gen IV SFR reactor systems provided an incentive and motivation for further technical interpretation and clarification of the SDC. Based on these incentives/motivations, the Phase II activity of the SDC-TF for the development of SDG, which was started in 2013, has continued throughout 2015. The SDG is conceived as a series of detailed guideline documents at the level lower than the SDC in a hierarchy of the safety standards as shown in the Figure 5.1. The first two expected output of the SDG development effort are the reports on "Guidelines on Safety Approach and Design Conditions of Generation IV SFR systems" (so-called "Safety Approach SDG"), and on "Safety Design Guidelines on the Key Structures, Systems and Components" (so-called "SSC-SDG").

The first report on "Safety Approach SDG" aims to provide guidance on safety approaches according to the SDC. It covers specific safety issues on "prevention and mitigation of severe accidents (issues related to fast reactor core reactivity)" and "accident conditions to be practically eliminated (issues related to loss of heat removal)". Based on the progress for facilitating the common understandings on specific safety measures through the discussions not only by the TF but also with the stakeholders related to SFR development in a workshop and an international symposyum noted below, the final draft of the Safety Approach SDG report was developed by the TF and provided to the the Policy Group (PG) for the approval. The second report on "SSC-SDG" focuses on the functional requirements for SSCs important to safety and establish design parameters and constraints for postulated accidents. The TF focuses on the important elements for Gen IV SFR's reactor core system, reactor coolant system, and containment system, and the work on selected technical focal points will be summarised in a draft SSC-SDG report in 2016.

In order to discuss SDC/SDG with the stakeholders, the fifth joint GIF-IAEA Workshop on "Safety of Sodium-Cooled Fast Reactors"[1] was held on 23-24 June 2015 at the IAEA Headquarters. The main purpose of the workshop was to present, discuss, and review: i) "Safety Approach SDG"; ii) the information on implementation of SDC/SDG by the designers of innovative SFR concepts; and iii) the review comments on the SDC Phase I report from external organisations and their resolutions. The development status of "SSC-SDG" and its general contents were presented as well. At the 3rd GIF Symposium[2] at Makuhari in May 2015, the status on the SDC external review and the SDG development was presented, and the general discussions on the SDC/SDG development efforts for the Gen IV SFR system designs were conducted by the GIF participants.

The next SDC-TF meeting is planned in February 2016 and several additional meetings are foreseen to complete the SDC-TF Phase II activity in 2016.

Figure 5.1: **Hierarchy of GIF safety standards**



5.2 Interim Task Force on Sustainability

Background

As noted in the 2014 GIF annual report, the Policy Group authorised creation of an interim task force on sustainability in May 2014. Initial activities focused on organisation and scope, with the main work deferred until 2015. Maintaining GIF's narrow definition of sustainability was the key decision in guiding the TF's future work.

Keeping in mind its focus on i) resource utilisation and ii) waste management and minimisation, the interim TF met with chairs of GIF's three methodology working groups in the

1. Available at www.iaea.org/NuclearPower/Meetings/2015/2015-06-23-06-24-NPTDS.html.
2. Available at www.gen-4.org/gif/jcms/c_74878/generation-iv-international-forum-gif-symposium.

wings of the GIF-INPRO Interface Meeting in Vienna on 5 May 2015. To some extent, INPRO had stimulated interest in a sustainability assessment by requesting GIF participation in its project on sustainable nuclear energy development. One question for the Sustainability Task Force was determining what gaps, if any, GIF would need to fill in the INPRO methodology.

The principal outcome of the May 2015 meeting was to schedule a sustainability meeting at OECD in September 2015 with ad hoc representation to discuss two topics:

- Review pertinent literature on sustainability produced during the early GIF screening process circa 2000-2002 and subsequently by member countries, IAEA, and NEA. Assess whether any notable gaps exist.
- Request each member country, on a voluntary basis, to discuss national views on sustainability. There was no intent to attempt to harmonise these views, which utilise different definitions, time horizons and assumptions.

The ad hoc members of the interim task force met at the OECD on 16-17 September 2015. China, France, Japan, Korea, Russia, the United States, the NEA and IAEA/INPRO were represented. The participants presented a broad range of relevant technical subjects during the 1½-day meeting.

Review of previous and ongoing sustainability assessments

The interim TF discussed sustainability evaluations presented in three categories: i) screening evaluations during the formation of GIF; ii) national and international fuel cycle assessments from 1980 on; and iii) ongoing INPRO project on sustainability.

Two participants had been key players in the screening assessments and methodology development that preceded the down selection to the six GIF systems and production of the initial Research and Development (R&D) Roadmap that was published in 2002. From the discussion, it was clear that a great deal of effort by experts had gone into the evaluations. Further the outcomes were technically solid, well supported by the evaluations. However, the effort to produce more precise evaluation methodology proved elusive due to the imprecision of definitions, goals, assumptions, economic data, etc. No doubt this experience contributed to the lack of formation of a sustainability methodology working group, in contrast to the other three top-level goals of the forum.

The interim TF noted the sizeable list of extensive fuel cycle option evaluation reports that looked at resource utilisation and waste management dating from the International Nuclear Fuel Cycle Evaluation in 1980 to the US Fuel Cycle Options Study from 2011-2014. Two aspects of this survey stand out. The first is that the number of fuel cycle options analysed and categorised has filled most fuel cycle space that has been previously imagined. Second, considering the narrow GIF criteria of efficient resource utilisation and waste reduction, conclusions regarding the optimum fuel cycle have not changed in the past five decades – physics dictates that a fast spectrum reactor with continuous recycle performs best. What has changed is a broad recognition that future energy systems will comprise a mix of reactor types providing different services, with fast reactors significantly contributing to sustainability of the overall system.

The recently completed US Department of Energy fuel cycle options study was the most relevant report for the interim TF. The study evaluated hundreds of fuel cycles with uranium, plutonium, minor actinide and thorium options in fast and thermal systems. The evaluation used five waste management criteria, proliferation risk, nuclear material security risk, safety, four environmental impact criteria, and uranium resources. The best performing evaluation groups are shown in Table 5.1, with only GIF sustainability-relevant criteria listed, with the once-through thermal reactor cycle shown for comparison.

Table 5.1: **Summary of metrics for the best performing fuel cycle evaluation groups**

Fuel cycle option	Once-through US system	U/Pu recycle fast systems	U/TRU recycle, fast systems	U/TRU recycle, fast and thermal systems
Nuclear waste management				
Mass of SNF+HLW, t/GWe-year	12-36	<1.65	<1.65	<1.65
Activity@100 years, MCi/GWe-year	1.05-1.60	0.67-1.05	0.67-1.05	0.67-1.05
Activity@100 000 years, MCi/GWe-year	0.001-0.0023	0.0005-0.001	0.0005-0.001	0.0005-0.001
Mass of DU+RU+RTh, t/GWe-year	120-200	<1	<1	<1
Volume of LLW, m^3/GWe-year	252-634	252-634	252-634	252-634
Resource utilisation				
Uranium resources	>145	<3.8	<3.8	<3.8

The relevance of the US Fuel Cycle Options Study to GIF is that in addition to being as complete as reasonably possible, the catalogue of results is publically available at https://connect.sandia.gov/sites/NuclearFuelCycleOptionCatalog/SitePages/a/homepage.aspx and a tool set for doing further evaluations is also publically available at https://inlportal.inl.gov/portal/server.pt/community/nuclear_science_and_technology/337/fuel_cycle_evaluation_and_screening_set_tool. The widespread availability of these results, together with recent publications by NEA and ongoing work by INPRO, obviated the need for further immediate work on sustainability methodology by the interim TF.

NEA presented its perspective on sustainability through four recent reports: *Nuclear Fuel Cycle Transition Scenarios Studies* (2009); *Potential Benefits and Impacts of Advanced Nuclear Fuel Cycles and Actinide Partitioning and Transmutation* (2011); *Trends in the Nuclear Fuel Cycle: Economic, Environmental and Social Aspects* (2001); and *Transition towards a Sustainable Nuclear Fuel Cycle* (2013). Each of these reports can be downloaded from the NEA website.

The interim TF reviewed the extensive sustainability activities ongoing in INPRO, in which GIF member countries and other interested nations participate. The INPRO evaluation methodology considers safety, security, affordability, and environment in addition to waste and resources. INPRO would welcome more participation from GIF members interested in extended sustainability evaluations.

Although the TF collected information on uranium resource projections and prospects for economical recovery of unconventional resources, consideration of the impact of resource assumptions on sustainability was tabled. Even large-scale variations in resource projections would have no near-term impact on technology choices.

National views on sustainability

After some initial discussion, the interim TF decided that an informal discussion of national fuel cycle policy, and thus sustainability, would be more productive in identifying key assumptions and trends. While some countries, notably France, Korea and Russia, had relatively well defined policies, the situation tended to be more complex in other countries. Within most countries there are ongoing discussions about the approach to sustainability, or at least what steps are considered important now. For example, there is little common ground on the level of concern about future uranium availability, the mix of reactor types supplying different services (electricity, industrial heat and waste transmutation), the rate of introduction of advanced technology, the definition of sustainability, and the interaction with future geologic waste repositories.

National policies tended to look at sustainability in a broader sense, in particular: safety, land and water resource use, public acceptance, completion from other technology options, surety of energy supply, waste repository siting and operation, long-term uranium prices, contribution to carbon-free energy production, integration with renewables on the grid, and short-term market conditions. If nuclear energy endures, part of the common vision is that fast reactors with continuous recycle will be part of the mix and adequate waste disposal facilities will be available. Not so clear is whether global co-operation will succeed in distributing regional centres to supply these specialised services for nations that choose to participate.

5.3 Education and training

Background

Jacques Bouchard, former GIF Chairman launched in 2009 the first Education and Training Task Force (ETTF), but its level of activity remained low. The ETTF was re-established in 2015. Per request of John Kelly, the GIF Chairman, Patricia Paviet, during the Experts Group (EG) meeting in Chiba, Japan in May 2015, presented the National Analytical Management Program (NAMP) initiative. The NAMP education and training subcommittee developed several series of one to two-hour webinar presented by experts on different topics relevant to radiochemistry. The US Departments of Energy and Homeland Security offer these free webinars as live, interactive conferences. Recorded and archived versions comprise a library vital to future generations of radiochemists and scientists interested in radiochemistry.

Developing webinars from initial concept to full realisation which could provide a basic understanding of the different Gen IV systems was the concept presented during the meeting and it was fully accepted by the EG and PG members.

Following the meeting in Japan, members were nominated and the task force was established. A strong, multinational working team of volunteers has been formed:

- Takatsugu Mihara, Japan.
- Pavel Alekseev, Russia.
- Nolitha Mpoza, South Africa.
- Konstantin Mikityuk, Switzerland.
- Concetta Fazio, European Union.
- Yougmi Nam, Korea.
- Il Soon Hwang, Korea.
- Claude Renault, France.
- Patricia Paviet, United States.
- Massimiliano Fratoni, United States.

Terms of reference

At the GIF meeting held in October 2015 in Saint Petersburg, Russia, the scope for the GIF-ETTF was proposed, discussed and endorsed with an action to develop the terms of reference (TOR).

Objectives

The GIF-ETTF will serve as a platform to enhance open education and training (E&T) as well as communication and networking of people and organisations in support of GIF.

The task force will work to reach the following objectives:

- Identify the stakeholder groups and assess their needs for generation IV E&T.
- Create and maintain a social medium platform to exchange information and ideas on general generation IV research and development (R&D) topics as well as related GIF E&T activities.

- Structure and disseminate open E&T materials using social media, the GIF website and other platforms.
- Reach out to the stakeholder groups by developing a comprehensive downloadable brochure describing the GIF-ETTF initiatives.
- Develop and launch prototype webinar series on one or more generation IV systems and/or on the cross-cutting methodologies.
- Propose, organise and/or support open generation IV E&T seminars, e.g. at the triennial GIF Symposium.
- Evaluate the need for developing massive open online courses on generation IV systems and assess necessary resources (if applicable). Make a recommendation to the GIF Experts Group.
- Evaluate the need for developing a regular summer/winter school on generation IV systems and assess necessary resources (if applicable). Make a recommendation to the GIF Experts Group.
- At the end of the task force period assess the need of creating a standing Working Group of Education and Training within GIF. Make a recommendation to the GIF Policy Group (final objective).

Organisation

The GIF-ETTF consists of members nominated by the GIF Policy Group members and by GIF participants on a volunteer basis.

The GIF-ETTF will work during the period of 2016-2018.

Each member will participate to the GIF-ETTF with its own financial and human resources.

The GIF-ETTF Chair and Co-chair will be elected and:

- will guide the overall activities to reach the set objectives;
- will interact with the GIF Technical Secretariat to make use of its services and capabilities, including the management of the GIF public website. Labelling of GIF-ETTF events (schools, workshops, symposia, seminars) may use the GIF Logo upon review by the TD and TS;
- will organise a regular (monthly) conference call;
- will organise once a year, a co-ordination meeting to summarise progress of the ongoing activities and develop the activity for the following year;
- will report to the GIF Experts Group on the progress of the activities;
- will report back to the GIF-ETTF members.

Development of GIF Education and Training LinkedIn Group

The GIF Education and Training LinkedIn Group was created on 6 October 2015, www.linkedin.com/grp/home?gid=8416234.

This group has been created to promote the GIF-related educational activities. This is a way to disseminate the information about the group in order to invite i) PhD students working on one or several Gen IV systems or on relevant cross-cutting topic; and ii) professors, lecturers involved in teaching the courses which are relevant to Gen IV systems; iii) and people who want to learn more about Gen IV reactors.

The group settings enable the promotions features (to promote ongoing PhDs, summer schools, training), 2-enable the job features (to advertise the open PhD positions). At this point of time, members are free to post comments only and submit everything else. Anyone interested in Gen IV systems is encouraged to join the group.

Chapter 6. Senior Industry Advisory Panel (SIAP)

The Senior Industry Advisory Panel (SIAP) provides advice to the Generation IV International Forum (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. Over the last years, the SIAP has met at least once per year to consider systems and/or cross-cutting issues identified by the Policy Group (PG), to provide its recommendations relative to long-term strategic issues, including regulatory, commercial or technical considerations.

At its meeting of May 2015, the PG asked the SIAP to consider and advisce on the harmonisation of codes and standards (C&S). Based on the return of experience of C&S harmonisation for Gen III light water reactors (LWRs) and, in particular, the international efforts by organisations such as WNA CORDEL (for industry) and MDEP (for safety authorities), SIAP was requested to recommend methods to prevent divergences of current C&S, to identify candidate systems and areas for stronger harmonisation while keeping a realistic approach, and, as a result, to propose a process for the effective initiation of harmonisation for Gen IV systems.

The PG also asked SIAP to list attributes of Gen IV systems that are most attractive from a vendor or utility perspective, such as supply chain maturity, inherent safety, fuel cycle sustainability, overall economics, power level, modularity of construction,… SIAP was requested to provide views on the need for additional research and development (R&D) in these fields. The request was motivated by the need for a higher flexibility of the nuclear systems of the future to integrate energy networks with increasing proportion of intermittent Renewable Energy Sources (RES).

Finally the PG wanted to know more on the industry views on market conditions and expected timelines for the commercialisation of Gen IV reactors, and on R&D priorities to expedite commercialisation. Beyond the economic conditions to foster investment, the issue was linked in particular to the Gen IV sustainability aspects associated with security of supply of uranium and the long-term management of high-level waste.

SIAP met on the sides of the PG meeting in Saint Petersburg in October 2015 under the chairmanship of the recently elected Chair, Dr Haeryong Hwang. The meeting was attended by seven members (or alternates), two Experts Group (EG) representatives ensuring the continuity with former years, and the support provided by the NEA.

On the topic of harmonisation of C&S, SIAP discussed the situation for LWRs and noted that C&S have been developed by integrating the feedback of construction and operational experience. This cannot be the case yet for Gen IV systems at early stages of industrial design. Hoping to have full harmonisation of C&S for a Gen IV system seems unrealistic due to the associated commercial aspects. Therefore it would be more realistic to focus the harmonisation on a limited number of issues where there is common interest.

In practical terms, the SIAP proposed to ask the system steering committees (SSCs) to define lists of issues for which harmonisation would be most useful. Issues such as qualification of materials, NDE methods and processes, modelling and simulation methods and tools, all related to time consuming and expensive data acquisition efforts and data bases management, were flagged. The EG could then consolidate the SSCs inputs and provide its “GIF prioritised list of topics for harmonisation” to the standards developing organisations (SDOs) with a request for them to co-operate in developing common approaches, up to C&S if possible. Where appropriate, the NEA might be proposed to support associated international benchmarks and data bases management, since experience is already there.

SIAP considered that, if indeed SDOs follow-up the GIF request, it would give a strong signal to industry to engage into the research and development on innovative materials, NDE techniques, modelling and simulation tools, etc.

On the topic of Gen IV systems attributes attractive for industry, SIAP considered that the list of attributes proposed by the PG was not all consistent. For industry, to bring Gen IV to the market, there are two main priority attributes: economics and public acceptance, both being linked. Public acceptance is associated with the perception of sustainability related to safety and waste management. Economics embraces the aspects of the cost and risks of the capital investment, but also aspects of affordability for the consumers. Both are linked but a number of influencing factors are not under control when it comes to energy price making. As a starting point, a full costs approach for the diverse sources of energy is necessary to bring some clarity.

From there the discussion led to a “new” important attribute for Gen IV systems: the capacity to integrate into the energy systems and markets of the future, with more RES and so the need for higher flexibility. SIAP considers that a step change is necessary and the development of Gen IV systems might provide an opportunity for this step change.

SIAP recommended that, in addition of keeping focus on the highest safety levels, on effective and improved long-term waste management, and on cost control, GIF puts more emphasis on the flexibility aspects to foster the integration of Gen IV systems in the energy systems of the future.

On the topic of market conditions and timelines for Gen IV systems commercialisation, SIAP considers that it is impossible to predict the future, since the energy policies and market conditions are too unstable, with too much parameters and uncertainties. Uranium supply is not a problem for the coming decades. On the longer term it will depend on the Gen III construction rate, which again is uncertain. On the sustainability dimension associated with long-term waste management and the minor actinides recycling of Gen IV systems, SIAP indeed considers it is mainly an issue of public acceptance and therefore it will be to governments to decide the way forward, recycling or not before going to geological disposal. Industry will adapt, as long as it stays economically viable. Therefore, industry needs to have the options to make the choices when the time will come.

For this, SIAP makes a strong plea, noting the long lead times for developments and innovation in the nuclear sector, to prepare without delay for the building of the necessary demonstrators. The goals of such demonstrators, close enough to full scale, are to prove the technical and industrial feasibility of Gen IV systems, their safety levels and licensability, their flexibility and integrability in energy systems (synergies with Gen III and RES in particular), their sustainability (in particular for recycling and waste management aspects). It would also provide elements on the economics, which would then be further consolidated via the next stage, a first-of-a-kind (FOAK) “commercial” plant, leading to further deployment.

SIAP had also a discussion on its further involvement in GIF. It proposed to the PG to meet twice a year. In addition to the standard meeting on the sides of the autumn PG to discuss the yearly “charge”, SIAP would meet also in spring, on its own, to discuss how industry can be more proactive in support of Gen IV systems developments. A first spring meeting would take place on the sides of the spring 2016 EG meeting, and SIAP would report on its deliberation to the EG, who may then integrate this outcome in its elaboration of the next “charge”.

In order to be able to play a more proactive role, the SIAP asked the PG to be involved, as appropriate, in policy-related issues and meetings, such as the elaboration of GIF policy statements for political leaders, activities of the economics WG, in addition to a standing invitation to attend the SSCs, EG and PG.

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 Generation IV International Forum (GIF) has been working on cross-cutting 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.

The 9th GIF-INPRO interface meeting was held at the IAEA Headquarters, 4-5 March 2015. It provided the opportunity for an informal information exchange on GIF and IAEA activities covering various research and development (R&D) areas. The GIF presented the status of the development efforts of all six GIF systems (SFR, very-high-temperature reactor [VHTR], gas-cooled fast reactor [GFR], supercritical-water-cooled reactor [SCWR], lead-cooled fast reactor [LFR] and molten salt reactor [MSR]), while the IAEA presented the status of its reactor technology development activities for high-temperature gas-cooled, supercritical water, and small modular reactors. In the safety area, the GIF presented the recent activities of the Risk and Safety Working Group and of the Safety Design Criteria Task Force. The IAEA presented safety standards revised in the wake of the Fukushima Daiichi accident. The respective activities and the path forwarded in the area of proliferation resistance were presented, with IAEA presenting the update of the Safeguards by Design document. Status reports of the methodology activities were given, more specifically of the INPRO methodology and the GIF Economic Modeling Working Group.

Several topics amenable to collaboration were identified, e.g. promoting the collaboration between IAEA HTGR and GIG VHTR safety-related activities, and various methodology areas. Last but not least, the GIF-IAEA Coordination Matrix and Action Items was updated.

GIF/IAEA Safety Workshop

In 2015, the GIF has continued to co-operate with the IAEA for organisation of workshops on sodium-cooled fast reactor (SFR) safety. As a follow-up on previous four such workshops in earlier years, the 5th GIF-IAEA SFR Safety Workshop was held at the IAEA Headquarters in Vienna on 23-24 June 2015. Objectives of the workshop included a discussion of the comments on the GIF SFR Safety Design Criteria (SDC) report from the regulators, review of the SFR Safety Design Guidelines (SDG) currently being prepared by the GIF Task Force, and information sharing on implementation of the criteria and guidelines by the designers of innovative SFR concepts.

The participants included representatives from NSC (Canada), CIAE (China), JRC (EU), CEA, AREVA and EDF (France), GRS (Germany), JAEA (Japan), KAERI and KEPCO (Korea), IPPE and OKBM (Russia), Argonne National Laboratory (ANL), GEH, INL, and NRC (United States), and the NEA. Specific discussion items included an update by the GIF Task Force on revisions to SFR SDC based on the comments received from the regulators, and on preparation of the first SDG report focusing on core reactivity and loss of decay heat removal issues. In a separate sessions, implementation of SDC by SFR designers were covered in presentations from AREVA (ASTRID),

GRS, JAEA (JSFR), KAERI (PGSFR), OKBM (BN-1200), JRC (ESFR), CIAE (CDFR), IGCAR (PFBR), JAEA (JSFR), GE-Hitachi (PRISM). The next workshop is planned on 21-22 June 2016 in Vienna.

7.2 Interaction with regulators (GSAR)

In 2014, the NEA helped GIF begin a dialogue with the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations on the safety of advanced reactors. Subsequently these two NEA committees created the Ad hoc Group on the Safety of Advanced Reactors (GSAR), which among other things will help identify needed safety research in anticipation of licensing. The mandate of GSAR was approved by the CNRA and CSNI in June 2015. The GSAR experts met with GIF observers in September 2015 and decided to focus their efforts on the SFR system.

Appendix 1. GIF technology goals and systems

A.1 Technology goals of the Generation IV International Forum (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 off-site 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)
CD&BOP	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
ETTF	Education and Training Task Force
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)
HTR	High-temperature gas-cooled reactor
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
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
TD	Technical director
TF	Task force
TH&S	Thermal-hydraulics and Safety (SCWR signed project)
TS	Technical secretariat

VHTR	Very-high-temperature reactor
WG	Working group

Technical terms

AGR	Advanced gas-cooled reactor (United States)
ALFRED	Advanced lead fast reactor European demonstrator
ASTRID	Advanced sodium technological reactor for industrial demonstration
ATR	Advanced test reactor (at INL)
AVR	*Arbeitsgemeinschaft Versuchsreaktor*
BWR	Boiling water reactor
CEFR	China experimental fast reactor
CFD	Computational fluid dynamics
COL	Combined construction and operating licence
CRP	Co-ordinated research project
DHR	Decay heat removal
DNB	Departure from nucleate boiling
ELFR	European lead fast reactor
ESFR	Example sodium fast reactor
EVOL	Evaluation and viability of liquid fuel fast reactor system (Euratom FP7 Project)
FSA	Fuel sub assembly
FHR	Fluoride salt-cooled high-temperature reactor
FOAK	First-of-a-kind
GHG	Greenhouse gas
GTHTR300C	Gas turbine high-temperature reactor 300 for cogeneration
GT-MHR	Gas turbine-modular helium reactor
GV	Guard vessel
HANARO	High-flux advanced neutron application reactor
HF	Hydrogen fluoride
HLM	Heavy liquid metal
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
IHX	Intermediate heat exchanger
IRRS	Integrated Regulatory Review Service
JSFR	Japanese sodium-cooled fast reactor
LOCA	Loss-of-coolant accident
LWR	Light water reactor
MA	Minor actinides
MELCOR	Methods for estimation of leakages and consequences of release (NRC code developed by Sandia National Laboratories)
MOX	Mixed oxide fuel
MSFR	Molten salt fast reactor
NGNP	New generation nuclear plant
NHDD	Nuclear hydrogen development and demonstration
NPP	Nuclear power plant
ODS	Oxide dispersion-strengthened
PBMR	Pebble bed modular reactor
PDC	Plant dynamics code
PHX	PRACS (Pool Reactor Auxiliary Cooling System) heat exchanger
PIE	Post-irradiation examinations
PWR	Pressurised water reactor
PYCASSO	Pyrocarbon irradiation for creep and shrinkage/swelling on objects

R&D	Research and development
RV	Reactor vessel
SCC	Stress corrosion cracking
SDG	Safety design guideline
SEM	Scanning electron microscopy
SCW	Supercritical water
SMART	System-integrated modular advanced reactor
SMR	Small modular reactor
SSTAR	Small, sealed, transportable, autonomous reactor
STELLA	Sodium integral effect test loop for safety simulation and assessment
TEM	Transmission electron microscopy
THTR	Thorium high-temperature reactor
TRISO	Tristructural isotopic (nuclear fuel)
TRU	Transuranic
XRD	X-ray diffraction
ZrC	Zirconium carbide

Organisations, programmes and projects

ANL	Argonne National Laboratory
ANRE	Agency for Natural Resources and Energy (Japan)
ANS	American Nuclear Society
ARC	DOE Office of Advanced Reactor Concepts (United States)
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire (French nuclear safety authority)
CAEA	China Atomic Energy Authority (China)
CEA	Commissariat à l'énergie atomique et aux énergies alternatives (France)
CNL	Canadian Nuclear Laboratories
CNRS	Centre national de la recherche scientifique (France)
CNSC	Canadian Nuclear Safety Commission
DEN	Direction de l'énergie nucléaire (Commissariat à l'énergie atomique, CEA)
DOE	Department of Energy (United States)
EC	European Commission
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
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
INET	Institute of Nuclear and New Energy Technology
INL	Idaho National Laboratory (United States)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
ITU	(IAEA)
LEADER	Institute for Transuranium Elements
	Lead-cooled European Advanced Demonstration Reactor
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre (Euratom)
KAERI	Korea Atomic Energy Research Institute
KIT	Karlsruhe Institute of Technology (Germany)
MDEP	Multinational Design Evaluation Programme
MOST	Ministry of Science and Technology (China)
MTA	Hungarian Academy of Sciences Centre for Energy Research
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission (United States)
NRCan	Department of Natural Resources (Canada)

NRG	Dutch Nuclear Safety Research Institute
NTPD	Nuclear Power Technology Development Section (IAEA)
NUBIKI	Hungarian Nuclear Safety Research Institute
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory (United States)
PBMR Pty	Pebble Bed Modular Reactor (Pty) Limited (South Africa)
PSI	Paul Scherrer Institute (Switzerland)
RIAR	Research Institute of Atomic Reactors
SUSEN	The Sustainable Energy Project (Czech Republic)
VTT	Valtion Teknillinen Tutkimuskeskus (Technical Research Centre of Finland)
VUJE	Slovakian engineering company

www.gen-4.org

This ninth edition of the Generation IV International Forum (GIF) Annual Report highlights the main achievements of the Forum in 2015. On 26 February 2015, the Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy Systems was extended for another ten years, thereby paving the way for continued collaboration among participating countries. GIF organised the 3rd Symposium in Makuhari Messe, Japan in May 2015 to present progress made in the development of the six generation IV systems: the gas-cooled fast reactor, the sodium-cooled fast reactor, the supercritical-water-cooled reactor, the very-high-temperature reactor, the lead-cooled fast reactor and the molten salt reactor. The report gives a detailed description of progress made in the 11 existing project arrangements. It also describes the development of safety design criteria and guidelines for the sodium-cooled fast reactor, in addition to the outcome of GIF engagement with regulators on safety approaches for generation IV systems.

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