ANNUAL REPORT 2016

GEN IV International Forum℠

Foreword from the Chair

It is my privilege as the new Chair of the Generation IV International Forum (GIF) to present our 2016 GIF Annual Report, an important publication that offers a comprehensive update on the achievements of collaboration under the GIF Framework.

Being elected by the GIF Policy Group (PG) in April 2016 to become the fifth GIF Chair was a great honour, especially following in the footsteps of John Kelly (United States) and his prestigious predecessors. Under John's leadership, GIF activities have made some important progress, including the launching in 2014 of the process to extend the GIF Framework Agreement and to update the GIF *Technology Roadmap Update for Generation IV Nuclear Energy Systems*, initiated under Yutaka Sagayama's chairmanship. This flagship GIF publication began the engagement with regulators on safety design criteria and guidelines.

Such actions contributed to strengthening the GIF strategic vision for moving towards the demonstration phase of Gen IV systems, an endeavour that I will continue to promote as Chair.

To support our activities, I am fortunate to have the support of three Vice-Chairs that were tasked in 2016 with specific missions:

- Hideki Kamide (Japan) is leading an important project on market issues that aims to review how the market environment has evolved since the launch of GIF at the turn of the millennium and what it means in terms of challenges and opportunities for the development and future deployment of Gen IV systems.
- John Kelly is leading our activities on regulatory issues, including engagement with safety authorities in venues such as the Ad hoc Group on the Safety of Advanced Reactors (GSAR) co-ordinated by the Nuclear Energy Agency (NEA), as well as the development of safety design criteria and guidelines.
- Hark Rho Kim (Korea) is leading GIF outreach activities related to the identification of research infrastructure as well as international collaborations in order to promote the role, vision and results of GIF activities outside of our community.

In addition to the new leadership role of our three Vice-Chairs, activities benefit from the efficient support of the GIF Technical Secretariat provided by the NEA and co-ordinated by Henri Paillère. I am also pleased to have Alexander Stanculescu (United States/Idaho National Laboratory) serving as Technical Director, providing a high level of expertise to the PG and Chairing the Experts Group (EG), as well as François Storrer (France/CEA), as Policy Director and Chief of Staff, in charge of legal and communications issues, and the management of PG activities.

In 2016, a significant event for the GIF was to welcome Australia as its 14^{th} member following unanimous approval by the GIF PG. The GIF Charter was signed in June 2016 by Adrian Paterson of the Australian Nuclear Science and Technology Organisation (ANSTO) and the formal process for the signature of the Framework Agreement was initiated. We now look forward to Australia becoming fully engaged in the GIF research activities, particularly in the area of materials for very-high-temperature and molten salt reactors for which ANSTO has expressed an interest.

By the end of 2016, the Extension of the Framework Agreement had been signed by all of the GIF members: this offers a new momentum to foster the development of GIF activities over the next decade. GIF members have also renewed their commitment to the collaborative research and development (R&D) activities that are taking place at the system level by extending the System Arrangements (SAs) for the sodium-cooled fast reactor (SFR) (February 2016) and for the very-high-temperature reactor (VHTR), the supercritical water-cooled reactor (SCWR) and the gas-cooled fast reactor (GFR), all extended in November 2016. The molten salt reactor (MSR) and lead-cooled fast reactor (LFR) systems are not impacted by this process as they still operate under a Memorandum of Understanding.

In parallel, the PG tasked the Policy Director to oversee a revision of GIF procedures in terms of engagement with private companies, including start-ups which focus on Gen IV systems. This reflects the increasing interest of the private sector in GIF activities that could offer some new collaboration opportunities.

GIF cross-cutting activities are also continuing through dedicated task forces. The Education and Training Task Force was active in launching a series of monthly GIF webinars. The first presentation was delivered in September 2016 by John Kelly on "Atoms for Peace – the Next Generation" and gathered more than 350 participants. In addition, the task force contributed to international summer schools and is expending its network through the use of social media.

The task force responsible for developing safety design criteria (SDC) for SFR delivered some important results. The draft SFR SDC report has been extensively presented and reviewed by external stakeholders, including national safety authorities and the International Atomic Energy Agency. The task force has now started a Phase II of its activities with the development of safety design guidelines (SDG) that aim to assist SFR developers and vendors to use SDC in their design process. In parallel, it was also decided to extend the SDC/SG to other GIF systems, starting with the LFR and the VHTR.

The GIF Senior Industry Advisory Panel (SIAP), which provides strategic advice to the PG, increased its engagement with the forum in 2016. A number of new members joined and the SIAP Charter was updated to encourage a more diverse membership base. With a new three-year plan, the SIAP has established a dedicated work programme that includes support to the Vice-Chair mission on market issues as well as additional interactions with the GIF methodology working groups. SIAP members also aim to further engage with the GIF systems to support their plans to progress towards the demonstration phase of the different systems. As GIF Chair, I am indeed convinced that we need to maintain a strong relationship and dialogue with the industry in order to be successful in the implementation of the GIF Roadmap, moving towards the licensing of Gen IV systems.

Finally, important progress continued to take place within the collaborative R&D of our six systems and through the activities of our three methodological work groups: namely, the Economic Modelling Working Group (EMWG), the Proliferation Resistance and Physical Protection Working Group (PRPPWG), and the Risk and Safety Working Group (RSWG). In 2017, the System Steering Committees and methodology working groups will work together under the guidance of our Technical Director to produce an update of the GIF 2009 R&D outlook. This publication will be released in 2018, ahead of the Fourth GIF Symposium.

Dr François Gauché
GIF Policy Group Chairman

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

1.1. GIF membership

The Generation IV International Forum (GIF) has 14 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. After approval of its bid to join the GIF, Australia signed the charter in June 2016 becoming the 14th GIF member. Signatories of the charter are expected to maintain an appropriate level of active participation in GIF collaborative projects.

Table 1.1: **Parties to GIF Framework Agreement, System Arrangements and Memoranda of Understanding as of 31 March 2017**

Member	Implementing agents	Framework Agreement	System Arrangements (SA) (Extension)				Memoranda of Understanding	
		Date of signature or receipt of the instrument of accession (Extension)	GFR	SCWR	SFR	VHTR	LFR	MSR
Argentina (AR)								
Australia (AU)								
Brazil (BR)								
Canada (CA)	Department of Natural Resources (NRCan)	02/2005 (10/2016)		11/2006 (12/2016)				
Euratom (EU)	European Commission's Joint Research Centre (JRC)	02/2006 (11/2016)	11/2006 (03/2017)	11/2006 (03/2017)	11/2006 (03/2017)	11/2006 (03/2017)	11/2010	10/2010
France (FR)	Commissariat à l'énergie atomique et aux énergies alternatives (CEA)	02/2005 (02/2015)	11/2006 (11/2016)		02/2006 (02/2016)	11/2006 (12/2016)		10/2010
Japan (JP)	Agency for Natural Resources and Energy (ANRE) Japan Atomic Energy Agency (JAEA)	02/2005 (02/2015)	11/2006 (10/2016)	02/2007 (11/2016)	02/2006 (02/2016)	11/2006 (11/2016)	11/2010	
Korea (KR)	Ministry of Science, ICT and Future Planning (MSIP) and Korea Nuclear International Cooperation Foundation (KONICOF)	08/2005 (02/2015)			04/2006 (02/2016)	11/2006	11/2015	
People's Republic of China (CN)	China Atomic Energy Authority (CAEA) and Ministry of Science and Technology (MOST)	12/2007 (06/2016)		05/2014 (12/2016)	03/2009 (08/2016)	10/2008 (12/2016)		
Russia (RU)	State Atomic Energy Corporation "Rosatom" (Rosatom)	12/2009 (06/2015)		07/2011 (11/2016)	07/2010 (02/2016)		07/2011	11/2013
South Africa (ZA)	Department of Energy (DoE)	04/2008 (09/2015)						
Switzerland (CH)	Paul Scherrer Institute (PSI)	05/2005 (08/2015)				11/2006 (12/2016)		11/2015
United Kingdom (GB)								
United States (US)	Department of Energy (DOE)	02/2005 (02/2015)			02/2006 (02/2016)	11/2006 (11/2016)		01/2017

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

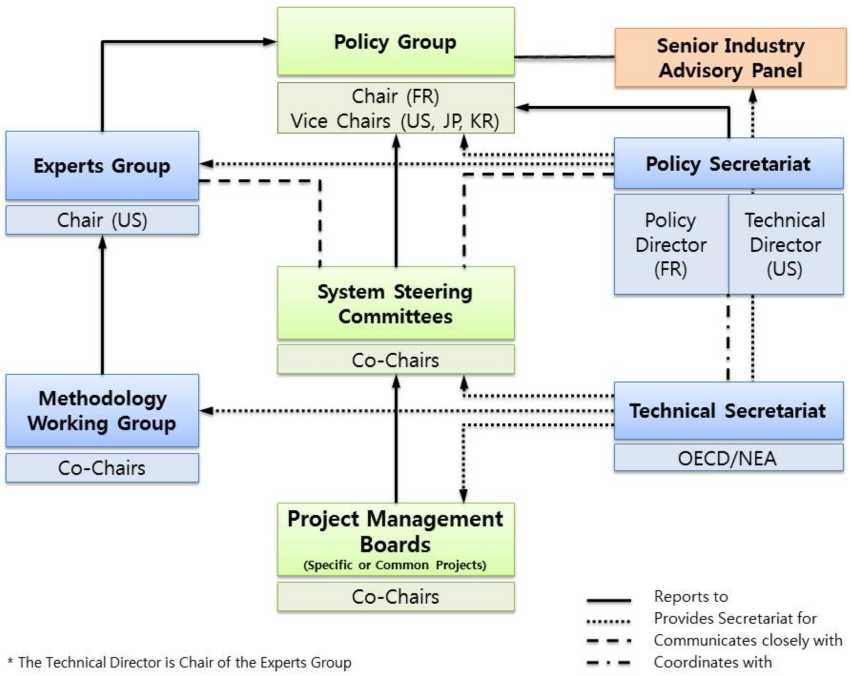
Among the signatories to the charter, ten members (Canada, France, Japan, China, Korea, Russia, South Africa, Switzerland, the United States and Euratom) have signed or acceded to the Framework Agreement (FA) and its extension as shown in Table 1.1. Parties to the FA formally agree to participate in the development of one or more Generation IV systems selected by GIF for further research and development (R&D). Each party to the FA designates one or more implementing agent to undertake the development of systems and the advancement of their underlying technologies. Argentina, Brazil and the United Kingdom[2] have signed the GIF Charter but did not accede to the FA; accordingly, within the GIF, they are designated as "non-active members". Australia, which signed the charter in June 2016, is preparing to deposit an instrument of accession to the Framework Agreement as extended and will likely become an active member in 2017.

Members interested in implementing co-operative R&D on one or more of the selected systems have signed corresponding System Arrangements (SA) consistent with the provisions of the FA. This is the case for the sodium-cooled fast reactor (SFR), the very-high-temperature reactor (VHTR), the supercritical water-cooled reactor (SCWR) and the gas-cooled fast reactor (GFR). All four SAs were extended in 2016 for another ten years. Co-operation on the molten salt reactor (MSR) and the lead-cooled fast reactor (LFR) systems takes place under Memoranda of Understanding (MOU). The participation of GIF members in SAs and MOU is also shown in Table 1.1.

1.2. GIF organisation

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

Figure 1.1: **GIF governance structure in 2016**



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

As detailed in its charter and subsequent GIF policy statements, the GIF is led by the Policy Group (PG) which is responsible for the overall steering of the GIF co-operative efforts, the establishment of policies governing GIF activities, and interactions with third parties. Every GIF member nominates up to two representatives in the PG. The PG usually meets twice a year. In 2016, the two PG meetings were held in Paris in April, hosted by the Nuclear Energy Agency (NEA) at the OECD Conference Centre, and in Seoul in October, hosted by Korea (Figure 1.2).

Figure 1.2: **Policy Group in Paris at the Château de la Muette (April 2016)**

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

Signatories of each SA have formed a System Steering Committee (SSC) in order to plan and oversee the R&D required for the corresponding system. R&D activities for each GIF system are implemented through a set of project arrangements (PAs) signed by interested bodies. A PA typically addresses the R&D needs of the corresponding system in a broad technical area (e.g. fuel technology, advanced materials and components, energy conversion technology, plant safety). A Project Management Board (PMB) is established by the signatories to each PA in order to oversee the project activities described in a detailed multi-annual Project Plan (PP) that aims to establish the viability and performance of the relevant Generation IV system in the technical area concerned. Until the PA is signed, a provisional project management board 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 require unanimous approval of the corresponding SSC. The PG may provide recommendations to the SSC on the participation in GIF R&D projects by organisations from non-GIF members.

Three methodology working groups (MWGs) – the Economic Modelling Working Group (EMWG), the Proliferation Resistance and Physical Protection Working Group (PRPPWG), and the Risk and Safety Working Group (RSWG) – are responsible for developing and implementing methods for the assessment of Generation IV systems against GIF goals in the fields of economics, proliferation resistance and physical protection, and risk and safety. The MWGs report to the EG which provides guidance and periodically reviews their work plans and progress. Members of the MWGs are appointed by the PG representatives of each GIF member.

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

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. A revision of the SIAP Charter was approved in April 2016, and was followed by a renewal of a large part of the membership through nominations by PG members and approval under written procedure.

The GIF Secretariat is the day-to-day co-ordinator of GIF activities and communications. It includes two groups: the Policy Secretariat and the Technical Secretariat. The Policy Secretariat assists the PG and EG in the fulfilment of their responsibilities. Within the Policy Secretariat, the policy director assists the PG on policy matters whereas the Technical Director serves as Chair of the EG and assists the PG on technical matters. The Technical Secretariat, provided by the NEA, supports the SSCs, PMBs, MWGs and TFs, 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, Japan withdrew from the two signed SCWR project arrangements, materials and chemistry, and thermal-hydraulics and safety, but remains a member of the SCWR System Steering Committee having signed the extension to the system arrangement. Table 1.2 shows the list of signed arrangements and provisional co-operation within GIF as of 31 March 2017.

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 March 2017**

	Effective since	CA	EU	FR	JP	CN	KR	ZA	RU	CH	US
VHTR SA	Extended 30 Nov 2016		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	Extended 16 Feb 2016		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	Extended 30 Nov 2016	X	X		X	X			X		
M&C PA	6-Dec-10	X	X		O	S			O		
TH&S PA	5-Oct-09	X	X		O	S			O		
SIA PA	Provisional	P	P		P	P			P		
GFR SA	Extended 30 Nov 2016		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	X

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

PROJECT ACRONYMS

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

Chapter 2. Highlights from the year, Vice-Chair reports and country reports

2.1. General overview

With all GIF members but Australia having signed the GIF Framework Agreement (FA) extension, the year 2016 marked the completion of this process. In 2016, the Australian government signed the GIF Charter, thus making Australia a full member of the GIF. This has allowed its FA process to be initiated, whose completion is expected in 2017. The Australian Nuclear Science and Technology Organisation (ANSTO) will most likely become the implementing agent.

With regard to the GIF System Arrangements (SA), the SFR SA was extended in February 2016, while the VHTR, SCWR and GFR SAs (which all expired on 30 November 2016) are in the process of being extended. To become effective, these SAs will have to be signed by at least two signatories.

GIF maintains a long-standing collaborative relationship with the International Atomic Energy Agency (IAEA) with emphasis on the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). Co-operation on evaluation methodologies for economics, safety, physical protection, and proliferation resistance, as well as on GIF SFR safety design criteria (SDC) and safety design guidelines (SDG), has been ongoing for several years. The 10th GIF-INPRO Interface Meeting was held in April 2016 in Vienna, Austria. In addition to the discussion of potential areas of mutual interest and of the GIF-IAEA collaboration matrix, topics of the interface meeting included the presentation of the GIF reactor system development status, of the IAEA reactor technology development activities, as well as safety, proliferation resistance and economics topical sessions. Collaboration in terms of SFR SDC and SDG development had its main outcome in the 6th GIF-IAEA Joint Workshop on Safety held in November 2016 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. The 6th workshop focused on the presentation and discussion of the resolution of the comments made by the IAEA in its role of an external reviewer of the GIF SFR SDC Phase I Report.

GIF representatives, including members of the SFR SDC TF, attended the September 2016 meeting of the Ad hoc Group on the Safety of Advanced Reactors (GSAR), formed in co-operation with the NEA Committee on the Safety of Nuclear Installations (CSNI) and Committee on Nuclear Regulatory Activities (CNRA). Addressing sodium-cooled fast reactor safety aspects, the meeting offered, among other, the opportunity to introduce the GIF SDC TF's preliminary SFR System SDG approach report and initiate discussions in view of the GSAR review of this report.

In 2016, the GIF Education and Training Task Force (ETTF) became fully operational. It comprises 13 members representing all the active GIF members and covers all the GIF systems. In addition to maintaining a social media platform and collaborating in the 2016 Frédéric Joliot/Otto Hahn Summer School on Liquid Metal Fast Reactors, the ETTF held 4 of the planned 13 webinars addressing the GIF systems and cross-cutting topics.

2.2. Highlights from the Experts Group

The Experts Group 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.

SFR SDC Task Force (SFR SDC TF)

In response to the 2016 update of the IAEA Safety Standard Report SSR 2/1 (based on lessons learnt from the Fukushima Daiichi events), and within the framework of the 6th GIF-IAEA "Joint Workshop on Safety of Sodium-Cooled Fast Reactors" (November 2016), the SFR SDC TF reviewed the resolution of the comments made by the external reviewers of the SFR SDC Phase I report. At the same workshop, IAEA's preliminary review findings of the SFR SDC TF Phase II draft report on "SFR Safety Design Guidelines on Safety Approach and Design Conditions for Generation IV Sodium-Cooled Fast Reactor Systems" (focusing on prevention and mitigation of severe accidents and on inherent/ passive safety features) were presented and discussed with participants from IAEA, GIF countries, and national regulators from France, Russia and the United States. The Phase II draft report was also submitted to the GSAR for review. Presentations were made by the SFR SDC TF and discussions held at the fourth GSAR meeting (September 2016). The GSAR review of the Phase II draft is ongoing and expected to be concluded in 2017. Currently, the SFR SDC TF is working on three complementary SDG documents, viz. on key structures, systems and components. The SFR SDC TF is planning to extend its activities to the SDC/SDG development for the LFR and VHTR systems. I doing so, it will follow its established work methodology and build on the experience gained with the SFR SDC/SDG development. Moreover, it will pursue collaboration opportunities with the IAEA: in the case of the VHTR system by leveraging on GIF member countries contributions to an IAEA Coordinated Research Project, and in the case of the LFR system by using the established mechanism of joint GIF-IAEA workshops.

Education and Training Task Force (ETTF)

The Education and Training Task Force (ETTF) became fully operational in 2016. All active GIF members have nominated ETTF representatives through the GIP PG. Thematically, the ETTF covers all six GIF systems. The ETTF established and is maintaining a Gen IV information exchange and discussion forum on LinkedIn (www.linkedin.com/groups/8416234). The ETTF partnered with the Commissariat à l'énergie atomique et aux énergies alternatives (CEA) and Karlsruhe Institut für Technologie (KIT) in the 2016 Frédéric Joliot/Otto Hahn Summer School (FJOHSS) on Liquid Metal Fast Reactors, with the GIF logo being displayed on the FJOHSS material, the school being advertised on the GIF website, and the ETTF Co-Chair lecturing at the school. The ETTF is implementing a series of 13 Gen IV webinars on GIF systems and cross-cutting topics. Four of these webinars were held in 2016: "Atoms for Peace – The Next Generation", "Closing the Fuel Cycle", "Introduction to Nuclear Reactor Design", and "Sodium Cooled Fast Reactors". The topics and lecturers roster for the remaining nine webinars, to be held in 2017, were defined. All webinars are archived and made accessible from the GIF website.

Sustainability Task Force and frequently asked questions

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 TF also looked at what activities on sustainability have been performed by the IAEA, the NEA, and the US Fuel Cycle Options Study. Before embarking on a possible second phase of the Task Force, the PG has decided to request feedback from the GIF SSCs and pSSC on their respective views and opinions on the sustainability concept. The GIF TD has initiated this process by sending out a questionnaire to the GIF SSC and pSSC chairpersons.

A draft summarising 13 sustainability related frequently asked questions (FAQ) was developed and is currently being reviewed by the GIIF EG. After approval by the GIF PG, the FAQs will be published on the publication on GIF website.

Periodic SSC, pSSC, MWG and TF highlights

To improve communication and enhance both visibility and recognition of GIF's R&D achievements, starting in 2016 one pager highlights are drafted by the systems SSCs and pSSCs, as well as by the MWGs and TFs twice a year after the respective committee meetings. In addition to being used internally by the GIF members, the highlights will also facilitate drafting of Annual Report contributions.

GIF comments to the IRSN report

In 2016, a draft report was prepared (based on GIF system and provisional system Steering Committee inputs), summarising the GIF comments to the 2014 IRSN report on the safety and radiation protection review of the six GIF systems. The draft was reviewed by an EG member and is now being reviewed by the RSWG. The results of this review are expected after RSWG's next meeting, by the end of April 2017.

Fourth GIF Symposium

The 4th GIF Symposium will be held in conjunction with the 2018 fall GIF meetings in Paris. Preparation of the symposium has started with the Scientific Technical Committee being assembled and the basic structure of the two-day conference defined. The symposium will document progress made with regard to the development of the Gen IV systems. It will highlight the merits of the forum in developing innovative nuclear energy systems that are aligned with sustainable development criteria, thus enhancing GIF's recognition. Based on the progress made over the last decade and a half, the symposium will strive to present a credible path forward towards the goal of establishing nuclear energy as a necessary element in the long-term sustainable carbon-free energy mix. With a detailed Proceedings document abiding to high scientific standards and summarising findings, conclusions and recommendations (reviewed and approved by all GIF stakeholders) as output, the symposium is expected to provide valuable guidance for an update of the GIF Roadmap.

SIAP charges for 2016

The EG has identified the following two special charge s for the SIAP:

- to provide the panel's view and guidance on how the sustainability benefits of nuclear energy can be made more attractive to investors in both existing and future energy markets;
- to provide the Panel's view and guidance on i) which the key steps to deploy Gen IV reactors are; and ii) which R&D demonstration objectives should be associated with those steps.

Cross-cutting R&D topics

The GIF Technology Roadmap process included working groups dedicated to the definition of GIF systems assessment methodologies (in terms of risk and safety, proliferation resistance and physical protection, as well as economics modelling), and working groups dedicated to cross-cutting features and technologies of advanced nuclear systems (specifically: fuel cycle, fuels and materials, risk and safety, economics and energy products). More recently, the need to address innovative reactor cross-cutting R&D issues was noted in the 2012 US ARC Initiative Technical Review Panel Report, as well as in recommendations received from the SIAP at the October 2015 GIF St. Petersburg meetings. The former identified innovative reactor R&D cross-cutting areas in terms of advanced reactor licensing frameworks, novel power conversion technologies based on the Brayton cycle, advanced reactor modelling and analysis methods, accident-tolerant fuel development, as well as the integration of advanced reactors in the low-carbon energy concepts. The latter recommended considering industry related R&D cross-cutting topics advancing the time-to-market issues (i.e. reactor demonstration phase, licensing, as well as codes and standards), addressing the new market needs and assessing the role of nuclear energy in an energy mix with an increasing share of renewables. During 2016, the EG has worked on identifying cross-cutting R&D topics and related tasks. The status of these efforts was discussed at a side meeting held in conjunction with the GIF 2016 fall meetings attended by EG, SSC, pSSC, MWG and TF members. It is noted that the MWGs and SIAP are already addressing cross-cutting issues linked to licensing, safety and commercialisation aspects. The newly established ETTF will also provide a forum for addressing such issues. The establishment of well-defined responsibilities for the three GIF PG Vice-Chairs (regulatory issues, marketing issues, international co-operation) has opened up the possibility to structure GIF R&D cross-cutting efforts according to these responsibility areas: market related R&D cross-cutting issues involve the Vice-Chair on Market Issues, SIAP, the EMWG as well as the various SSCs and pSSCs; risk, safety and regulatory issues involve the Vice-Chair for Regulatory Issues, the RSWG, the SDC/SDG TF as well as the various SSCs and pSSCs. In addition to the issues linked to the Vice-Chair missions, a variety of generic advanced reactor R&D cross-cutting topics has been identified, e.g. in the areas of proliferation resistance and physical protection, liquid metal-cooled reactor systems (SFR, LFR) R&D, gas-cooled reactor systems (VHTR, GFR, FHR) R&D, materials science, modelling and simulation, power conversion systems, reactor instrumentation, etc.

R&D infrastructure

As part of the overall efforts to identify ways and means to enhance the GIF R&D collaboration, the EG (with support from the SSCs, pSSCs, MWGs and TFs), under the purview of the PG Vice-Chair for External Collaboration, is establishing a database on existing and planned R&D infrastructure in GIF member countries.

2.3. Report of the GIF Vice-Chair for Regulatory Issues, John E. Kelly

As the GIF Vice-Chair responsible for regulatory issues, I oversee GIF activities related to safety and regulatory frameworks. This includes promoting the external review of the SDC and SDG for the sodium fast reactor (SFR), leading the engagement with the Ad hoc Group on the Safety of Advanced Reactors (GSAR, created by the Committee of Nuclear Regulatory Activities and the Committee of the Safety of Nuclear Installations), extending the SDC/SDG development to other Gen IV system, engaging with System Steering Committees on safety research priorities, and keeping the Policy Group informed of these safety research priorities.

For the past several years, a GIF Task Force developed safety design criteria and safety design guidelines for the sodium fast reactor. The SFR SDC report was distributed for external review to national regulators and international organisations, such as the Multinational Design Evaluation Programme (MDEP), the Nuclear Energy Agency, and International Atomic Energy Agency (IAEA). Several workshops were held during 2013 and 2014 at the IAEA with reactor designers, regulators and safety experts to review the SDC/SDG for SFR.

In December 2014, at the NEA, GIF presented the SFR SDC to a meeting of the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI). The outcome was initiation of an ad hoc group that will co-ordinate international regulatory discussion on licensing of Generation IV reactors. GIF representatives attended GSAR meetings in 2015 and 2016. At the September 2016 meeting, a significant part of the agenda was devoted to a “deep dive” into the safety aspects of sodium fast reactors. At future GSAR meetings we anticipate having “deep dives” into the other GIF systems.

For the lead fast reactor (LFR) and high-temperature gas reactor, work has been initiated on the development of SDC/SDG as well. The Lead Fast Reactor System Steering Committee began an effort to develop SDC/SDG for the LFR. Using an IAEA Cooperative Research Project approach, SDC are being developed for high-temperature gas reactors. This progress on these efforts will be reviewed by the Risk and Safety Working Group to ensure consistency with the work that has been done on the SFR.

In 2014, IRSN published a report that reviewed the Generation IV systems. The GIF Technical Director has been working with the System Steering Committees to review this document and to provide an integrated response to IRSN. A draft of this response was completed in 2016, and will be reviewed by the RSWG prior to PG review.

In summary, over the last few years there has been significant progress in the interactions with the international regulator community on Generation IV systems. We anticipate that this dialogue will benefit not only GIF, but also the regulators and their technical support organisations.

2.4. Report of the GIF Vice-Chair on Market Issues, Hideki Kamide

Market issues for future deployment of Gen IV reactors are a common concern between the developer and user. The Senior Industry Advisory Panel (SIAP) put forward two important recommendations at the PG meeting held in October 2015; i) Identify the attributes of Gen IV systems most attractive for industry (vendor/utility), ii) Investigate market conditions and timelines for commercialisation of Gen IV reactors. The scope of the market issues are as follows:

- Better understanding of the drivers, opportunities and constraints related to the market environment for appropriate ways to carry out GIF activities.
- Close work with the SIAP, SSC chairs, and related TFs in carrying out its work and provide recommendations regarding the role and value of Gen IV systems in future market environments.
- Activities could take the form of surveys, economic evaluations, analysis of marketing issues development of end use options. Consideration should also be given to the development of deployment scenarios of Gen IV systems and the development of corresponding utility/end user requirements documents.

According to this scope, a two-year programme of three phases was proposed as a work plan and confirmed in the PG meeting in October 2016.

Phase 1: Survey of key points on the market issue

The following issues have been preliminarily picked up through the discussions with SIAP and EMWG:

- national and international market driver;
- opportunity (small modular reactors [SMR], integration of renewables and harmonisation, non-electric applications to exchange with fossil fuel heating);
- constrains;
- analysis of the key issues for political decisions of energy mix and role of advanced reactors in each country, e.g. international agreement on 2-degree C scenario of global warming, and energy security.

Phase 2: Development of concept of evaluation to enhance the market drivers

Three aspects of the concepts are picked up:

- evaluate economy of Gen IV reactors taking accounts of other merits on the sustainability;
- increase the opportunity of advanced nuclear reactor;
- reduce the constrains of advanced nuclear reactor use.

Phase 3: Understand and value the attributes of Gen IV systems

Understand and value the attributes of Gen IV systems for the stakeholders on the market issues, funding of Gen IV systems, and risk.

A questionnaire for PG members will be prepared as the phase 1 survey.

2.5. GIF/INPRO Interface Meeting, Hark Rho Kim

The 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 10th GIF-INPRO Interface Meeting was held at the IAEA Headquarters, 11-12 April 2016. It provided the opportunity for an informal information exchange on GIF and IAEA activities covering various R&D areas. The GIF presented the status of the development efforts of all six GIF systems while the IAEA presented the status of its reactor technology development activities. In the safety area, the IAEA presented a revision of IAEA safety standards after the Fukushima Daiichi accident and a status report on GIF-IAEA workshop series on SFR SDC and planning for next workshop. The respective activities and the path forwarded in the area of proliferation resistance were presented, with IAEA presenting the status report on NE series of the Safeguards by Design. Status reports of the methodology activities were given, more specifically of the INPRO methodology and the GIF Economic Modeling Working Group. Last but not least, other activities briefings of mutual interest, e.g. 13th INPRO Dialogue Forum on legal and institutional issues in the global deployment of SMRs, overview of the GIF Education and Training Task Force, IAEA's FR and HTR knowledge management activities were discussed.

INPRO Steering Committee

The 25th Meeting of the INPRO Steering Committee was held at the IAEA Headquarters, Vienna, from 31 October to 2 November 2016. The participants came from 24 member states, NEA and international initiatives, GIF and IFNEC. The INPRO/IAEA Secretariat provided the 2016-2017 INPRO Subprogramme Plan and progress reports since the last Steering Committee Meeting. New projects which will start next year were presented, and a draft of INPRO Vision for 2018-2021 as a strategic plan was also introduced. The vision will be circulated early next year. The INPRO Steering Committee confirmed the endorsement of the final 2016-2017 INPRO Subprogramme Plan, and thus the INPRO Subprogramme Plan for 2016-2017 was approved.

It was noted that international collaborative organisations and initiatives such as IAEA/INPRO, GIF, IFNEC and NEA are essential in looking for a mutually beneficial approach to the expansion of the peaceful uses of nuclear energy.

Nuclear Innovation 2050

The 3rd Advisory Panel Nuclear Innovation 2050(NI2050) was held at the NEA, Paris, from 20-21 June 2016. The 3rd advisory panel meeting concluded that there are two time frames to consider for NI2050: one for priority pre-competitive shared research and development to push innovative technologies to market for 2030, and another one for 2050 (and beyond for some technologies which will need more time). And the future of nuclear energy lies first (2030 perspective) in the safe and economic long-term operation of existing plants and on the ability to build new plants economically. This is much dependent on the evolving electricity market. R&D priorities needs to be defined assuming the market will become more effective with time. For the longer term (2050 perspective and beyond), it is impossible to know, for the present time, what are the real prospects for the commercialisation of Gen IV systems. R&D priorities need to be focused on providing viable options to allow industry to make commercial decisions at the appropriate time in the future. Therefore it seems reasonable to have some R&D priorities phased towards demonstration (as defined in the ToRs of the NI2050 Roadmapping). To reflect these two timelines and associated priorities, the Chair proposed to establish two subgroups of the Advisory Panel to analyse how R&D can be economically accelerated to bring innovative technologies to the market for commercial applications, in the 2030 and 2050+ perspectives, respectively.

2.6. Country reports

Australia

Since signing the GIF Charter in June to establish Australia's membership of the Generation IV International Forum, Australia and ANSTO have continued to work towards acceding to the Framework Agreement and its extension to become fully engaged with the forum. Administrative processes have commenced, with the Ministerial request for the necessary Parliamentary committee to consider the agreement expected shortly.

Following the anticipated accession to the agreement, and as noted in Australia's membership petition, Australia would expect to join the VHTR System Arrangement and Material Project Arrangement, the MSR MOU and the Risk and Safety Working Group. In due course, Australia could also contribute to the Education and Training, and Economic Modelling Working Groups.

As noted in the petition, ANSTO expects that the majority of Australia's initial contributions to GIF will be in the area of Nuclear Materials Science and Engineering and in anticipation, ANSTO's National Director of Australian GIF research has attended the VHTR System Steering Committee and the VHTR Materials Project Management Board meetings in November 2016 and plans to attend the MSR provisional System Steering Committee meeting in January 2017 to enable the integration of ANSTO's research into GIF in a timely manner.

ANSTO is also increasing its support of fusion energy research, signing a technical co-operation agreement with the International Thermonuclear Experimental Reactor (ITER) in September. Australia is the first non-ITER member to enter into such an agreement, which will allow ANSTO and Australian scientists to work with international experts on this important project.

Outside of Australia's membership in international fora, there has been continuing domestic consideration in South Australia of that state's role in the nuclear fuel cycle. The Premier of South Australia instigated a Royal Commission in March 2015 to examine the possible economic opportunities for further participation in the fuel cycle.

The commissioner tabled his final report in May this year, making 12 recommendations. Probably most notable among of these recommendations was that the South Australian government pursue the opportunity to establish a multinational spent nuclear fuel and intermediate level waste storage and disposal facility, should there be community support for such an action.

Obviously this would be a major undertaking for South Australia should it choose to proceed, and the state government is accordingly conducting a comprehensive state-wide public consultation process. This will form an important input for the South Australian Premier's response to the recommendations, expected to be delivered in late November. For its part, the national government is closely observing the South Australian process.

Canada

Nuclear power in Canada

The government of Canada's position is that nuclear energy, as a near emissions-free source of electricity, is safe, reliable and environmentally responsible, as long as it is developed within a robust international framework which adequately addresses security, non-proliferation safety and waste management concerns. Nuclear energy represents an important contribution to Canada's electricity mix. While the government of Canada has important responsibilities with respect to nuclear energy, investment decisions on energy supply mix and generation capacity, including the construction of new nuclear power reactors and the refurbishment of existing reactors, fall under provincial jurisdiction.

Nuclear energy developments

- Domestic

In the province of Ontario, a planned investment of CAD 25 billion over the next 15 years will extend the life of 10 nuclear reactors for another 25 to 30 years and maintain nuclear power capacity at 9.9 GWe. The first of these, unit 2 at the Darlington nuclear power plant, was taken offline in October 2016 for the start of a 40-month refurbishment. It is the first of the four units to be refurbished at Darlington station. In addition, the province of Ontario and Bruce Power reached an agreement to refurbish the remaining six units at the Bruce nuclear power plant. The first of these units, unit 6, is scheduled to come offline for refurbishment in 2020.

The Nuclear Liability and Compensation Act came into force on 1 January 2017. This Act sets the monetary limit for the liability of nuclear power plant operators to CAD 1 billion, to be phased in over four years from CAD 650 million at entry into force to CAD 1 billion beginning in 2020. The new liability amount – increased from the CAD 75 million under the 1976 Nuclear Liability Act – is commensurate with current international standards, notably the Convention on Supplementary Compensation for Nuclear Damage. Canada signed this convention on 3 December 2013, and expects to be officially a member in early 2017.

- International

In November 2015, at COP21, Canada announced its participation in Mission Innovation, a global initiative of 20 countries and the European Union, working together to accelerate clean energy innovation. As part of Mission Innovation, Canada will seek to double its baseline 2014-2015 funding for clean energy research, development and demonstration. Canada's baseline includes nuclear energy as a research and development focus area, including nuclear energy in its definition of clean energy.

The Protocol between the government of Canada and the government of Romania supplementing the agreement between the government of Canada and the Government of the Socialist Republic of Romania for Co-operation in the Development and Application of Atomic Energy for Peaceful Purposes, done at Ottawa, on 24 October 1977 came into force on 19 December 2016 permitting the export of tritium removal technology and equipment between the two countries.

Small modular reactors activities

The Canadian Nuclear Safety Commission (CNSC) has been approached by 18 SMR companies (Canadian and international) seeking information on Canada's regulatory process. As part of its services, the CNSC undertakes an optional preliminary step before the licensing process called a vendor design review (VDR). The VDR is completed at a vendor's request and expense to assess their understanding of Canada's regulatory requirements and the acceptability of a proposed design. Since the beginning of 2016, five SMR companies have started the VDR process with the likelihood that others will follow in the near term.

The CNSC also launched a discussion paper in May 2016 seeking input on potential issues it sees with licensing SMRs and how it plans to address them using existing regulatory tools and processes. The CNSC is currently reviewing the input received.

Natural Resources Canada (NRCan) is regularly engaging with SMR industry on SMR development in Canada. NRCan, in partnership with others Canadian organisations have concluded two studies, notably the Feasibility of the Potential Deployment of SMRs in Ontario (May 2016) and the Northern Indigenous Peoples and the Prospects for Nuclear Energy (July 2016).

Activities within GIF

In 2016, Canada signed the GIF Framework Agreement Extension. Subsequently, Canada signed the GIF Supercritical Water-Cooled Reactor (SCWR) System Arrangement Phase II as part of its continued active participation under this system.

Supercritical water-cooled reactor research and development

Canada is continuing the development of the Canadian SCWR concept after successful reviews by the Canadian nuclear industry in 2015 and international peers in 2016. Work in this phase focuses primarily on the verification and validation of key components of the concept, such as mechanical devices, manufacturing techniques and technology areas. Selection of these components is based on comments from reviewers and expertise of technical experts. A road map has been developed to lay out the strategy, plan and schedule for verification and validation of selected components. It specifies the verification or validation approach to be applied in various technology areas. The verification and validation tasks are subdivided into key technology areas: mechanical components, thermal-hydraulics, materials, chemistry, fuel channel behaviours, fuel, reactor physics and economic modelling. Outputs from these tasks will be contributed to GIF supporting Canada's participation in the development of a SCWR system.

The Canadian SCWR concept has been developed for large baseload power generation of 1 200 MWe, which would be excessive for off-grid small remote communities, mining operation and oil-sand production. Therefore, Canada is supporting the development of a scaled-down SCWR concept to generate 5-200 MWe power for these applications, in addition to the baseload concept. The modular configuration of the SCWR concept provides the flexibility in adjusting the power output to meet the local deployment needs.

China

Nuclear Energy Policy

In September this year, the G20 leaders summit was successfully held in Hangzhou, China. The summit placed the development dimension at the forefront of the global macroeconomic policy framework and for the first time developed an action plan for the implementation of the United Nations 2030 sustainable development agenda. Nuclear energy and nuclear technology in the protection of energy security, response to environmental pollution and climate change, strengthening the areas of anti-terrorism and security are widely used and will play a greater role in the sustainable development.

The Chinese government attaches great importance to sustainable development and climate change. On 3 September 2016, the Standing Committee of the National People's Congress of China approved China's accession to the "Paris Climate Change Agreement". China will develop nuclear energy as the important initiatives to promote energy conservation and pollution prevention, stabilise economic growth, optimise the energy structure and achieve green, sustainable development.

"Atomic energy law" and "nuclear safety law" have been included in the National People's Congress legislative plan. "Nuclear Security Regulations (draft)" and "Nuclear Power Management Regulations (draft)" were officially opened to solicit public opinion on 3 June and 19 September 2016, respectively.

Operation and construction of nuclear power plants

The in-service nuclear power units have kept a good record in safety and operation performance, and the projects under construction are progressing as scheduled. The first AP1000 reactor unit at Sanmen NPP is under hot tests. By the 11 October 2016, there are 35 units in operation, 20 units under construction, with the total installed capacity of 56.9 GWe in China's mainland. The country plans to have around 90 reactors in operation or under construction by 2020, with nuclear energy supplying about 4% of its electricity by then.

Gen IV nuclear energy systems R&D

On 23 June 2016, China signed the ten-year Extension Agreement of the Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy Systems at the OECD.

On 3 August 2016, China signed the System Agreement for the International Research and Development of the SFR Nuclear Energy System (Phase II). The SCWR and VHTR System Agreement extensions would also be signed soon.

VHTR

The R&D on VHTR has made encouraging progress. The equipment installation of HTR-PM demonstration plant was started in 2015. The first reactor pressure vessel was installed in March 2016. The review of Final Safety Analysis Report of HTR-PM was started in beginning of 2016. It is scheduled to be finished in beginning of 2017. The HTR-PM is scheduled to be connected into grid in the end of 2017. The research and development of 600 MWe HTR power plant, which includes six reactor modules in one unit, has been started in China already, which will further push forward the commercial deployment of V/HTR technology.

SFR

For SFR, since the last April, China Experimental Fast Reactor (CEFR) has been under maintenance, and the reactor is expected to be restarted at the end of 2016. The preliminary design of the demonstration fast reactor, CFR600, has been carried on gradually. Under the GIF SFR R&D framework, China Institute of Atomic Energy took a step forward on each of the projects; as one of the design tracks in the GIF SFR System Integration and Assessment Project, the pre-conceptual design of CFR1200 is going smoothly and the main effort lies in the study of inlet temperature of the reactor core; for the Safety and Operation Project, progress is obtained in the simulation of natural circulation of core catcher of CEFR and in the code development at transient analysis; for the Component Design and Balance-of-Plant (CD&BOP) Project, the China Institute of Atomic Energy (CIAE) did a lot of work in the study of supercritical CO_2 Brayton Cycle; for the Advanced Fuel Project, the GIF SFR AF Project Management Board (PMB) meeting was hosted successfully by CIAE on 21-23 March 2016 in Beijing.

SCWR

The 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. In terms of co-operation in SCWR, the Nuclear Power Institute of China (NPIC) and the Canadian Nuclear Laboratories (CNL) of Canada is preparing a new international benchmark exercise based on the SCW 2×2 rod bundle tests for assessing the computational fluid dynamics (CFD) models.

NPIC and China Liaison Office for GIF co-hosted The 11th GIF SCWR Information Exchange Meeting, which took place on 14-16 March 2016 in Chengdu, China. Over 75 participants attended it and 27 presentations associated with SCWR were delivered by GIF member countries and IAEA. The 8th International symposium on SCWR (ISSCWR-8) will be held in Chengdu, China on 13-15 March 2017, it is organised by NPIC in co-operation with China Nuclear Energy Association (CNEA), GIF-SCWR-SSC and IAEA. All the specialists of SCWR R&D in the world are welcome to attend this event in the SCWR field.

For these three systems that China had officially joined in, Ministry of Science and Technology and CAEA in China give much attention and try to gather the resource by supporting projects these years.

For the other systems, Chinese Academy of Sciences (CAS) has allocated several projects on LFR and MSR.

Reactor technology innovation

The CIAE successfully carried out a zero power critical experiment on low-enrichment uranium renovation project for the Ghana atomic micro reactor on 27 July. It is a significant landmark in the project, marking completion of all the technical preparations in China. It is also another breakthrough by China in practising its international commitment to co-operate in lessening high-enrichment uranium usage by putting the first micro reactor low-enrichment renovation project into full-power operation.

On 13 July 2016, China's Independent Nuclear-grade Digital I&C Platform-'FirmSys' passed the Review of IAEA. The Independent Engineering Review of I&C Systems (IERICS) Report was received by CGN.

Nuclear emergency preparedness

China's first state-level nuclear emergency rescue team was established on May 24. The team consists of 320 people in 6 groups: co-ordination and technical support, emergency rescue, project rescue, radiation monitoring and protection, decontamination and medical rescue. They will receive training in accident scenario simulation, operating skills and theory. The team will work on domestic major nuclear accident emergency rescue and can also assist internationally.

Euratom

The Commissioner for Education, Culture, Youth and Sport signed the Extension of the Framework Agreement on behalf of Euratom on 10 November 2016.

Euratom will proceed with the signature of the System Arrangement Extensions for the SFR, SCWR, VHTR and GFR in early 2017.

ENERGY

The European Commission (EC) has published a new Nuclear Illustrative Programme (PINC), pursuing Art. 40 of the Euratom Treaty, as announced in the state of the energy union, issued in November 2015. The objectives of the PINC are to stimulate action by persons and undertakings and to facilitate co-ordinated development of their investment in the nuclear field. As concerns R&D, the programme underlined the fast development of the use of nuclear energy outside the European Union (EU) and calls for keeping EU excellence in the technological and safety areas, for which continuous investment in research and development activities will be essential. Specifically Generation IV is been included in the research: "The implementation of the European Sustainable Nuclear Industrial Initiative which aims to prepare the future deployment of the Generation IV nuclear system relying on fast neutron technology with a closed fuel cycle. Several reactors are in the research stage (e.g. ALLEGRO, ALFRED, MYRRHA and ASTRID), which may already advance significantly by 2050".

In September 2016, the European Economic and Social Committee made comments on the published version of the PINC. According to these recommendations, the main topics to be re-addressed include:

- the competitiveness of nuclear power in the short, medium and long term;
- the related economic aspects;
- contribution to security of supply;
- climate change and carbon targets;
- public acceptability, liability for nuclear damages, transparency and effective national dialogue.

A decision on the revision of the PINC will be made later.

▪ SET-Plan: The European Strategic Energy Technology Plan (SET-Plan)

The European Strategic Energy Technology Plan (SET-Plan) aims to accelerate the development and deployment of low-carbon technologies. It seeks to improve new technologies and bring down costs by co-ordinating research and helping to finance projects. Within the SET-Plan the priority for nuclear energy is to support the development of the most advanced technologies, to maintain the highest level of safety in nuclear reactors, to improve the efficiency of operation, the back end of the fuel cycle and decommissioning. European Commission and MS have prepared an issue paper on nuclear energy agreed targets and Generation IV is one of the technologies that is included. The SET-Plan plenary met in Bratislava on the 1 and 2 December 2016.

Research

▪ EC-DG RTD

Euratom Horizon2020 Fission Call 2016-2017 closed on 5 October 2016.

There are specifically Gen IV topics with a budget of around EUR 40 million for two years for the topics below:

- topic 2 : ESNII systems (fast reactors);
- topic 3: Closed fuel cycle development;
- topic 4: Material for Gen 4;
- topic 5: Small modular reactors;
- topic 12: Education and training.

Seventy-two proposals have been received for this call. The expert desktop evaluation has started and the Evaluation Committee will meet in December 2016 to finalise the winning projects. Selected proposals will be announced in January 2017.

EC-JRC activity

Generation IV Research activity is included in its Work Programme 2017-2018 in project Safety of Advanced Nuclear Systems and Innovative Fuel Cycles (SANSIF) and Non-Proliferation and Strategic Trade Control (NPTC).

This project deals with the safety of planned advanced fission reactors and fuel cycle technologies in the European Union. Research covers safety assessment of:

- advanced fuels recovery and conversion;
- properties and behaviour of advanced materials;
- properties and irradiation performance of innovative fuels;
- safety performance code development for advanced reactors;
- innovative concepts and methodologies for nuclear safeguards and non-proliferation.

Within it there is support to other international organisations (IAEA, NEA) and support to EU platforms (ESNII, EERA-JPNM, NC2I)

JRC currently participates in a number of RTD calls proposals covering essentially all Gen IV systems.

Euratom co-ordination meeting: The JRC organised the yearly Euratom-GIF co-ordination meeting on 12 December 2016 in Brussels. This meeting brought together the Euratom research community involved in Generation IV research and development, included JRC activities, EC-funded project participants or nationally funded project participants. A draft report on ten years of Euratom activities related to GIF was distributed to the participants at that meeting.

France

Nuclear energy policy in France

In accordance with the 2015 French Energy Transition Law, a “multi-year national energy plan” (Planification Pluriannuelle de l’Énergie, PPE, in French) for France up to 2023 has been published in November 2016 by the French Energy Ministry following a formal consultation process.

Regarding nuclear power, the key terms of the energy plan are the following:

- Depending on the evolution of France electricity demand, nuclear production would be reduced between 10 and 65 TWh by 2023, compared to an annual production of 410 TWh in average over the last five years. This reduction of nuclear production could come from a reduction of nuclear capacity or load factor.

- The two-year formal process to close the Fessenheim nuclear power plant would be initiated by EDF in the near future.
- The French closed nuclear fuel cycle strategy remains unchanged. In case some of the reactors that currently using MOX would be shut down, additional pressurised water reactors (PWRs) could be modified to start using MOX.

Governance of French nuclear public bodies and companies

In 2016, the governance of French nuclear public bodies has been reformed:

- A new decree specifies the role, missions and governance of CEA with defence and civil nuclear as the first two missions of the research organisation.
- A new decree also specifies the role and governance of IRSN as a technical support for the nuclear safety authority and as a research institute in the area of nuclear safety. Its expertise can also be solicited in complex situations such as during the Fukushima Daiichi nuclear accident.

In parallel, Areva, EDF and CEA decided to create a tripartite consultative body – the French Nuclear Platform – to improve the joint effectiveness of the three entities through a shared vision of the medium- and long-term goals for the sector, which will contribute to the preparation and implementation of decisions taken by the French Presidential Nuclear Policy Council.

Recent developments in the French nuclear industry

A restructuring of the French nuclear industry is progressing in line with the strategic orientations decided by the French government. In June 2016, Areva outlined its restructuring plan following the signature of a Memorandum of Understanding setting out the terms and conditions for EDF to take a majority share in Areva's reactor business, Areva NP.

Regarding Areva fuel cycle assets, a new entity – tentatively referred to as "Areva New Co" and dealing with mining, front-end back-end activities – has been approved by Areva's shareholders in September 2016. It is expected that part of Areva parent company's (Areva SA) debt will be transferred to Areva New Co.

At the beginning of 2017, a total of EUR 5 billion for capital increase would then be divided between Areva parent company and Areva New Co. Following the transaction the French state will hold, either directly or indirectly, at least two-thirds of Areva New Co's capital, with the remainder held by strategic investors.

In parallel, Areva TA, the subsidiary in charge of naval propulsion and research reactors, has been taken over by the French government, the CEA, EDF and the naval shipbuilder DCNS Group. The aim of this transaction is to consolidate the French naval nuclear propulsion sector, of which the CEA, DCNS Group and Areva TA are the main players, and which is one of the pillars of the French nuclear deterrent force.

Construction of the EPR reactor in Flamanville

Jean Bernard Levy, EDF Chief Executive Officer, committed in late 2015 to a new schedule with three milestones for the start of the reactor by 2018. In February 2016, EDF reported that the construction is on track with the completion of the first milestone: the installation of large components.

In agreement with the French Nuclear Safety Authority, Areva and EDF decided to extend until the end of 2016 the programme for testing the mechanical properties of Flamanville 3's reactor pressure vessel. These tests have been initiated after the discovery of carbon concentration higher than the recommended level and will now be carried out on three samples (instead of two). Following these tests, the French Nuclear

Safety Authority will then produce a formal opinion regarding the safety of the EPR pressure vessel.

Pending approval from the safety authority, Flamanville 3 completion schedule remains unchanged with a start of the reactor by the end of 2018.

Progress of the ASTRID International Project

In December 2015, CEA completed the conceptual design phase of the ASTRID reactor including the preparation of a safety option report. Following this important milestone, the project is moving forward with the basic design phase until the end of 2019.

In 2016, this included the delivery and improving of a number of experimental facilities and modelling tools. For instance, the robot VENUS was installed in June 2016 at CEA Cadarache in order to conduct research on acoustic visualisation in a sodium environment.

In parallel, the performances of the computer code SIMMER have recently been significantly improved. A new version of this code has been co-developed by JAEA and CEA and is used to simulate serious accidents in sodium fast reactors. Thanks to the use of high-performance computers, the speed of the thermal-hydraulic module of the code has been increased by a factor around 30, which greatly facilitates simulation studies for the ASTRID Project.

Organisation of the World Nuclear Exhibition in Paris

The second edition of the World Nuclear Exhibition (WNE) took place at Paris Le Bourget in June 2016. This international and business-oriented event covers the whole nuclear energy field and gathered more than 600 exhibitors and nearly 9 000 visitors from all over the world.

Emmanuel Macron, then French Economy Minister, opened the event and praised the “dynamism of the sector”, and confirmed that “nuclear power, far from being confined to the past, is definitely an option that addresses the challenges of the 21st century”.

A number of prizes were presented in parallel to WNE. In particular, the remotely operated laser-cutting system MAESTRO – developed by CEA and implemented by ONET – received the Société Française d’Energie Nucléaire (SFEN) award for technological innovation and was nominated for the WNE innovation award.

This technology is especially suited to cutting very thick materials in a hazardous nuclear environment. It allows for easy remote operation with high position tolerance for cutting heterogeneous layers of materials while generating fewer aerosols than other available techniques. In 2016, this technology has demonstrated its full industrial potential in the ongoing project to dismantle dissolvers in a spent fuel reprocessing facility at the CEA Marcoule site in France.

ICERR Affiliates’ agreement signed between CEA and three research institutes

During the IAEA General Conference in Vienna in September 2016, Daniel Verwaerde, CEA Chief Executive Officer, signed agreements with the heads of the Centre National de l’Energie des Sciences et Techniques Nucléaires (CNESTEN, Morocco), the Jožef Stefan Institute (JSI, Slovenia) and the Centre National des Sciences et Technologies Nucléaires (CNSTN, Tunisia) in the framework of the IAEA International Centre based on Research Reactors (ICERR) initiative.

This international recognition will facilitate the use of CEA’s research reactors for education and trainings project, hands-on training and R&D projects and will lay the ground for the use of the Jules Horowitz reactor that is currently under construction.

Japan

Current status of nuclear policy

At the Ministerial Meeting for the Nuclear Energy Policy held in December, 2016, the new policy for fast reactor development in Japan was decided. It aims to develop a new fast reactor with both high level of safety and economic efficiency while maintaining and developing the world's highest level of technical infrastructure, achieve its commercialisation and maximise its leadership in international standardisation of fast reactors. Also it states that "Strategic Roadmap" (provisional name) that specifies development tasks in the coming ten years will be developed in around 2018.

Regarding the Prototype Fast Breeder Reactor Monju, the government's policy was decided, which defines that Monju transitions to decommissioning without resuming operation as a nuclear reactor and takes on a new role in the future fast reactor development.

As for R&D of high-temperature gas-cooled reactors (HTGR), the industrial-academic-government forum for HTGR was established in April 2015 and regularly discusses the vision of future commercial HTGR and challenges towards its realisation.

In regards to nuclear fuel cycle industries, in order to steadily implement the spent fuel reprocessing business amid the change of the business environment due to deregulation in the electricity market, the Nuclear Reprocessing Organization of Japan (NuRO) was established in October 2016. It is stipulated by law that the new organisation develops a master plan of overall nuclear reprocessing projects, decides the amount of and collects the contributions, and carry out the reprocessing activities. The external experts will participate in the organisation's decision making and the government will also make certain level of involvement in order to ensure its governance. The actual reprocessing activities, however, will continue to be commissioned to the Japan Nuclear Fuel Ltd (JNFL), which has accumulated the necessary technology and human resources.

Fukushima Daiichi nuclear power station (F1)

Dismantling of the wall panels of unit 1 building cover for removal of fuel from the spent fuel pool has been completed in November 2016.

In order to isolate the reactor buildings from ground water flows, freezing of the soil for the land-side impermeable wall started in March 2016. The freezing work for the sea side of land-side impermeable wall was completed in October 2016. As for four of five locations on the mountain side that had not been frozen, the frozen state was confirmed by the Nuclear Regulation Authority (NRA). The frozen state of the remaining one location will be judged from the change in the amount of ground water pumped up from wells around the reactor buildings

Safety review of Nuclear Power Stations (NPSs) and nuclear fuel cycle facilities by the Nuclear Regulation Authority (NRA)

Safety review applications for the restart of 26 units of 16 NPSs under the new regulation were submitted to the NRA, and as a result of NRA's review, 8 units of 4 NPSs have obtained permission of licence amendment (as of the end of December 2016).

Specifically, Kyushu Electric Power Company's Sendai NPS units 1 and 2, Shikoku Electric Power Company's Ikata NPS unit 3 and Kansai Electric Power Company's Takahama units 1 to 4 and Mihama NPS unit 3 obtained the permissions. Ikata unit 3 resumed commercial operation in September 2016.

The reprocessing facility and MOX fuel fabrication facility of Japan Nuclear Fuel Ltd and other nuclear fuel facilities are under safety review.

Circumstances of Japan Atomic Energy Agency (JAEA)'s owned facilities

In August 2016, JAEA has submitted to the NRA a report, summarising efforts to be made on security measures for the Prototype Fast Breeder Reactor Monju in response to the NRA's order issued in May 2013, which includes the suspension of preparation for commissioning (this report is the complete revised report submitted in December 2014). In addition, JAEA started maintenance work under the revised maintenance plan with renewed technical grounds. In December 2016, however, the decision of Monju to decommission was made. JAEA will carry out the decommissioning of Monju safely and reliably and proceed with its examination in regards to the new role of Monju in R&D of fast reactors.

As for the Experimental Fast Reactor Joyo, JAEA is preparing an application for alteration in the instalment licence for restart under the new regulatory standards by the end of this fiscal year (as of the end of March 2017). In addition, JAEA will develop an operational plan which includes Joyo's utilisation in international co-operation activities.

For the safety review of the High-Temperature Gas-cooled Test Reactor (HTTR) under the new regulation, JAEA prepared for meetings with and hearings from the NRA and completed discussions about evaluations against natural phenomena and safety. It will carry out measures for the seismic evaluation, aiming at the early restart. Meanwhile, JAEA has successfully produced hydrogen for 31 hours using a testing device made of industrial materials with the thermochemical iodine-sulphur process, an innovative hydrogen production technology, which is planned to be connected to the HTTR in the future.

Korea

Korea is currently operating a total of 25 nuclear power plants, including the Shin-Kori unit 3, which entered commercial operation in December 2016. Nuclear electricity from 25 reactors accounts for 22.5% (21 716 MWe) of all electric capacity in Korea. The construction permit for Shin-Kori unit 5 and 6 was granted by the Nuclear Safety and Security Commission.

In July 2016, the Ministry of Science, ICT and Future Planning established "the Strategy of Demonstration for Technology able to Reduce Volume and Radio-toxicity of Spent Fuel", which was passed by the 6th Nuclear Promotion Committee. According to this strategy, a joint research between Korea and United States will be conducted until 2020 for the feasibility study of the pyroprocessing technology for non-proliferation and economy.

At the same time, the Ministry of Trade, Industry and Energy established "the Basic Plan of High-level Radioactive Waste Management" as a follow-up to the "Recommendations to the Government on Spent Nuclear Fuel", which was also passed by the 6th Nuclear Promotion Committee. Based on this plan, first the government will go through a scientific and democratic process of selecting a final disposal site for a term of 12 years, which will be followed by a study of the licensing procedure of a final disposal facility, and the construction of an intermediate storage facility and an underground research laboratory for 14 years. The final stage is the construction of a final disposal facility which is planned to take ten years.

The Shin-Kori unit 3, which is APR1400 with 1 400 MWe output, was connected to the grid in January 2016 and started commercial operation in December. This nuclear power plant is the reference power plant of Barakah in the United Arab Emirates.

Korea has been constructing the four units of the Barakah nuclear power plants, which the Korea Electric Power Corporation (KEPCO) successfully won in December 2009. The Barakah unit 1 is under commissioning and scheduled for operation in May 2017. The unit 2's nuclear reactor was installed and structural construction is ongoing in the unit 3 and 4. All four of them will be completed by 2020.

As of today, the construction project of the Jordan Research and Test Reactor has reached nearly 100% of completion. The performance and utilisation tests fulfilling the reactor specification are being conducted and it is expected to be handed over to the Jordan's operator, JAEC (Jordan Nuclear Energy Commission), at the end of 2016.

In the middle of July 2016, Korea welcomed 36 staff members working for the King Abdulah City for Atomic and Renewable Energy (K.A.CARE) of the Kingdom of Saudi Arabia to train and educate them on the System-integrated Modular Advanced Reactor (SMART) system design through class room training and on-the-job training.

A Specific-design Safety Analysis Report of the Prototype Generation IV Sodium-Cooled Fast Reactor (PGSFR) to a regulatory body will be submitted by 2017 to obtain its design approval by 2020. As a preparatory step, the Korea Atomic Energy Research Institute (KAERI) prepared a preliminary safety information document of the PGSFR at the end of 2015, which is going to be submitted to the regulatory body to have an independent and authorised peer review on the safety of the PGSFR.

The nuclear hydrogen key technology development project has its purpose in developing and verifying key challenging technologies necessary to realise a nuclear hydrogen system. The key technologies are the base to the GIF VHTR co-operation which includes design analysis codes, high-temperature experiment technology, high-temperature material data, tri-structural isotropic (TRISO) fuel production and hydrogen production process.

Russia

Nuclear power in Russia

At present 35 nuclear power units are in operation in Russia with total electric power capacity 27.2 GWe: 9 power units and floating nuclear power unit are under construction.

On the 5 August 2016, the stage B1 – "Nuclear Fuel Download and Subcritical Testing" has been successfully completed at Novovoronezhskaya NPP unit 6 equipped with Generation 3+ VVER-1200 nuclear power reactor. Commercial operation of unit 6 is planned to commence by the end of the year 2016 after completing the acceptance programme and test operation at 100% design power level. One of main peculiarities of the innovative unit 6 is use of additional passive safety systems in conjunction with traditional active ones. New design envisions protection from earthquakes, tsunami, hurricanes and aircraft crash impact. Designed according to the new safety standards, the reactor hall is covered with a double layer protective containment; corium "trap" is installed under the reactor vessel; a passive residual heat removal system is implemented at the unit 6.

Concern Rosenergoatom (Rosatom's national generating company) continues analysing and improving safety of all the NPPs in operation, being constructed and designed with regard to events similar to the accident with the Japanese nuclear power plant "Fukushima-1".

The State Atomic Energy Corporation "Rosatom" Strategy in the Area of Innovative Reactor Technologies

The State Atomic Energy Corporation "Rosatom" R&D is focused on the innovative reactor technologies in the following areas within the GIF Framework:

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

Unlike the last three technologies being developed on a conceptual level, the first two ones (SFR and fast reactor with heavy liquid metal coolant) are 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 programme encompasses the particular projects of reactor facilities, the R&D and mastering of industrial production technology of promising dense nitride fuel, as well as activity support of the nuclear fuel cycle closure issues and is aimed at creation of new technology platform for the future nuclear power.

At the moment preparation and adoption of FTP-2 for the period up to 2030 is under consideration.

Sodium-Cooled fast reactors (SFR)

Now there are three units with fast neutron reactors operating in Russia:

- commercial power unit BN-600 (more than 36 years of operation);
- research reactor BOR-60 (47 years of operation);
- commercial power unit BN-800 (in commercial operation since 31 October 2016).

Operation term of BN-600 of was extended to 40 years (until the end of March 2020) and work on its extension up to 60 years is underway.

BN-600 unit demonstrates stable and reliable work at the design power level. BN-600 load factor exceeded 87.45% in 2016 being the record over the whole operating lifetime of the power unit.

Lifetime of the BOR-60 reactor has been prolonged up to the end of 2019.

Pilot operational stage of power unit with BN-800 reactor was successfully completed by 15-days continuous complex test run at 100% design power level (17-31 August 2016). On 21-23 September 2016, on-site emergency response exercises took place at the Beloyarskaya NPP with initiation of beyond-design-basis accident with "full loss of power supply" for units 3 and 4 as a consequence of hypothetical earthquake. On 31 October 2016, unit 4 with BN-800 reactor was put into commercial operation after completion of acceptance test program.

The technical design of the high power fast sodium reactor BN-1200 is completed. Design of the power unit with the BN-1200 reactor facility is underway to meet the requirements of the 4th generation reactor energy systems. Beloyarskaya NPP site is under consideration as a candidate to host the first-of-a-kind power unit with BN-1200 reactor expected to be built by 2030.

The reactor MBIR which is intended to replace a research reactor BOR-60 is under construction at the site of RIAR since 2015. The MBIR unique research capabilities and capacity will be used in the framework of the International Research Center (IRC) organised by Rosatom and open for participation to all interested parties.

In 2017, upgrading of the fast critical facilities (BFS) at Obninsk Institute of Physics and Power Engineering (IPPE) will be completed. BFS complex is capable of full-scale modelling of cores of any power level to justify prospective designs of nuclear facilities.

Fast reactors with heavy liquid metal coolant (HLMC)

Primary focus within the HLMC direction is placed on R&D programme to substantiate lead-cooled BREST-OD-300 design.

Activities within the GIF

An extension of the GIF System Arrangement on SCWR is planned and ongoing. However, entering into a GIF Project Agreement on SCWR thermal-hydraulics and safety is suspended for the time being due to the revising of the item internal plans.

South Africa

Nuclear policy and energy planning

The 2008 Nuclear Energy Policy of South Africa set the scene for an energy mix and nuclear being part of the energy landscape for South Africa. In addition, South Africa's approved Integrated Resource Plan 2010-2030 stipulates the need for an additional 9.6 GWe of nuclear power by 2030. Currently nuclear capacity is 1.8 GWe from the Koeberg nuclear power station.

The Integrated Resource Plan is currently under review in November 2016, the Department of Energy gazetted a draft revised Integrated Resource Plan and Integrated Energy Plan for public consultation. Government has reiterated that nuclear power will be procured at a "scale and pace that the country can afford".

Nuclear new build programme

Following a Cabinet decision of 9 December 2015, South Africa planned to release the Request for Proposal at the end of September 2016. The complexity that surrounds nuclear procurement, internal due diligence and stakeholder consultation that had to be undertaken hindered the realisation of this deliverable and resulted in delays. However, South Africa remains committed to ensure energy security for the country, through the roll-out of the nuclear new build programme as an integral part of the energy mix. The nuclear new build programme will enable the country to create jobs, develop skills, create industries and contribute to the country's knowledge economy.

In November 2016, Cabinet designated Eskom as the Procurer, Owner and Operator of nuclear power plants with Necsa as an owner and operator of front-end fuel cycle facilities including the Multi-purpose Reactor with the US Department of Energy continuing its policy setting mandate and assuming a co-ordination role for the programme.

A Ministerial Determination under Section 34 of the 2006 Electricity Regulation Act was gazetted, in December 2016. Following this, Eskom and Necsa jointly issued an open Request for Information (RFI) to all potential suppliers for the nuclear power programme. At the closing date for expression of interest (31 January 2017), Eskom had received responses from 27 companies intending to provide a response to the RFI. The list includes major nuclear vendors from China (SNPTC), France (EDF), Russia (Rosatom Overseas) and South Korea (KEPCO). The closing date for the RFI is 28 April 2017. Thereafter Eskom plans to issue a competitive procurement process by mid-2017.

Nuclear safety and licensing

Eskom submitted two applications for nuclear installation site licence (NISL) to the National Nuclear Regulator (NNR) on 10 March 2016. The sites applied for are; Thyspunt in the Eastern Cape and Duynefontein in the Western Cape. In July 2016, the NNR completed an initial review of the NISL applications received and found both applications to be compliant with relevant national policies as well as the NNR Act and associated regulations and has accepted the applications for further processing.

Following Eskom's submission, Eskom hosted technical specialists from the National Nuclear Regulator at Thyspunt in the Eastern Cape on a site familiarisation exercise, which forms part of their process of reviewing Eskom's application for a nuclear installation site licence.

The National Nuclear Regulator is in the process of assessing the suitability and acceptability of both sites to accommodate nuclear installations in accordance with the regulatory requirements. They have been following their internal processes of verification of the application and have conducted an audit on the Eskom processes related the development of the Site Safety Report. The National Nuclear Regulator has appointed local and international consultants to assist with the evaluation of the Site Safety Report and Safety Case.

Having submitted its final Environmental Impact Report (as part of the Environmental Impact Assessment process) extensive public consultation was undertaken and Eskom awaits a record of decision from the Department of Environmental Affairs.

Plant life extension and ageing management

The Koeberg nuclear power station contributes significantly to the economy of the country and the Western Cape Province. This power station has over a period of five years contributed ZAR 53.3 billion (South African rand) to the country's economy and ZAR 29 billion to the Western Cape Province.

Eskom steam generator replacement proceeds as scheduled. The steam generator replacement project is part of the plant life extension strategy, and is intended to result in a life extension to 60 years as well as a 10% Thermal Power Uprate.

IAEA expert missions

In June 2016, in preparation for the mission the NNR hosted an IRRS preparatory meeting South Africa will be hosting the IAEA Integrated Regulatory Review Mission from 5 to 15 December 2016 to review South Africa's regulatory system, both technical and policy, including the status of development of the regulatory infrastructure against IAEA safety standards and international best practices.

On 15 December 2016, the IAEA completed an 11-day Integrated Regulatory Review Service mission on the National Nuclear Regulator and Department of Health Directorate Radiation Control with the objective of enhancing and strengthening regulatory infrastructure of among others nuclear radiation, waste management and transport safety.

Following the undertaking of the IAEA Safety Aspects of Long Term Operation (SALTO) Mission by Eskom Koeberg's nuclear power plant in November 2015 with the objective reviewing the programme and activities of the Koeberg nuclear power plant as it pertains to safe long-term operation – the utility received recommendations, suggestions and were commended on good practices as it pertains to the aspects of the mission and is has developed an implementation plan currently being rolled out to address SALTO mission recommendations.

High-temperature reactor research and development

Eskom is currently considering several options regarding a future Advanced High-Temperature Reactor (AHTR) Project. During the initial phase plans are to establish the potential of specific technologies at a research level, a programme to design and obtain regulatory permits for a "proof of concept" reactor. The other programme would be the reestablishment of the operation of the pebble fuel laboratories at Pelindaba.

Nuclear skills development

South Africa through the National Nuclear Regulator launched in September 2016 its 1st Centre of Excellence for Nuclear Safety and Security (CNSS). The centre will among others, provide i) continuous supply of personnel trained in nuclear safety to serve the needs of the nuclear regulatory body and the nuclear industry in general; ii) continuous professional development programmes in nuclear safety, undertaking of nuclear safety research to support regulatory activities and decision making; and iii) technical support services in nuclear safety to the regulatory body and the nuclear industry.

As part of nuclear skills development, among others, South Africa through the South African Nuclear Energy Corporation (Necsa) entered into an Agreement under South African Civil Nuclear Energy Training Program organised by State Nuclear Power Technology Corporation (SNPTC) of China for training in engineering design, project management, commissioning and start-up, module manufacture and construction technology. To date, phase I and II of the South African Civil Nuclear Energy Training Programme (SACNET) Training has been completed in June 2016 and we look forward to phase III of the training.

In 2016, Russia in partnership with the Department of Higher Education and Training of South Africa opened a call for applications offering ten scholarships in nuclear physics and technologies to study in Russia.

As part of human capacity development, South Africa further sends students to KINGS (KEPCO International Nuclear Graduate School) for training which includes a two-year Master's programme accredited by the Korean Ministry of Education.

Hosting of the Generation IV Forum in October 2017

South Africa has accepted the hosting of the 38th Expert Group and 44th Policy Group Meetings of Generation IV International Forum from 16-20 October 2017 in Cape Town.

Switzerland

General decision of Switzerland about nuclear power future

Shortly after the Fukushima Daiichi accidents the Swiss government decided to phase out nuclear energy, which in practice means that the currently operating five nuclear units will not be replaced after the end of their lifetime. The duration of the remaining operation time is determined by safety considerations according to the Swiss licensing regime; operation until 60 years of lifetime and beyond is therefore in principle permitted. This was confirmed by the decisions taken by the National Council, one of the two chambers of the Federal Assembly and to be confirmed by the Council of States, the second chamber. On 27 November 2016, a public referendum took place in Switzerland regarding an accelerated phase-out of nuclear energy production, which would have in practice meant the shutdown of the last Swiss NPP in 2027 already. However, the referendum was not accepted by the Swiss voters.

Operation of Swiss nuclear power plants

There are four nuclear power plants in the country with five units (two BWR and three PWR units).

One utility (BKW Energie AG) has announced that it will shut down the Mühleberg BWR-4 by 2019, after 47 years of successful nuclear operation. The corresponding decommissioning project was submitted to the authorities earlier in 2016.

Unit 1 of the Beznau NPP is still in shutdown, and will most likely not restart before spring 2017. Extensive ultrasound measurements have been taken in the base material of both pressure vessels. For unit 1, indications for small defects were noted, and the evaluation of its safety relevance is ongoing. The regulator (ENSI) has implemented a dedicated Review Panel with recognised international experts to assess the results of the safety evaluation. A mock-up part of the pressure vessel has been forged, in order to demonstrate that the ultrasound indications are in fact precipitations as the result of the forging process, and existed since the beginning of operation.

Unit 2 went online again on 23 December 2015, after a successful replacement of the upper vessel head.

In the Leibstadt power plant, indications of extended dry out periods with enhanced fuel rod oxidation were observed in the upper part of several fuel assemblies. Until further clarification, the unit is currently shut down and expected to restart in early 2017.

Generation of baseload electricity continues facing considerable economic challenges in Switzerland, in terms of declining market prices for the electrical kWh.

The search for a deep geological waste disposal is progressing well, with three different locations preliminary selected as candidate storage sites in Opalinus clay. The national association for waste disposal is in charge of implementing the disposal site in a publicly transparent manner.

Nuclear power related research in Switzerland

In spite of the decision on the phase-out, the government decided to continue the nuclear-related research and education. The key centre of nuclear excellence in Switzerland is the Nuclear Energy and Safety Division of the Paul Scherrer Institute, and the key mission of the division is to maintain nuclear competence for the foreseeable future. As of 1 January 2016, the Nuclear Energy and Safety Division now includes a newly created Laboratory for Radiochemistry. The PSI hot laboratory, one of the few European hot labs that is able to handle and analyse full-length fuel rods, has just filed for an approval for lifetime extension until 2026 and beyond.

The focus is on the safety of light water reactors (LWRs) and scientific support for deep geological waste repositories. Strong dedication to Nuclear Education (with four university professors and many senior scientists as lecturers) will help ensuring an adequate inflow of competent researchers into the nuclear field.

By signing the Memorandum of Understanding on 20 November 2015, Switzerland entered the Generation IV International Forum Molten Salt Reactor (MSR) Project, joining Euratom (represented by JRC), France (represented by CEA) and Russia (represented by Rosatom) in their collaborative efforts to develop this Generation IV system.

The involvement with Gen IV reactor concepts includes research related to:

- high-temperature materials for VHTR and GFR;
- design and safety studies of molten salt reactor (supported via Swiss National Science Foundation and Horizon2020 SAMOFAR Project);
- bilateral co-operation between CEA and PSI on ASTRID safety.

These Gen IV-related activities offer attractive opportunities for innovative research especially important for keeping young researchers in the field. At the same time, it allows Switzerland to closely monitor the international progress of reactor technology towards more sustainable nuclear energy.

United Kingdom

In late 2015, the UK government committed to invest in an ambitious nuclear research, development and innovation programme. 2016 saw the launch of this programme. It has been underpinned by the completion of new facilities capable of fuel and materials research into Gen IV technologies and the continuation of the UK government's assessment process of SMRs. This saw the completion of extensive techno-economic assessment work and the launch of a competition to identify the best value SMR design for the United Kingdom.

Innovation programme areas

The UK government's updated nuclear innovation programme supports innovation in the civil nuclear sector across five major areas. These build on recommendations by the UK Nuclear Innovation and Research Advisory Board: a body established by government to provide independent, expert advice on the research and innovation needed for nuclear energy to play a significant role in the UK's future energy mix and for the UK nuclear industry to contribute significantly to the UK economy.

The key areas of focus of this programme are:

- Leading edge work on advanced nuclear fuels that may provide greater levels of efficiency

Fuel research includes the development, manufacture and irradiation of non-oxide accident-tolerant fuels and cladding, initially intended for thermal spectrum light water reactors. The fuel development work extends beyond LWR fuels to cover research into improved manufacturing processes for coated particle fuels, such as those used in high-temperature reactors. This includes the exploration of a range of coatings and deposition and fabrication techniques for the fuel kernels. The fuels programme also encompasses fast reactor fuels, through its aim to demonstrate manufacturing and characterisation processes required to produce plutonium containing fuels for fast reactors.

This experimental work is complemented by a programme to develop and validate innovative techniques to model the physics and performance of new reactor fuel types developed through this work, as part of their validation prior to reactor testing.

- Research into fuel recycling processes to reduce future environmental and financial burdens

The aim of this work is a five-year programme to demonstrate radical improvements in economics, proliferation resistance, waste generation and the environmental impact of nuclear fuel recycle technologies. The programme has an initial focus in its first year of developing the basic processes required for an aqueous recycle process for LWR UOx and thermal MOx fuels that improves on the above areas, relative to the current Plutonium Uranium Redox Extraction (PUREX) process. In subsequent years, the aim will be to take forward work in a similar manner on fast reactor recycle processes, including pyroprocessing techniques.

- Developing materials, advanced manufacturing and modular build for the reactors of the future

This is an integrated programme of R&D on advanced materials and manufacturing, encompassing the development of new nuclear materials, the mechanisation and automation of nuclear component manufacture at different scales, pre-fabricated module development and verification and development of appropriate nuclear design codes and standards. It involves laboratory scale research to develop materials performance data and gain a fundamental understanding of materials and manufacturing processes suitable for use in the development of Gen IV reactors, as well as the modularisation and more effective manufacture of reactors in general.

- Research that underpins the development, safety and efficiency of the next generation of nuclear reactor designs

Reactor design work focuses on increasing the widespread uptake of modern digital engineering practices and simulation tools to improve predictive modelling capability and the understanding of passive safety arguments in new reactor designs. The aim is to lead to enhanced designs, increased productivity and a step change in the way that nuclear design, development and construction programmes are implemented. This platform is intended for establishing collaborative design projects with partners, with areas of focus being on Generation IV designs and on increased modularity and off-site manufacture for current and future reactors.

This is complemented with the development of improved reactor design methodologies for security and safeguards. The intention is to improve understanding of the safety aspects of through-life performance of reactor components, to enhance security modelling and simulation assessment methodologies and to develop advanced regulatory safety case methodologies for current and future reactor systems.

This work is complemented by the development of a suite of toolkits and underpinning data that will enhance the UK government's knowledge basis for future decision making in the nuclear sector up to 2050.

New nuclear research facilities

2016 also saw the opening of new suites of facilities intended to advance research, development and innovation into current and future generations of reactors. These have been developed with the help of grants from the UK government and include:

- The High Temperature Facility

The High Temperature Facility (HTF) Alliance has built an open access materials testing laboratory for research organisations looking to investigate, develop and advance structural materials technology for future systems applications. Generation IV nuclear fission, nuclear fusion, advanced gas turbine materials and other advanced energy concepts fall within its scope. The HTF Alliance consists of a team of UK companies and universities with in-depth knowledge of advanced nuclear fission systems design, manufacture, operation and the regulation needed to align R&D programmes to establish the innovative experimental rigs required to address priority research challenges.

The HTF offers rigs capable of testing materials at temperatures up to 1 000°C and with temperature cycling in a range of novel, demanding environments (pressurised gas for VHTR/HTR, liquid metal for SFR/LFR, inert atmospheres). The HTF will also enable new predictive models to be developed, and new data to be generated that will underpin the selection, manufacture and performance of advanced materials for future generation technologies.

- The UTGARD Lab

The U/Th/beta-Gamma Active process chemistry R&D (UTGARD) lab at the University of Lancaster is now operational for work on beta- and gamma-active fission products, uranium, thorium and low-level alpha tracers. This is oriented towards development of safe, economic, efficient and proliferation-resistant aqueous fuel recycling technology. Key areas of work focus on uranium- and thorium-based fuel cycles, hydrometallurgical processing and the interface with pyrochemical reprocessing routes.

- Pyroprocessing

The UK's new pyroprocessing research laboratory, opened during 2016 and based at the University of Edinburgh, enhances the UK's capability by providing the equipment and infrastructure required to demonstrate the essential components of a fast reactor fuel pyroprocessing recycle technology, with a view to allowing subsequent hot cell testing of these processes.

Small modular reactor assessment and competition

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. SMRs are in the early stages of development and there are no commercially operational examples that can be used to validate their potential.

Following a number of preliminary studies, the UK government launched a techno-economic assessment of SMRs, which was completed during 2016. This covered the following areas:

- Comprehensive analysis and assessment, involving in-depth data collection and evidence based analysis of a very broad base of SMR technologies.
- Systems optimisation modelling, covering strategic analysis of SMRs and other energy sources and their impact on a balanced UK energy system.
- A strategic analysis of emerging nuclear technologies, and their applicability in the UK context.
- Consideration of how SMRs fare in the context of the UK's safety and security regime.
- An analysis of various advanced manufacturing techniques apply to SMR technologies.
- Analysis of how modularisation on containment, shielding and structural elements, safety systems and assembly process contributes to the deployment of small reactors. This also assessed the impact on maintenance, tolerances and construction risk.
- An analysis of the impact of extensibility of control systems on multiple systems, safety and fail-safe systems, improved load following capability.

The study encompassed both LWR based SMRs, as well as metal cooled fast reactor, molten salt reactor and gas-cooled high-temperature reactor designs.

In March 2016, the government launched the first phase of a competition to gauge market interest among technology developers, utilities, potential investors and funders in developing, commercialising and financing SMRs in the United Kingdom. The government is keen to ensure that any subsequent stages of the competition are informed by participants' views on how to secure commercial deployment of SMRs and on potential time frames for deployment of different families of reactor technologies, including those based on Generation IV technologies. This phase of the competition has consisted of a structured dialogue between government and participants, with the outcome of this is expected in 2017.

United States

Nuclear energy continues to be a vital part of the United States' energy strategy for a secure, sustainable, clean energy future. A number of initiatives (such as Gateway for Accelerated Innovation in Nuclear) proposed under the previous and current Administration are intended to properly recognise the value of reliable, emission-free nuclear energy in the electricity market and to encourage the development of advanced reactor designs.

With the growing emphasis on nuclear energy, the Office of Nuclear Energy recently completed an internal reorganisation to improve staff alignment with programmatic responsibilities. This includes the establishment of the new Office for Spent Fuel and Waste Disposition to focus on spent fuel transportation and storage and consent-based siting of storage locations and the establishment of the Office of Nuclear Technology Demonstration and Deployment to sharpen our focus on commercialisation. The Office of Advanced Reactor Technologies, now reporting to the new Office of Nuclear Technology Research and Development, will continue to perform research to develop technologies and subsystems that are critical for advanced concepts, with an emphasis on fast reactors, high-temperature reactors and generic advanced reactor technologies.

In the area of light water reactors (LWRs), 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 completed loading fuel last December, began power ascension testing during the summer, and declared full commercial status on 19 October 2016. Watts Bar 2 is the first US reactor to be completed since Watts Bar unit 1 began operating in 1996. Collectively, these five reactors will have a combined generation capacity of 5 500 MWe, enough to power approximately 3.25 million homes. The NRC is currently reviewing Dominion Resources' combined licence (COL) application for a GE economic simplified boiling water reactor (ESBWR) at the North Anna unit 3 site in Virginia. Development and licensing of the ESBWR and the AP1000 designs was supported through cost-share arrangements with the Department of Energy's (DOE) Nuclear Power 2010 programme.

The DOE LWR Sustainability (LWRS) programme is conducting research and development (R&D) on advanced technologies that improve reliability, sustain safety and extend life of the current LWR fleet. The LWRS programme is also helping the industry address current economic challenges by introducing new technologies through its Pilot Plant programmes to help gain efficiencies and improve safety. Both Dominion Resources and Exelon have announced their intention to seek an extension of the operating licence of the Surry plant in Virginia and the Peach Bottom plant in Pennsylvania for another 20 years, which would mean a total of up to 80 years of operating for these reactors. A final decision on approving these subsequent licence renewals (SLRs) would likely be made by the early part of the next decade. If granted, these initial SLRs would create a precedent for other operators. Although a number of plants are under economic pressure to close due to low natural gas prices, state governments and regional electricity markets are considering changes to properly value nuclear power's contributions to clean energy production and grid stability. For example, starting in April 2017, New York State's Clean Energy Standard will require all six New York investor-owned utilities and other energy suppliers to pay for the intrinsic value of carbon-free emissions from nuclear power plants by purchasing Zero-Emission Credits, which are estimated to be worth USD 965 million in the first two years. This recent action by the New York State government prevented the premature shutdown of the Fitzpatrick plant.

The DOE stands firmly behind SMRs as an emerging technology that can meet the nation's growing energy demands – including possibly 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 programme 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, for which a preferred site has been identified at the Idaho National Laboratory. A UAMPS COL application for this project is planned for submittal in 2017 with commercial operation set for the mid-2020s, pending a decision to proceed in December 2016. Tennessee Valley Authority submitted to the NRC in May 2016 a technology-neutral early site permit (ESP) application for their Clinch River site in Tennessee. The ESP application references a plant parameter envelope that encompasses characteristics of all US light water-based SMR designs.

In the area of advanced reactor technologies, a number of important actions have recently been completed:

- First, the congressionally directed study to evaluate options for a new advanced test or demonstration reactor has been completed. The study evaluated a range of options towards deploying advanced irradiation and technology demonstration reactors. The final report, available at www.inl.gov/article/inl-partners-with-fellow-national-labs-to-evaluate-technologies-for-the-next-test-and-demonstration-reactor, was prepared by a team from the national laboratories. This report received favourable reviews from the Nuclear Energy Advisory Committee (NEAC). Acting Assistant Secretary for Nuclear Energy John Kotek has further tasked NEAC to explore the needs, capabilities and options for an irradiation test reactor from the long-term perspective, that is 2030 and beyond.

- Second, on 6 June 2016, DOE published the Draft Vision and Strategy for the Development and Deployment of Advance Reactors and is available online at http://energy.gov/ne/downloads/draft-vision-and-strategy-development-and-deployment-advanced-reactors. This document highlights the need for the development and deployment of advanced reactors in anticipation of the need for almost 200 GWe of new capacity due to retirements, growth and replacement of carbon-based electrical generation. The vision reflects the role that advanced reactors will play in the 2050 time frame and highlights as its goal to bring at least two non-light water advanced reactor concepts to a point sufficient to allow construction to go forward, including completion of necessary licensing reviews by the NRC. The strategy necessary to achieve this vision and goals identifies six strategic areas for which DOE will pursue long-term actions in support of the development and deployment of advanced reactors. These include improving access to our national laboratory infrastructure and expertise by vendors, demonstrating performance and retiring technical risk, developing fuel cycle pathways, supporting the establishment of an efficient and predictable regulatory framework, working with the private sector to effectively leverage resources to accelerate deployment, and providing for human capital and workforce development.

- Third, on 22 September 2016, the Secretary of Energy Advisory Board's (SEAB) Task Force on the Future of Nuclear Power presented the findings of its draft report, which are available at www.energy.gov/seab/downloads/draft-report-task-force-future-nuclear-power. The report discusses the current status of nuclear energy development and recommends that the United States undertake an advanced nuclear reactor programme to support the design, development, demonstration, licensing and construction of a first-of-a-kind commercial scale reactor through a four phase approach that would require 25 years and USD 11.5 Billion, including private sector cost share.

- Fourth, DOE is continuing efforts, begun in 2012, to seek interactions with industry for the development of its R&D programme. DOE has finalised the two awards with X-energy and Southern Company previously announced on 15 January 2016. These cost-shared awards will support the further development of advanced reactor concepts. Currently USD 24 million is available to support these two efforts, including USD 10 million to conduct R&D DOE national laboratories. X-energy is pursuing a high-temperature gas reactor, and Southern Company LLC is pursuing a molten chloride salt fast reactor.

- Lastly, in regards to licensing efforts, DOE drafted 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. As a step in developing the guidance the NRC has provided a draft set of advanced reactor design criteria adapted from the General Design Criteria for industry comment in conjunction with the DOE-NRC joint initiative in this area. It is accessible through the NRC's website and The Agency-wide Documents Access and Management System (ADAMS) at www.nrc.gov/docs/ML1609/ML16096A420.pdf. The NRC held a public meeting in October 2016 and will hold meetings with their advisory committee in early 2017. The NRC anticipates publishing the final guidance in late 2017. Also in regards to licensing, the NRC and DOE hosted two-day workshops in September 2015 and June 2016 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 first workshop focused on regulatory needs and the second reported on recent initiatives and examined qualification of advanced reactor fuel. The workshops included participants from government, industry, national laboratories and nuclear-related organisations. The next workshop with the NRC is scheduled for 25-26 April 2017.

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 working to train the next generation of nuclear engineers and scientists by sponsoring research and student educational opportunities at US universities. In FY 2016, DOE made 90 awards totalling USD 66.6 million for nuclear energy research and infrastructure enhancements. DOE anticipates making roughly 80 awards in FY 2017 – valued at approximately USD 77 million – for university- and national laboratory-led nuclear R&D projects and infrastructure grants. For the FY 2017 awards, 720 R&D pre-applications were received. Invitations to submit full applications are expected in mid-December, with final award notifications anticipated for June 2017. Additionally, DOE offers scholarships and fellowships to encourage careers and research in nuclear energy-related fields to meet expected future workforce needs. In FY 2016, DOE awarded USD 5.2 million for 56 undergraduate scholarships and 33 graduate fellowships to students at universities across the United States and plans to issue awards again in FY 2017. DOE has awarded more than USD 33 million in scholarships and fellowships since its programme began in 2009.

In July 2016, DOE completed hosting eight public meetings around the country on the Department's consent-based siting initiative for facilities supporting an integrated waste management system needed to manage our nation's nuclear waste. DOE collected over 10 000 email inputs and hundreds of additional comments from the public, communities, states, Tribal governments, and others on what matters to them as the Department moves forward in developing a consent-based process for siting facilities to store, transport and dispose of spent nuclear fuel and high-level radioactive waste. A draft summary report that reflects these inputs was released for public comment in September and will be finalised in December. A draft consent-based siting process will be released in December 2016.

As DOE strives to meet the challenges of energy security in environmentally benign ways – the United States will rely heavily upon nuclear energy as a key element in the United States' energy portfolio.

Chapter 3. System reports

This chapter gives a detailed overview of the achievements made in 2016 in the R&D activities carried out under the four System Arrangements (VHTR, SFR, SCWR, GFR) and under the two MOUs (LFR and MSR).

3.1. 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 while 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 and simplified coolant handling. 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 demonstration 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-cladded 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 the 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 and 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

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;
- development of high power blowing machines.

Experience feedback and current research relating to the 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.

Main activities and outcomes of ALLEGRO

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 by ÚJV Řež, 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 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 station blackout and LOCA) 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 station blackout in passive mode.
- Solution of potential unprotected transients.
- Development of severe accident mitigation measures in the ALLEGRO design.

Thermal-hydraulic benchmark activities

The purpose of the thermal-hydraulic benchmarking activities carried out in the framework of the VINCO Project (Horizon 2020) is to share the knowledge and mutual learning between participating Central European laboratories joined to the V4G4 Centre of Excellence and the associated CEA (France) institute taking part in the ALLEGRO demonstrator development.

The ongoing benchmark activities are focused on the case analyses related to the key safety issues and better understanding of gas-cooled ALLEGRO demonstrator performance and capabilities. The various thermal-hydraulic computational tools (RELAP5-3D, CATHARE2 and MELCOR) are utilised by different users. The aim is to assess the capabilities and limitations of current system codes to reproduce dynamics of gas-cooled system correctly as well as to support the development of consistent ALLEGRO models in different organisations.

The organisations which participate on the ALLEGRO models development and benchmark are MTA-EK (Hungary), National Centre of Nuclear Research (Poland), ÚJV Řež (Czech Republic) and co-ordinated by VUJE (Slovak Republic).

The thermal-hydraulic benchmark activities are carried out in the following steps:

- Summarising the data necessary to create new and improve existing ALLEGRO TH models. In the past the huge effort was spent on the activities to design and evaluate safety aspects of the 75 MW ALLEGRO demonstrator and also other specific gas-cooled applications. The deliverable documents and published papers from EU FP7 Project GoFastR, its predecessors (e.g. EU FP4 Project RAPHAEL) and other projects oriented on Gen IV reactors are serving as the information and data source to create the database.
- Development of new ALLEGRO thermal-hydraulic models. The two brand new ALLEGRO TH models are being prepared in the VUJE (Slovak Republic) and ÚJV Řež (Czech Republic) institutes using RELAP5-3D and MELCOR codes.
- The qualification of the utilised models on the steady state level in order to ensure the consistency of the initial and boundary conditions prior the transient calculations. Identification and definition of the model distortions according to the findings during steady state qualification process.
- Execution of transients defined in the benchmark specification. For the purpose of the benchmark the 3 inch LOCA on the cold duct no. 1 and the total station blackout using decay heat removal (DHR) loop no. 1 for the core residual heat removal in natural circulation regime were selected. The reason for selection of these transients was to cover the challenging situations of the gas-cooled system during both depressurised and pressurised conditions and to evaluate response of the main safety systems.

Neutronic benchmarks activities

Two neutronic benchmarks were launched in the VINCO Project (Horizon 2020) based on experience from the whole core reactor physics benchmark at ESNII+ EU FP7 Project.

Evolution of ALLEGRO core is driven by two factors – problems with DHR proportional to power density and by better availability of uranium dioxide (UOX) fuel for first cycles (in comparison with MOX). First round of calculations oriented on UOX fuel feasibility including resulting direction of core modifications is characterised.

First VINCO Neutronic Methodological Benchmark

The goal of the benchmark is to verify physical effects of the core neutronic modelling as follows:

- nuclear data uncertainties;
- satisfactorily detailed energy discretisation;
- resonance self-shielding in the energy region of the resolved resonances;
- resonance self-shielding in the unresolved region by using statistical approach;
- representing anisotropy of the scattering in the leakage calculation;
- representing anisotropy of the flux in the leakage calculation.

The main features of the benchmark are as follows:

- based on 2D calculation of MOX and UOX pin at radially infinite net;
- MOX and UOX fuel.

Requested results:

- multiplication factors;
- critical buckling;
- concentrations of key actinides;
- reactivity effect of fuel temperature (Doppler);
- reactivity effect of pellet radial expansion;
- reactivity effect of He dilution (void effect).

Participants: ÚJV Řež with ECCO, VUJE with HELIOS and SERPENT, National Centre of Nuclear Research with SCALE 6.2 and SCALE 6.1.3, MTA-EK with ECCO.

Although the deviations of the results for the MOX fuel are smaller than in case of earlier 3D calculation exercises, they remain considerable as a result of the impact of the following modelling characteristics:

- nuclear data uncertainties;
- resonance self-shielding in the energy region of the resolved and unresolved resonances.

The not zero buckling prescription of the benchmark could show the impact of the leakage models not only for the k-eff but also for the Doppler coefficient and the pellet expansion coefficient. The probable reason of these deviations can be the modelling differences of the anisotropy of the scattering in the leakage calculation and the anisotropy of the flux, which are leading to different migration areas and must be important also in the 3D core calculations. The problem can also be demonstrated by the fact that the differences of the k-eff values are much larger than those for the kinf.

For some isotopes, the impact of the leakage on the spectrum leads to different number densities during the burnup process.

VINCO Neutronic Assembly Oriented Benchmark

The benchmark is defined as a broadening of the methodological benchmark with the same physical effects for verification and the same participants. The physical effects for verification and potential participants are the same as in the previous benchmark. Differences in comparison with methodological benchmark are as follows:

- 2D numerical models of ALLEGRO fuel assembly;
- infinite lattice without fixed buckling;
- more detailed comparison of deterministic and Monte Carlo (MC) calculations;
- approximation of realistic temperature distribution;
- results based on infinite multiplication factor.

Regardless of relatively good kinf agreement, significant differences at burnup process (surprising differences even at U-235 and Pu-239 concentrations, high discrepancies for Cm-242), reactivity effects and kinetic parameters indicate influence of nuclear data libraries (and its uncertainties) and methods used. Another effort is needed to eliminate discrepancies caused by user effect.

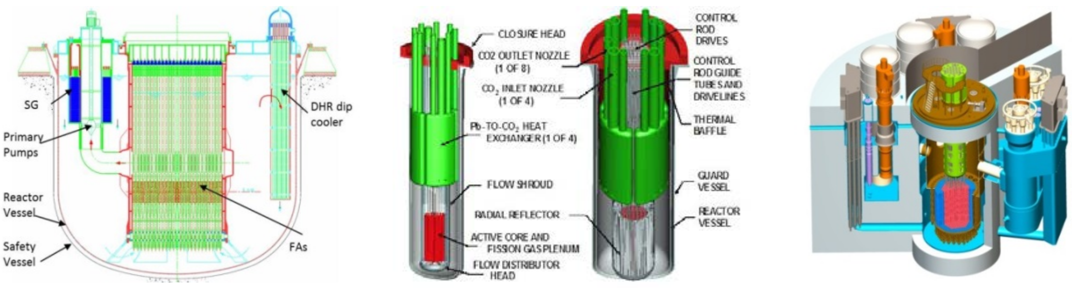
3.2. Lead-cooled fast reactor (LFR)

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 LFR concepts identified by GIF include three reference systems. The options considered are a large system rated at 600 MWe (ELFR, EU), intended for central station power generation, a 300 MWe system of intermediate size (BREST-300, Russia), and a small transportable system of 10-100 MWe size (Small, Secure Transportable Autonomous Reactor [SSTAR], United States) that features a very long core life, Figure 3.1 The expected secondary cycle efficiency of each of the LFR reference systems is at or 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 reference 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 3.1: **Sketches of GIF-LFR Reference Systems: ELFR, BREST and SSTAR**



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

Table 3.1: **Key design parameters of the 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 eutectic 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-erosion 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 2030; and higher-performance reactors by 2040. Note however that in one case (i.e. the BREST-300 demonstration/prototype reactor), licensing is currently underway, and operation is expected as early as 2022. Following the reformulation of GIF-LFR-pSSC in 2012, the SRP was completely revised, and a final draft was prepared by the SSC and sent to the GIF Expert Group for review. Comments to the SRP were received from one industrial partner (Westinghouse Electric Corporation, WEC, United States) through the VHTR SSC. The comments were the object of a detailed discussion during the Moscow GIF-LFR-pSSC meeting on 30 September 2016 thanks to the participation of a WEC representative at the meeting. The SRP is presently under review and is expected to be issued by the beginning of 2017.

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 reference system thrusts. The plan proposes co-ordinated research along parallel paths leading to one or more pilot facilities that can serve the research and demonstration needs of the reference concepts while reducing the unnecessary expense of separate major facilities and research efforts for each reference system.

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

Main activities and outcomes

Two meetings of the GIF-LFR-pSSC took place during 2016. The first was held on 16-17 February at the OECD Conference Centre in Paris, and the second was conducted in Moscow in conjunction with NIKIET 2016 conference (Fourth International Scientific and Technical Conference “Innovative Designs and Technologies of Nuclear Power”) on 27-30 September.

This second meeting was characterised by the presentations of the status of activities in MOU signatories and observer countries; these presentations were made to the public as part of the conference programme. A closed session was also held on 30 September to discuss other internal business of the pSSC.

The activities of the LFR-pSSC during 2016 centred on top level reports for GIF. After the issue of the LFR White Paper on Safety in collaboration with GIF RSWG in 2014, the pSSC was very active on the following main lines:

- LFR safety design criteria (SDC): Development of the LFR 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 of the features identified for the SFR to be adopted for the LFR (note that IAEA SSR2/1 refers substantially to LWR technology). At the end of 2016, the LFR-pSSC received comments on its draft SDC from French GIF members and from the EURATOM ARCADIA Project partners. The LFR SDC is presently under review to address these comments and suggestions for improvements. A revised version will be completed in early 2017.
- LFR system safety assessment: In 2014, the RSWG asked SSC 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 LFR assessment report was prepared by the LFR-pSSC and sent to the RSWG for comments at the end of September 2015. The RSWG provided comments in November and the final pSSC version was sent to the RSWG in early 2016.
- LFR safety design guidelines (SDG): The LFR-pSSC received from the RSWG the SFR safety design guidelines on Safety Approach and Design Conditions in October 2016. This is being used as a basis for the development of the corresponding LFR-SDG report. A draft of this report will be prepared during 2017.

LFR-pSSC comments to the IRSN report on the safety of Generation IV reactors: In June 2015, the pSSC took the initiative to analyse in detail the IRSN report on the safety of Generation IV reactors and provide comments. The committee sincerely appreciated the technically comprehensive review of LFR safety aspects provided by IRSN. However, the committee also felt that the results of recently concluded as well as ongoing R&D efforts were possibly not considered by the IRSN when drawing some of their conclusions. The comments provided by the pSSC are expected to form the basis for further discussions and possible update of the IRSN report in the future once the parts developed by other SSCs become available.

Update of the GIF-LFR website: The pSSC realised that the information given on the public section of the OECD website required updating to reflect the latest developments. The text for the updated website was developed and transmitted to OECD TS at the end of 2016.

Euratom-Rosatom co-operation agreement

Following the signature in May 2014 of a co-operation agreement between the BREST and LEADER projects, by NIKIET (on behalf of Rosatom) and Ansaldo (on behalf of the LEADER consortium), a first co-operation meeting was organised in Genova on 9-11 December 2015. A second meeting took place in Moscow on 3-4 October 2016. During the meetings, presentations were made covering both the BREST and ALFRED designs and safety features as well as many specific aspects related to thermal-hydraulics, fuel design, cooling etc. The meeting was concluded with a general positive assessment of the information exchange.

Main activities in Russia

BREST-OD-300, an innovative inherent-safe fast reactor, is being developed as a pilot and demonstration prototype for the base commercial reactor facilities of future nuclear power operating with a closed nuclear cycle.

The inherent properties of lead as a coolant:

- in combination with (U-Pu)N fuel, allow for complete breeding of fissile materials in the reactor core, maintaining a constant small reactivity margin preventing the disastrous effects of an uncontrolled power increase due to equipment failures or personnel errors;
- make it possible to avoid the void reactivity effect due to a high boiling point and the high density of lead;
- prevent coolant losses from the circuit in the event of vessel damage because of the high melting/solidification point of the coolant and the use of an integral layout of the reactor;
- provide for high heat capacity of the coolant circuit which decreases a possibility of fuel damage;
- allow for utilisation of the high density of lead and its albedo properties for flattening the FA power distribution and the fuel pin temperatures respectively, as well as in the safety systems;
- facilitate larger time lags of the transient processes in the circuit, which makes it possible to lower the requirements for the safety systems' rates of response.

One of the BREST-OD-300 development objectives is the practical justification of the main design approaches applied to the reactor facility with the lead coolant based on the closed nuclear fuel cycle, and of the foundations of the inherent safety ensuring concept, on which these approaches are based. For this reason, special attention is paid to confirmation of serviceability of the reactor core and its components.

Mixed uranium-plutonium nitride is used to ensure complete breeding of fuel in the core and a constant small reactivity margin preventing any prompt neutron excursion during reactor operation. A low-swelling ferrite-martensitic steel is used as the fuel cladding.

To confirm fuel serviceability, radiation tests of fuel elements are being conducted in the BN-600 power reactor and in the BOR-60 research reactor. At the present time, eight FAs with nitride fuel elements are being irradiated in the BN-600 reactor, and the fuel elements of a previously withdrawn FA are being subjected to post-irradiation studies. Seven FAs with nitride fuel elements are being irradiated in the BOR-60 research reactor.

In the design of the reactor core items, novelty was coupled with reference solutions. The FA has a shroudless hexagonal design. Such a solution eliminates the possibility of fuel melting due to FA flow area blockage; even in the event that the flow area at the inlet of a 7-FA group is blocked, the safe operation limits in terms of the fuel cladding temperature are not exceeded. Another positive point is a 30% reduction in the metal content of the shroudless FA as compared to the shrouded option. Technologically, the adopted design is based on the experience gained when fabricating the FAs for the VVER reactors.

To justify the FA design serviceability, full-scale mock-ups (Figure 3.2) were manufactured and subjected to mechanical, hydraulic and vibration tests in air and water environments. The mechanical tests included transverse bending, torsion, axial tension and compression. The vibration tests were conducted using running and stationary water. Also, the vibration tests were performed in air.

Figure 3.2: **Full-scale FA mock-up**

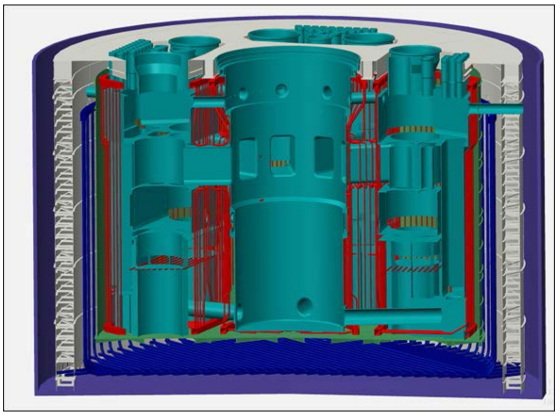
In the reactor core composed of the shroudless FAs, important in terms of the fuel element temperature determination is the knowledge of local flow rates within hydraulic cells. To determine the inter-cell and inter-cassette mixing coefficients, specific experiments in liquid metal and air were carried out.

A mock-up 37-rod fuel bundle was used in the liquid metal experiments to refine the heat transfer coefficients. Thus, a large quantity of data was obtained, used for validation of the codes intended for thermal-hydraulic calculations of the reactor core. To confirm the corrosion resistance of the FA elements in the lead coolant, tests using small-scale fuel-free mock-ups of the FAs at different temperatures were conducted.

Absence of data on the physical experiments with nitride fuel led to the necessity of carrying out an experiment using the BFS critical facility. In the simulation lead, plutonium and uranium nitride were used. Based on the results of the new experiments and the data obtained from the previous critical experiments, the calculation codes were validated for neutronic calculations. The results of the calculations carried out using the validated software tools show the possibility to achieve a small reactivity margin during the reactor operation and provision of a practically stable power density field during the duration of the fuel lifetime.

An integral layout is used in the reactor facility to avoid coolant losses. The reactor vessel material is multilayer metal concrete; the lead coolant and the main components of the primary circuit are located in the reactor vessel (Figure 3.3).

Figure 3.3: **Vessel of BREST reactor facility**



A wide range of calculations and experimental studies were required to confirm the serviceability of such a vessel type, which is novel for the nuclear power industry. The experimental justification is based on investigations and testing of the small- and full-scale components. Using the developed full-scale mock-up of the vessel bottom a capability to ensure the required temperature of the building structures has been demonstrated, and joint thermal movements of the components have been determined. Using the developed full-scale mock-up of the central part of the vessel (Figure 3.4), heating-up modes have been optimised, and the gas emission parameters have been determined.

Figure 3.4: **Full-scale mock-up of reactor vessel's central part**

To carry out strength calculations, it was necessary to obtain the properties of the constituent materials, which required performance of several experimental works. The properties of concrete have been determined for the anticipated operating temperatures and irradiation levels as well. With respect to metals, corrosion resistance experiments in the lead coolant environment have been conducted. To verify safety, the region of lead-concrete chemical interaction needs to be determined. The depth of lead penetration was experimentally determined to be no more than 0.5 mm without chemical interaction. The structural analysis of the vessel was performed using newly developed techniques. The analysis took into account the actual geometrical and physical-mechanical properties of the vessel components and complex three-dimensional contact interaction between them, the non-linear concrete properties and the formation of cracks in it. The analytical justification showed that the adopted vessel design ensures the probability of formation of a leak with partial coolant loss of no more than 9.7×10-10 1/year.

The integral layout with a steam generator (SG) located in the reactor unit vessel imposes a high responsibility on the developers, designers and experimentalists involved in the confirmation of serviceability and safety of the SG. Therefore, a thorough justification of the steam generator components and the processes taking place in the steam generator has been planned and is being carried out.

In the course of the SG experimental justification several mock-ups had been developed, which were used to verify (check) the parameters, which were identified in the detailed design. To determine the thermal-hydraulic characteristics including the impact of centrifugal acceleration on the thermal-hydraulic stability, an 18-tube model was developed. From the results of the 18-tube model tests, the heat transfer coefficients and hydraulic characteristics in the steam-water and lead circuits were obtained, as well as the temperature distribution in the lead circuit. Thermal-hydraulic stability was demonstrated in the investigated ranges.

To determine the steam generator life, thermal cyclic strength tests of the unit for securing the tubes between the tube sheets were carried out. The degree of reliability of the "tube-tube sheet" joints was determined for superheated steam removal and feed-water supply chambers in the SG modules, and the fulfilment of the thermal cyclic strength conditions were confirmed for the heat exchange tubes and the points where they are welded to the tube sheet. Tribological tests of the "tube-spacer grid" contact points in the lead coolant environment were performed. As a result, experimental data was obtained on the wear of the friction-coupled components of the specimens in the characteristic range of stresses and movements within the contact areas. A complex three-dimensional analytic justification of the steam generator serviceability was carried out, which included thermal-hydraulic calculations, strength calculations for all operating conditions, vibration strength calculations, seismic effect, aircraft crash and air shock wave calculations, and other design analyses. To verify the vibration calculations, a mock-up of the steam generator with actual geometric parameters is being developed.

Because of a high specific weight of lead, it was necessary to analyse the possibility of a secondary failure of the steam generator tubes if one of the tubes breaks. The dependent failure and the subsequent ingress of steam into the coolant may in turn affect the circulation in the circuit and consequently impair the thermal condition of the fuel elements. Based on a series of conducted experiments (Figure 3.5), it was demonstrated that it is impossible for a single SG tube rupture to develop into a multiple tube rupture (dependent rupture exclusion).

Figure 3.5: **Tube rupture experiment**

The reactor main coolant pump is intended to establish the lead coolant head and provide for its circulation in the circuit. To confirm its serviceability, several mock-ups of the pump set have been developed, as well as the test sections to check their performance:

- a medium-scale test section operating with liquid lead and a main coolant pump mock-up have been developed;
- the flow characteristics of the lead coolant flow path have been obtained in the order of 80% of the required ones (test bench limitations);
- the serviceability of a hydrostatic bearing unit has been demonstrated in the conditions of the medium-scale test bench (over 300 start-up-shutdown sequences);
- the energy performance of the flow path in water has been optimised; the required flow, head and positive suction head have been obtained.

In the future, a test bench base will be set up for the tests of the full-scale prototype of the reactor coolant pump, including endurance tests.

Other main and ancillary components are being justified at small- and medium-scale test benches; the properties of structural materials in the operating temperature ranges and rated operating conditions, including irradiation, are being obtained. The main (largest) components developed for the BREST reactor facility have been justified through the experiments and calculations and are now being prepared for prototype testing.

Another critically important direction of safety justification is the acquisition of data on radionuclide transport in the reactor facility. To investigate the processes of radioactivity transport in the liquid-metal phase and the radionuclide exchange between the liquid-metal and gaseous phases, the following components were developed: an ex-vessel loop facility with lead and gas coolants, a reactor loop facility with gas coolant, a reactor loop facility with lead and gas coolants. Transport of coolant activation products (lead impurities) ^{110m}Ag, ^{123m}Te, ^{124}Sb, ^{210}Po, ^{65}Zn and ^{210}Hg, as well as fission products (^{131}I, ^{137}Cs) and inert radioactive gases was investigated. The experimental results made it possible to perform validated calculations of the reactor facility's irradiation characteristics.

It has been shown based on the calculation results that the probability of reactor core damage (without core melting) does not exceed 8.65·10-9 1/year, which ensures the acceptable level of safety when reactor facilities of such type are used for the power industry development. The detailed design of the BREST-OD-300 reactor facility has been justified using small- and medium-scale test benches and test sections, as well as validated software tools, and the design has met the key parameters specified and the licensing procedure has been started.

Main activities in Japan

Theoretical studies of fast reactors using lead-bismuth eutectic as a coolant have been performed in Japan since the beginning of LFR activities. One of the advantages of lead or lead-bismuth coolant is that it is possible to maintain better neutron economy in the core due to the hard neutron spectrum and the small neutron leakage. These features make it easy to realise the once-through fuel cycle fast reactor concept. The concepts of the Breed and Burn reactor and the CANDLE burning reactor were studied mainly at the Tokyo Institute of Technology. One of the important issues in the CANDLE burning reactor concept is to maintain the integrity of the fuel elements in very high burnup conditions. The research shows the possibility to solve the problem by the introduction of a melt and refining process based on metallic fuel. The study also considered the use of plutonium from LWR spent fuel for the start-up core in a CANDLE reactor to achieve effective utilisation of the plutonium.

The progress of the studies was reported at the American Nuclear Society 2016 Winter Meeting, ISTC NIKIET-2016, and the 2016 Annual meeting and the Autumn Meeting of the Atomic Energy Society of Japan. The progress was also published in the technical Journal Annals of Nuclear Energy.

Experimental studies on the mass transfer of metal and non-metal impurities in lead-bismuth coolant system have been performed. The diffusion behaviours of metal impurities such as Fe and Ni in lead-bismuth were investigated by means of long capillary experiment and molecular dynamic simulation. The diffusion coefficients of these elements were newly obtained for various temperatures. The design of solid electrolyte type oxygen sensor was improved to have better response in high-temperature lead-bismuth coolant system. The chemical behaviours of air in the coolant system in the situation of air ingress accident were studied. The oxidation characteristics of the coolant were metallurgically investigated.

Main activities in Korea

Seoul National University joined the GIF-LFR pSSC by signing the MOU at the NEA in November 2015. LFR R&D progress has been made mainly within university programmes during the past 20 years, since the first study in 1996 at 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 Korean first LFR-based burner Proliferation-resistant Environment-friendly Accident-tolerant Continual-energy Economical Reactor (PEACER) has been developed to transmute long-lived wastes in spent nuclear fuel into short-lived low-intermediate level wastes, since 1996. In 2008, the Korean Ministry of Science and Technology selected the SFR as the technology for long-lived waste transmutation. Since then, LFR R&D for transmutation in Korea has turned its direction towards an ADS-driven Th-based transmutation system designated as Thorium Optimized Radioisotope Incineration Arena (TORIA) with the leadership of the Nuclear Transmutation Energy Research Centre (NUTRECK) of Korea at Seoul National University.

For the second goal Korea has also started to develop 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 Ubiquitous, Rugged, Accident-forgiving, Non-proliferating, and Ultra-lasting Sustainer (URANUS).

PEACER (Proliferation-resistant Environment-friendly Accident-tolerant Continual-energy Economical Reactor)

PEACER is a Pb-Bi cooled fast reactor being developed at the 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 a square lattice cooled by forced circulation by a main coolant pump, and using the Rankine cycle for power generation. As with other Pb-Bi cooled fast reactor concepts, the operating coolant temperature spans over 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, burnup compensation and reactor shutdown. PEACER includes in-house pyroprocessing units for spent nuclear fuel recycling under 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 a ThO2 matrix with the assistance of proton cyclotrons. TORIA operates at a k-eff of about 0.98, and can burn transuranic (TRU) wastes that would be discharged from pyrochemical separation of spent nuclear fuels. The majority of separated TRU wastes are transmuted in multiple units of a large-scale SFR in order to allow the sustainability of Korea's 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. The ultimate waste from the SFR-TORIA symbiosis will be transformed into intermediate level waste, requiring an institutional control period of less than 300 years.

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

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 the lead-bismuth eutectic coolant. The PASCAR design employs a pool-type capsule including a core of U-TRU-Zr-alloy fuel rods in an 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 UO2 fuel rods in a hexagonal geometry. Like the PASCAR design, URANUS is targeted for 20 years of operation without on-site refuelling at an electric power up to 100 MW with a Rankine cycle efficiency of 40%. 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 solutions to meet the demand for passive safety and security at competitive levelised cost of electricity.

Current URANUS R&D is focused on i) three-dimensioanl neutronic and thermal-hydraulic analysis code development; ii) corrosion-resistant Functionally Graded Composite (FGC) materials production; and iii) an integral mock-up test of about 1/200 scale (about 500 kW) using electrical heaters. In this regard, a coupled code called MARS-FREK has been developed, which is capable of calculation of thermal feedback in several reactivity-induced transients by coupling a three-dimensional reactor kinetics module FREK and a one-dimensional system code MARS. As part of the material development, a group of researchers designed a FGC tube pilgering process using three-dimensional finite element analysis. In this study, it was shown that the curvature and plastic strain are developed on the rolled product with same roll speed and same friction coefficient, and two methods of controlling the upper/lower roll speed ratio and adjusting the upper/lower friction coefficient and contacts are suggested to ensure manufacturability. The mock-up, designated as Pool-type Integral Leading test facility for Lead-Alloy-cooled Small Modular Reactor (PILLAR), has been designed and will be built and operated by early 2017.

A new approach for reactor core design has been tried with an inverted core concept that reverses the nuclear fuel region and coolant channel. With a preliminary neutronic study, it is found that the diameter of the active core can be reduced and a more compact design can be achieved. The reduction of the core diameter improves the economy, productivity and transportability of SMRs.

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 Research Centre Řež laboratory (Czech Republic) in December 2014. The consortium successfully involved a number of additional European partners through the signature of a number of Memoranda of Agreement expanding throughout Europe as much as possible the interest in the development of lead technology.

In 2016, the main activities related to the ALFRED design development included: i) development of a new conceptual design configuration for the primary side; ii) evaluation of options for steam generators (SGs), including bayonet double wall as well as helical SGs configurations; iii) evaluation of different options for primary pumps; iv) integration of a new DHR system in the primary pool; v) optimisation studies of core and fuel assemblies; and vi) development of a new anti-freezing system for DHRs; a testing facility of the system is expected to start construction in 2017 following a grant from the Italian government.

During 2016 the FALCON activities suffered from the lack of sufficient funding, mainly due to a delay of the activities dedicated to securing structural funds in Romania. The situation will hopefully be improved in 2017 as the FALCON consortium is expecting to promote the ALFRED Project as one of the major projects for Romania during this year.

Relevant material activities were performed in 2016. Studies to further optimise the composition of double-stabilised DS4 low-swelling austenitic steel were performed. In the frame of experimental qualification of corrosion protection barriers, several coating techniques were developed and tested. FeCrAl alloys were deposited by the physical vapour deposition (PVD) method on AISI 316L, AISI 304, AISI 441 and P91 steels. Good adhesion but non-uniform thickness were observed. The chemical vapour deposition technique, suitable for complex shapes, was used as well to coat P91 and 15-15 Ti by the same alloys as by PVD. Namely, the pack cementation and diffusion coating processes were employed. Although promising, these processes, due to their high temperatures, induce modifications of the substrate microstructure. Two other different techniques were adopted: Thermal spray (high velocity oxygen fuel /HVOF/) of FeCrAlY alloy and laser ablation (pulsed laser deposition /PLD/) of alumina. The former assures absence of porosities and good adhesion but need further mechanical grinding to reduce the excessive roughness. On the other hand, PLD produces excellent coating under any point of view. Corrosion tests were performed showing excellent resistance by all the barriers except on the local defects of PVD. PLD specimens were exposed to heavy ion irradiation to verify their damage resistance up to 450 dpa. Progressive crystallisation of the amorphous phase was observed, but no delamination or cracking was found, even at the highest levels of irradiation damage.

In the area of core design, refuelling studies of the ALFRED reactor were made. The conceptual design of an experimental facility to test operating procedures for FA handling and the reliability of the fuel-handling machine was drafted. The core design was optimised by enlargement of the fuel assembly to avoid overheating of corner pins. An improvement of core shielding by additional rings of dummy elements was studied. The studied solution, besides extending the inner vessel lifetime, would also allow temporary positioning of spent fuel elements to allow decay and to provide enough space to adjust the critical mass.

With respect to MYRRHA, the Front-End Engineering Design (FEED) contract, awarded in October 2013 to a consortium formed by Areva, ANSALDO, EMPRESARIOS AGRUPADOS and GRONTMIJ, was suspended in the beginning of 2015. The reason for the suspension is that a deep review of the primary system configuration of MYRRHA was needed.

Activities have been conducted during 2016 to improve the reactor design configuration and are expected to be continued as well during 2017.

In September 2016 the MYRRHA Management Team took a major decision to concentrate the activities on the development of a 100 MeV accelerator, expected to be operational by 2024. At the same time activities on reactor and upgrade of the accelerator to 600 MeV are underway in order to be able in 2024 to start procurement for both reactor and the accelerator upgrade.

The Euratom H2020 call for project proposals, launched in September 2015, closed on 5 October 2016. A number of projects related to lead technology have been proposed including those on technology and component qualification, experimental facilities, material R&D, as well as on studies for LFR-SMR solutions. Successful projects are expected to start in 2017. In support to member states, Euratom also conducts R&D in direct actions implemented by the European Commission's Joint Research Centre. This includes development of experimental facility for pre-normative testing of candidate structural materials for LFRs.

Main activities in China (observer)

In China, the Chinese Academy of Sciences (CAS) launched a project to develop ADS and lead-based fast reactors technology since 2011. The China Lead-based Reactor (CLEAR), proposed by the Institute of Nuclear Energy Safety Technology, was selected as the reference reactor for ADS development, as well as for the technology development of the Generation IV lead-cooled fast reactor. The programme consists of three stages with the goal of developing a 10 MWth lead-based research reactor (CLEAR-I), a 100 MWth lead-based engineering demonstration reactor (CLEAR-II) and a 1 000 MWth lead-based commercial prototype reactor (CLEAR-III).

To promote the CLEAR Project successfully, INEST is deeply involved in the reactor design, reactor safety assessment as well as in design and analysis, software development, testing activities using lead-bismuth experimental loops, key technologies and components R&D activities.

The 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 the ADS transmutation system and the LFR technologies. The KYLIN series lead-bismuth eutectic experimental loops have been constructed and have operated for more than 10 000 h. R&D activities on structural material corrosion experiments, oxygen control technology development, 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 the lead-based reactor, the lead alloy-cooled non-nuclear reactor CLEAR-S, the lead-based zero power nuclear reactor CLEAR-0, and the lead-based virtual reactor CLEAR-V are under realisation.

In addition, series of innovative concepts for different purposes are being developed to enlarge the application perspective of lead-based reactors and technology, which are not only for ADS and fast reactor, 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 lead-bismuth eutectic 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 a small modular reactor (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 and more recent efforts at the University of Alaska and Texas A&M University to design a Passively Operated Lead Arctic Reactor.

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/fertile material is dissolved in the molten salt and it serves both as fuel and coolant in the primary circuit. In the second subclass, the molten salt serves as the coolant to a carbon moderated, ceramic fuel core similar to that employed in VHTRs. In order to distinguish the reactor types, the solid fuel variant is typically referred to as a FHR. Within the GIF PSSC-MSR (provisional System Steering Committee), research is performed on both subclasses, under an MOU signed by Euratom, France, Russia, Switzerland and the United States, with China, Japan, Australia and Korea and as observers.

Fast spectrum molten salt reactor concepts

From the outset MSRs were thermal-neutron-spectrum graphite-moderated designs. Since 2005, liquid-fuelled MSR R&D has focused on fast spectrum MSR options combining the generic advantages of fast neutron reactors (extended resource utilisation, waste minimisation) with those related to molten salt fluorides as both fluid fuel and coolant (low-pressure, high boiling temperature and, optical transparency). Recent MSR developments in Russia on the 1 000 MWe molten salt actinide recycler and transmuter (MOSART) and in France on the 1 400 MWe thorium molten salt reactor (MSFR) address the concept of large power units with a fast neutron spectrum in the core without graphite moderator. The fast neutron spectrum molten salt reactors open promising possibilities to exploit the ^{232}Th-^{233}U cycle and can also contribute, in the transmuter mode, to significantly diminishing the radiotoxic inventory from current-reactor used fuel in particular by lowering the masses of transuranic elements (TRU). In 2016, the US government began supporting development of a molten chloride fast spectrum reactor (MCFR) concept that has been under private development by TerraPower Inc. for the past five years. The MCFR is intended to have a very hard neutron spectrum to avoid requiring any fissile material input after its initial core load or separation of fissile materials from the remainder of the fuel salt.

Fast MSRs have strongly negative reactivity coefficients, a unique safety characteristic not found in solid fuel fast reactors. Compared with solid-fuelled reactors, those systems have lower fissile inventories, no radiation damage constraints on attainable fuel burnup, reduced radiation damages to materials since no solid matter is located in the centre of the core, no used nuclear fuel, no requirement to fabricate and handle solid fuel, and a homogeneous isotopic composition of fuel 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.

In 2016, the private company Kairos Power was founded in the United States to commercially develop FHRs. Kairos' technology is an outgrowth of the pebble-bed FHRs that have been investigated by the University of California at Berkely and other US universities for the past decade. 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.

R&D objectives

The common objective of MSR projects is to propose a conceptual design with the best system configuration – resulting from physical, chemical and material studies – for the reactor core, the fuel salt processing unit and wastes conditioning. The mastering of MSR technically challenging technology will require concerted, long-term international R&D efforts, namely:

- additional studying the salt physical, chemical and thermodynamic properties;
- system design and safety analysis, including development of advanced neutronic and thermal-hydraulic coupling models;
- development of advanced materials, including studies on their compatibility with molten salts and behaviour under high neutron fluxes at high temperature;
- mastering of corrosion and tritium release prevention technologies, based on proper molten salt Redox control;
- development of efficient techniques of gaseous fission products extraction from the fuel salt by He bubbling;
- fuel salt processing flowsheet, including reductive extraction tests (actinide-lanthanide separation);
- development of a safety, safeguards, security and proliferation resistance approaches dedicated to liquid-fuelled reactors.

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

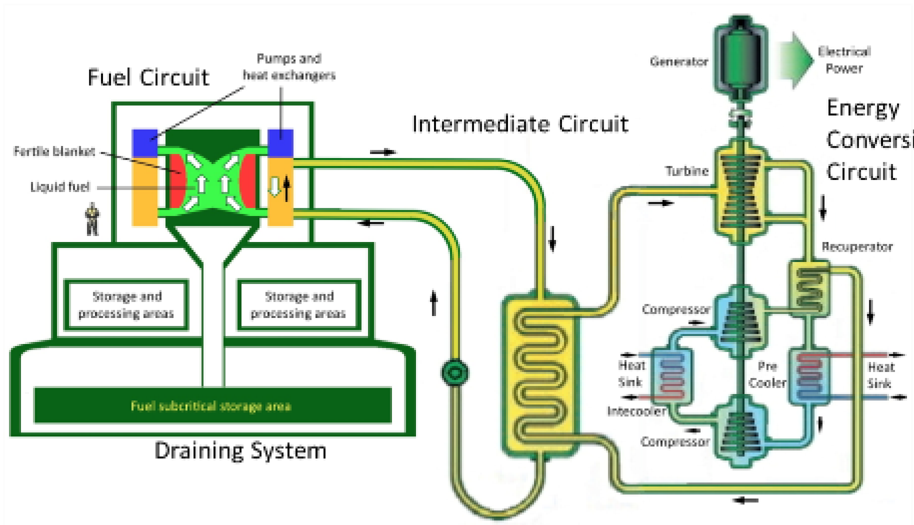
- continuous fibre ceramic composites;
- FHR specific fuel elements and assemblies.

Main activities and outcomes

Design evolutions of the MSFR fuel circuit

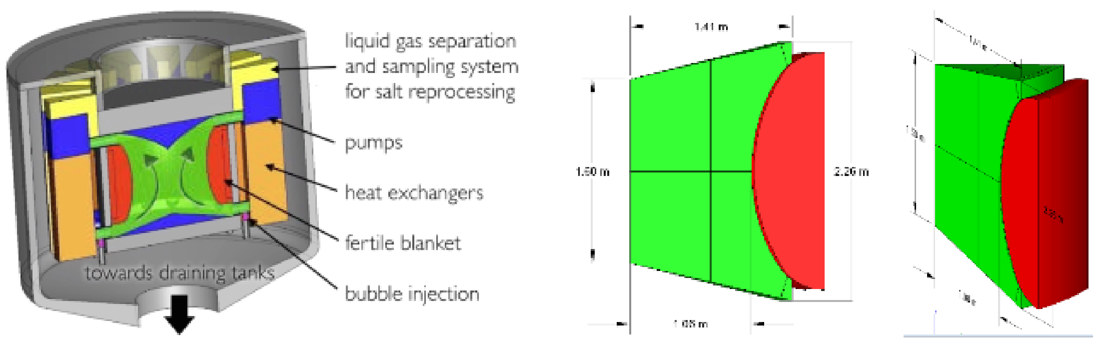
The MSFR plant includes three circuits involved in power generation (see Figure 3.7): the fuel circuit, the intermediate circuit and the power conversion circuit. These circuits are associated to other systems composing the whole power plant: the emergency draining system, the routine draining system to the storage areas and the reprocessing units (Allibert et al., 2015).

Figure 3.6: **MSFR power plant**



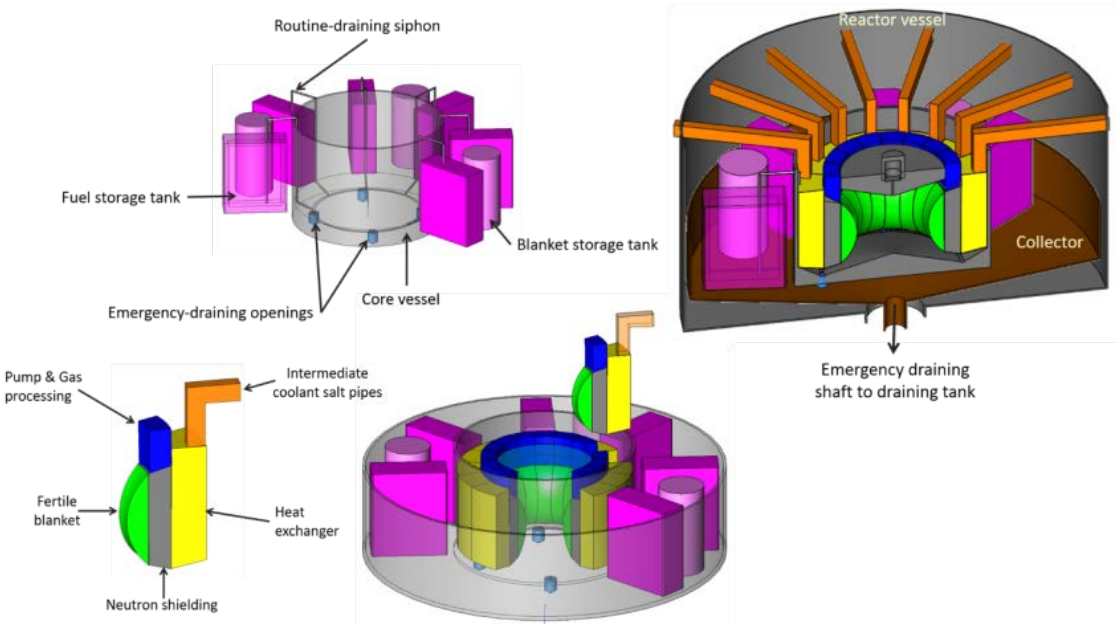
The fuel circuit (see Figure 3.6 – left), defined as the circuit containing the fuel salt during power generation, includes the core cavity and the recirculation loops (also called "sectors" in the following) comprising the inlet and outlet pipes, a gas injection system, salt-bubble separators, pumps and fuel heat exchangers (Brovchenko et al., 2014). The dimensions of the core cavity are given in Figure 3.8 (middle and right); (Laureau, 2015a; Rouch et al., 2014).

Figure 3.7: **Schematic representation of the reference MSFR fuel circuit (left) and dimensions of the active core cavity (middle and right) with the fuel salt in green and the fertile blanket in red**



To prevent the risk of fuel salt leakage through pipe rupture highlighted by preliminary safety studies (Brovchenko et al., 2013), an improved design of the fuel circuit is being studied in the frame of the SAMOFAR Euratom Project of the Horizon2020 programme. The core is enclosed in a vessel that serves as the container for the fuel salt as illustrated in Figure 3.7 (top). The 16 circulation loops are in the form of 16 sectors arranged circumferentially around the vessel (Figure 3.8, bottom right), inserted from the top as shown in Figure 3.8.

Figure 3.8: **Segmented geometry of the MSFR circuit**



Each sector comprises: a heat exchanger, a circulation pump, a bubble injector and a gas separator, a blanket salt tank, and a heat exchanger (Figure 3.9, bottom left). Each sector is connected to an intermediate fluid circuit (four circuits, each feeding four sectors, for example).

Based on the same approach, the design of the emergency draining system of the MSFR is also under study both regarding neutronic and thermal studies, in collaboration between the Centre National de la Recherche Scientifique (CNRS), KIT and EDF. This task is undertaken in 2017 in the frame of the SAMOFAR Project.

Based on this new design of the MSFR system, a preliminary accident list of the fuel circuit and the emergency draining system has been identified thanks to:

- the analysis of the accident type identified for current operated reactors (PWR);
- the Euratom Evaluation and Viability of Liquid fuel fast reactor systems (EVOL) Project of the Framework Program 7 (Brovchenko et al., 2014; Wang et al., 2014; Brovchenko, 2013; Brovchenko et al., 2013);
- accidental transient calculations of the MSFR with dedicated tools (Laureau, 2015a; Laureau et al., 2015c);
- preliminary systemic risk analyses with a qualitative re-evaluation to take into account new design.

Discussions and exchanges have been launched in 2016 with the RSWG to have an evaluation and expert advices on the safety analysis and approach currently developed for such liquid circulating fast reactors. This will continue in 2017.

SAMOFAR European Project (H2020)

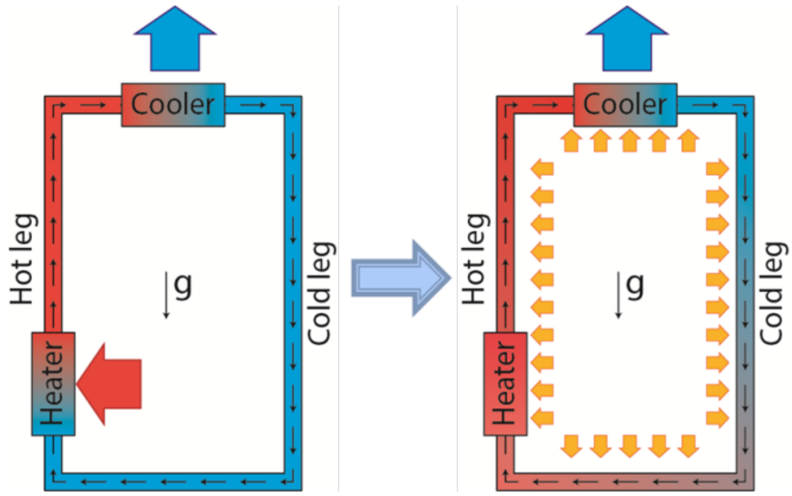
Objectives of the project

The Safety Assessment of the Molten Salt Fast Reactor (SAMOFAR) Project is one of the major research and innovation projects in the Horizon 2020 Euratom research programme with a total budget of around USD 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, to deliver a breakthrough in nuclear safety and optimal waste management, and to create a consortium of stakeholders containing TSO's and industry to advance with the MSFR up to the demonstration phase of this technology. In total 11 partners participate in the project as shown in Table 3.2.

Progress in the project

POLIMI (Italy) converted the DYNASTY facility to a loop with a distributed heating system (see Figure 3.9). Originally, the DYNASTY facility was a simple loop to study the dynamics of cooling systems with a separate heater section and a separate cooling section. In the framework of SAMOFAR, the loop has been converted into a facility with a homogeneously distributed heating section, resembling the fact that in an MSR the fission products are homogeneously distributed in the whole primary circuit. In DYNASTY; this is accomplished by an external electrical heating after verifying computationally that external heat generation well resembles internal heating. This difference with solid fuel nuclear reactors is expected to have a significant influence on the ranges of operating parameters at which the reactor is stable. In 2016, POLIMI has finished the conversion of the DYNASTY loop. First experimental results are expected in 2017.

Figure 3.9: **Layout of the original DYNASTY facility (left) and adapted version**



Note: The latter version will be extended at a later date to include the intermediate cooling circuit as well.

POLITO (Italy), in co-operation with CNRS in France, has set up a framework for safety analysis based on the Functional Framework Mode and Effect Analysis (FFMEA), which has previously been applied to the Demonstration Power Station (DEMO) fusion reactor design. The method has been further developed and will be applied to the MSFR in 2017.

KIT (Germany) has set up calculation models to simulate the decay heat removal in the emergency draining tanks beneath the reactor core. In case of emergencies, the salt is supposed to be drained and passively cooled. CNRS has made up a preliminary design of special cooling assemblies to be assembled in the draining tank, containing a water coolant channel and a fuel salt storage volume (see Figure 3.11). In between these two regions a material is placed to guide the heat from the hot fuel salt to the water. Two cases were investigated by KIT: one in which the volume is filled with solidified inert salt and one in which the volume is filled with steel. In the first case the temperature of the fuel salt may become too high because of the low heat conduction in the inert salt, while in the second case a crust layer may form at the inner side of the hot fuel salt channel. Further optimisation is foreseen in 2017.

Technische Universiteit Delft (Netherlands) has worked on several issues. First of all a numerical code package has been programmed based on discontinuous Galerkin methods both for the deterministic neutron solver and the computational fluid dynamics code. Benchmarking is going on and the first results are expected in 2017. Secondly a viscometer has been developed based on an ultrasound method. Testing of the method with a well-known low-temperature fluid is in progress, after which a new version able to work with hot salt will be designed and built. More results are expected in 2017. Thirdly an irradiation rig has been designed and built to irradiate various salts at moderate temperature in the research reactor of Technische Universiteit Delft in order to measure radiolysis of salt due to gamma radiation. These measurements will be conducted in co-operation with NRG in Petten to support the Salt Irradiation and Examination of Nuclide Trapping (SALIENT) irradiations in the High-Flux Reactor in Petten. Fourthly binary phase diagrams have been calculated and measured with a Differential Calorimeter System on two mixtures of salts, namely LiF-CsF and CsF-CsI. Fifthly, a preliminary set-up to study helium bubbling has been designed and built to study the parameters of influence on the separation efficiency of the noble metals by helium bubbling. After scoping studies, a new facility will be built containing hot salt.

Figure 3.10: **Provisional design of the fuel salt storage assemblies in the Emergency Decay Tank**

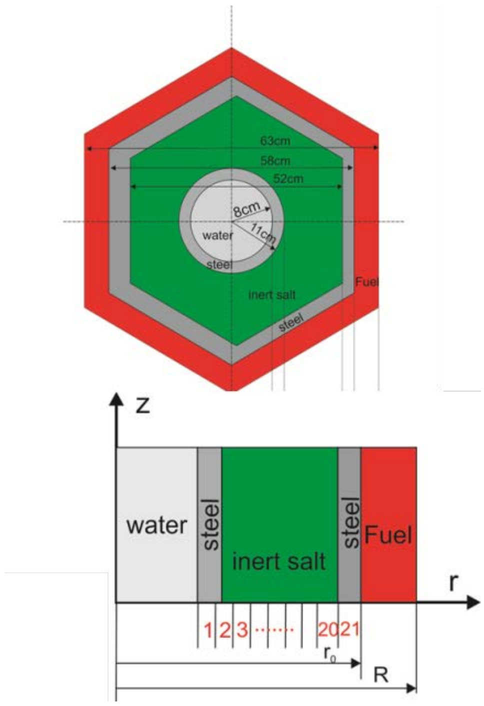


Table 3.2: **Organisations participating in the SAMOFAR Project**

Number	Organisation name	Country
1	Technische Universiteit Delft (TU Delft)	The Netherlands
2	Centre National de la Recherche Scientifique (CNRS)	France
3	Joint Research Centre (JRC), European Commission	Germany
4	Consorzio Interuniversitario Nazionale per la Ricerca Tecnologica Nucleare (CIRTEN)	Italy
5	Institut de Radioprotection et de Sûreté Nucléaire (IRSN)	France
6	Centro de Investigaciony de Estudios Avanzados del Instituto Politecnico Nacional (CINVESTAV)	Mexico
7	Areva NP SAS (Areva)	France
8	Commissariat à l´énergie atomique et aux énergies alternatives (CEA)	France
9	Électricité de France S.A. (EDF)	France
10	Paul Scherrer Institute (PSI)	Switzerland
11	Karlsruher Institut für Technologie (KIT)	Germany

Chemistry and reprocessing for MSFR concept

A reprocessing scheme has been established in the EVOL Project (European Project FP7). The different steps have been validated both by bibliographic study and some experimental determinations. In the frame of SAMOFAR Project (H2020 European Project), one objective is to study a design of the processing plant based on chemical and safety issues. Indeed, the viability of the concept MSFR requires the reprocessing (both online and offline) of the fuel which combines the management of gaseous fission products, liquid metals and molten salts. Both chemical and radioprotection safety will be assessed in SAMOFAR Project including also the material issue of the chemical plant.

The design and the safety of the chemical plant requires the determination of the nuclide inventory at various stages of the reprocessing. This inventory can be estimated on the basis of core inventory and basic data such as activity coefficients and redox potential values. Some elements have been chosen in order to examine their specific behaviour during the various stages of the reprocessing:

- Iodine as a representative of halogens: It was shown that iodine can be extracted during the fluorination step if oxide ions are not in the salt. In presence of oxides, iodine is oxidised to iodate (which is not in gaseous state) and not removed from the salt. The case of iodine is particular because its behaviour is strongly dependent on the temperature. Studies at low and high temperature will be done (500-700°C).
- Zr because it is produced at a high level in the reactor core. Its extraction has to be considered. A dedicated step will be proposed in the processing scheme, probably between the fluorination and the first reductive extraction. Preliminary studies have shown a strong dependence of zirconium behaviour with the temperature. At 650°C, in FLiNaK molten salt, the instability of metallic zirconium has been observed in presence of ZrF_4 which could be explained by a disproportionation reaction to form ZrF_2. The behaviour of ZrF_4 has to be studied in LiF-ThF_4 molten salt in order to conclude on the stable oxidation states of this element in the fuel salt. Secondly, the influence of temperature will be examined. Indeed, the vaporisation temperature of pure ZrF_4 is close to 900°C. It is probable that a part of ZrF_4 will remain in the salt during the fluorination step. ATG-DSC measurements will be done to study the behaviour of LiF-ThF_4-ZrF_4 salt at given temperatures ranging between 700 and 900°C under argon flux.

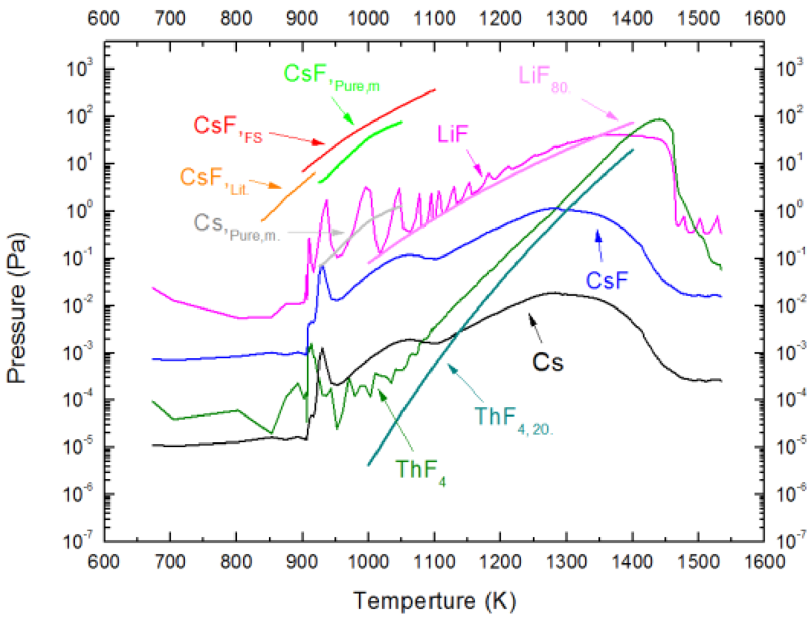
- U because it is the fissile material and used to control the redox potential of the salt. We have determined the activity coefficients of UF_4 and UF_3 in LiF-ThF_4 molten salt at 650°C.
- In the future, transition metals and alkali earth elements will be studied to follow their reactivity in the chemical plant and calculate the radioprotection in the chemical plan.

Volatility of fission products and influence on melting temperature

Caesium and iodine which are formed during a fission process in a nuclear reactor are considered as major fission products responsible for environmental burden upon nuclear accident. From safety point of view it is thus important to understand its release mechanism upon overheating. In this context we performed an experimental investigation of caesium iodide and caesium fluoride behaviour in molten salt reactor fuel during high-temperature event, demonstrating if these major fission products stay retained in the molten fuel as in contrast to the solid oxide fuel used in commercial power plants nowadays.

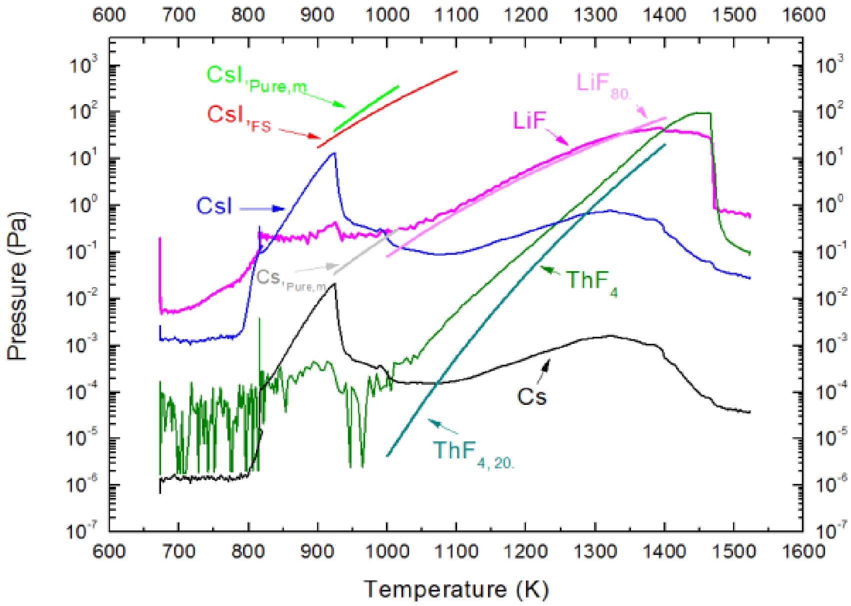
A series of measurements using Knudsen effusion mass spectrometry have been done on samples containing CsF and CsI compounds (as thermodynamic stable forms of caesium and iodine in molten salt fuel) and it was found that while CsF is fully dissolved in molten LiF-ThF_4 fuel salt, and thus suppresses its volatility, the CsI compound has very limited solubility in the fuel salt and the undissolved CsI would remain a volatile element in case of a high-temperature event. We note that this experiment was performed with CsI concentration of 1 molar % which is overestimated amount in comparison to what will be present in the real fuel, however it showed the tendency of the compound behaviour. Figure 3.11 shows the decrease of CsF volatility (blue line) when dissolved in the LiF-ThF_4 matrix by ca. factor of 100 compared to pure, undissolved CsF (light green line). On contrary, Figure 3.12 shows CsI release when mixed with LiF-ThF_4 matrix (blue line) which is about the same order of magnitude as pure CsI (light green line).

Figure 3.11: **The vapour pressure of 1 mol% CsF in a eutectic mixture of LiF-ThF_4**



Note: Blue and black lines are the partial vapour pressures of gaseous CsF, resp. Cs gaseous species in equilibrium with condensed CsF, while light green line corresponds to measurement of pure CsF, indicating much higher volatility compared to dissolved CsF.

Figure 3.12: **The vapour pressure of 1 mol% CsI in a eutectic mixture of LiF-ThF$_4$**



Blue and black lines are the partial vapour pressures of gaseous CsF, resp. Cs gaseous species in equilibrium with condensed CsF, while light green line corresponds to measurement of pure CsF, indicating much similar volatility compared to CsF in the mixture, demonstrating its reluctance to dissolve in the fuel matrix.

The volatility of fission products is not the only safety concern, but the influence on the melting point is also an important parameter to understand the fuel behaviour with respect to the operational time. Therefore, measurements using differential scanning calorimeter have been performed on samples containing 1 molar % of CsI and CsF dissolved in FLiNaK and LiF-ThF$_4$ solvents revealing no, or slight lowering of melting temperature.

Thermodynamic database development

As explained in many earlier, published papers, the development of thermodynamic database is a strong tool to predict many relevant properties of molten salt reactor fuel. In this context we have extended the JRC database by adding CsF which is considered as one of the likely formed fission product compounds during irradiation. We first focused on the full thermodynamic assessment of the ternary CsF-ThF$_4$-LiF system, which in the first stage required optimisation of the three binary subsystems: LiF-CsF, LiF-ThF$_4$ and CsF-ThF$_4$. Since the first two systems were already included in the database, the subject of study was the remaining CsF-ThF$_4$ system which has been assessed using various techniques, including differential scanning calorimetry for the determination of equilibrium data, Knudsen effusion mass spectrometry for the determination of activity coefficients (and thus the vapour pressure), complemented with series of X-ray diffraction measurements to reveal the structure and stability of various intermediate compounds. Using this novel data input the CsF-ThF$_4$ system has been assessed and is illustrated, together with the measured equilibrium points, in Figure 3.14. Figure 3.15 shows the activity coefficients obtained by performing vapour pressure measurements on series of compositions at T=1 350 K.

With the thermodynamic assessment of the CsF-ThF$_4$ system the ternary CsF-ThF$_4$-LiF system has been calculated with the liquidus projection given in Figure 3.15. It revealed lowest melting point at CsF-ThF$_4$-LiF (8-26-66 mol%) composition and T = 752 K. The lowest eutectic point was furthermore experimentally verified by differential scanning calorimetry showing a very good agreement to the calculated phase diagram, within ±2K.

Figure 3.13: **The assessed CsF-ThF₄ phase diagram**

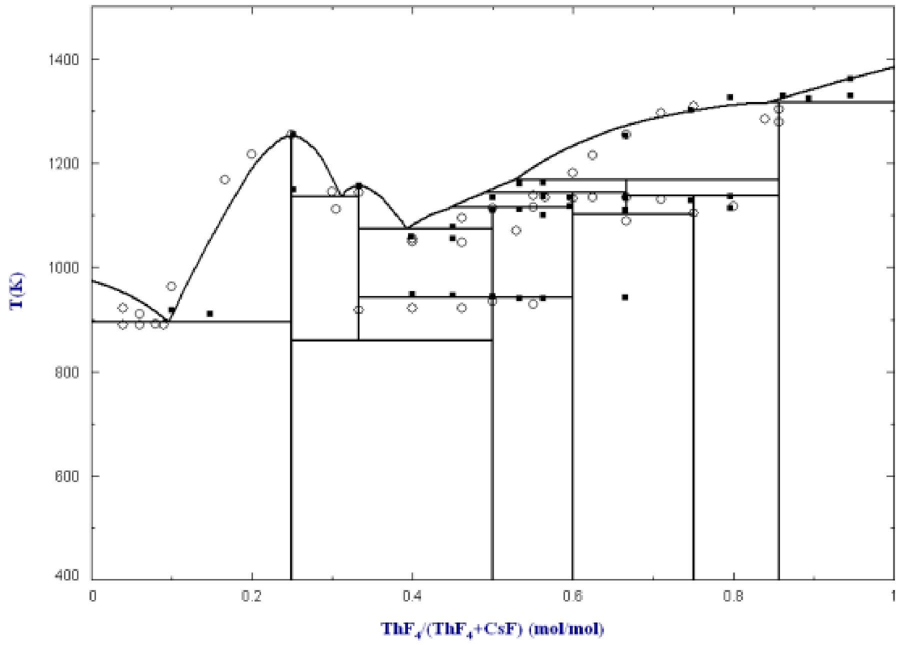


Figure 3.14: **Activity coefficients of the CsF and ThF₄ species obtained by Knudsen effusion mass spectrometry**

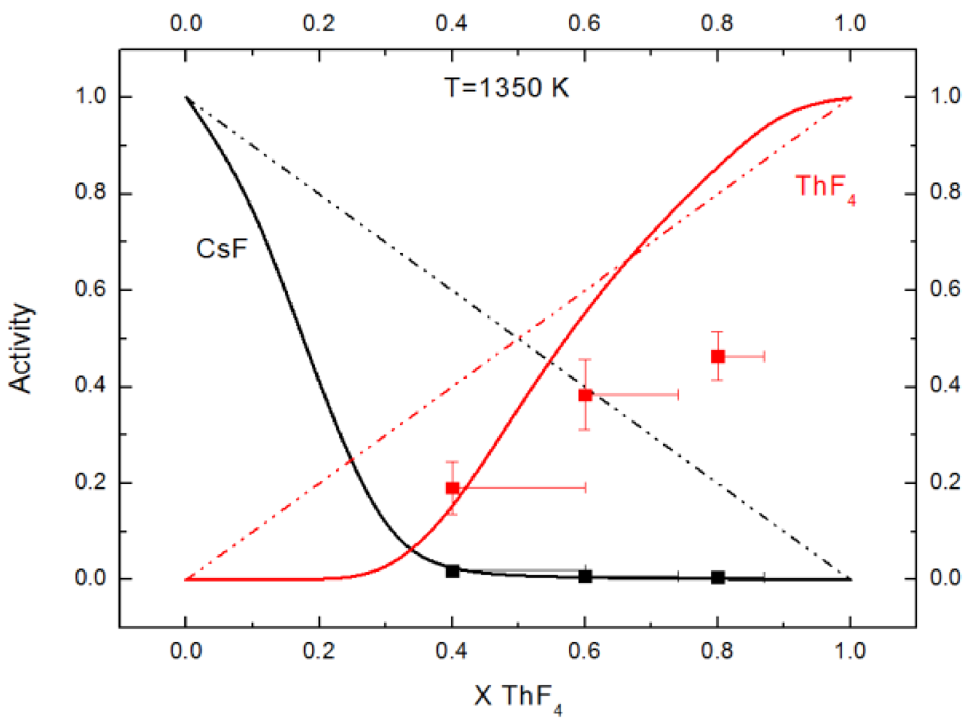
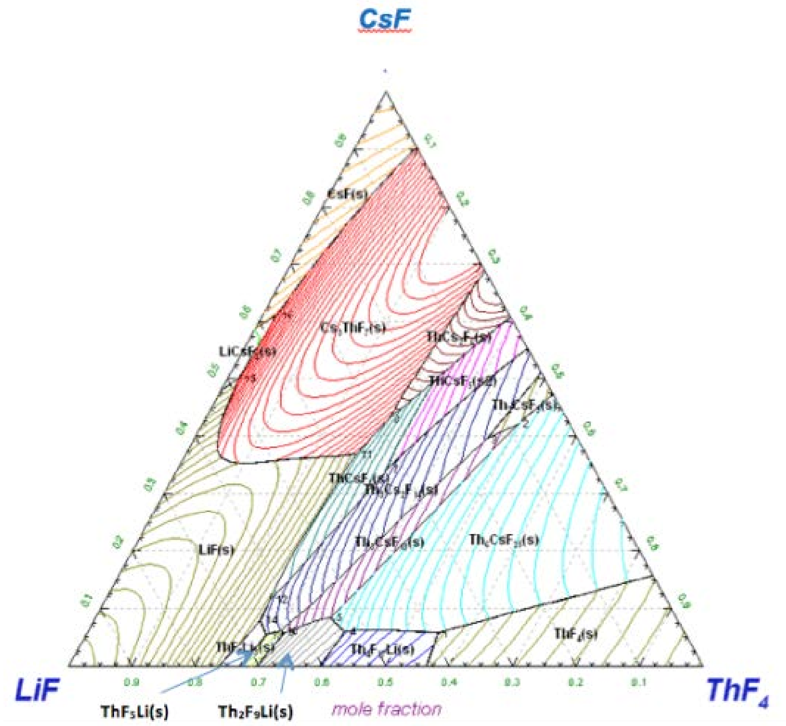


Figure 3.15: **Calculated liquidus projection of the ternary CsF-ThF_4-LiF phase diagram**



The phases indicate primary crystallisation forms for given regions.

Synthesis and electrochemistry of actinide fluorides

During the past three years, the experimental equipment for the synthesis of pure actinides fluorides has been installed and methods for preparation of very pure ThF_4 and UF_4 have been established and verified. The work in 2016 was focused partly on the synthesis of sufficient amounts of ThF_4 needed for electrochemical studies using the existing equipment, but the main task was to modify and improve the fluorination Inconel reactor. The new design enables a flow-through fluorination, which should decrease the corrosion observed in the past due to inefficient removal of the formed water, which condensed on the colder parts of the reactor, when HF gas was still present. In addition, the possibility of supplying two gases at the same time for the reaction was included in the new design, enabling simultaneous fluorination and reduction by applying e.g. HF and H_2 gas in the same time. The reactor was manufactured in 2016 and will be installed and tested beginning 2017. A scheme showing a cross section of the reactor in the furnace is shown in Figure 3.16.

The studies of electrochemical properties of ThF_4 in LiF-CaF_2 melt were restarted after a long break needed for synthesis works. An electrochemically pure melt containing no residual oxygen was prepared, as shown by a linear sweep voltammogram (LSV) measured on a gold working electrode and a cyclic voltammogram (CV) taken on a tungsten electrode, as shown in Figure 3.17. The LSV before the melt purification (red line) indicated very significant oxygen formation on the cathode, despite the melt was purchased as moisture and oxygen free and packed under Ar atmosphere in a sealed ampoule. After bubbling of pure HF gas into the melt, the melt was purified and all oxygen was removed, as proven by the LSC after purification (blue line), where only oxidation of the electrode material is visible at potentials more positive than corresponding to oxygen evolution. On the CV, the only significant peak corresponds to decomposition of the carrier melt, i.e. redox couple Li/Li+.

However, after addition of ThF_4, some unexpected behaviour was observed and the melt requires further treatment, which was not possible to provide due to technical problems. The work will continue during 2017.

Figure 3.16: **A scheme of an improved design of the fluorination reactor enabling flow-through dynamic fluorinations and simultaneous supply of two reaction gases**

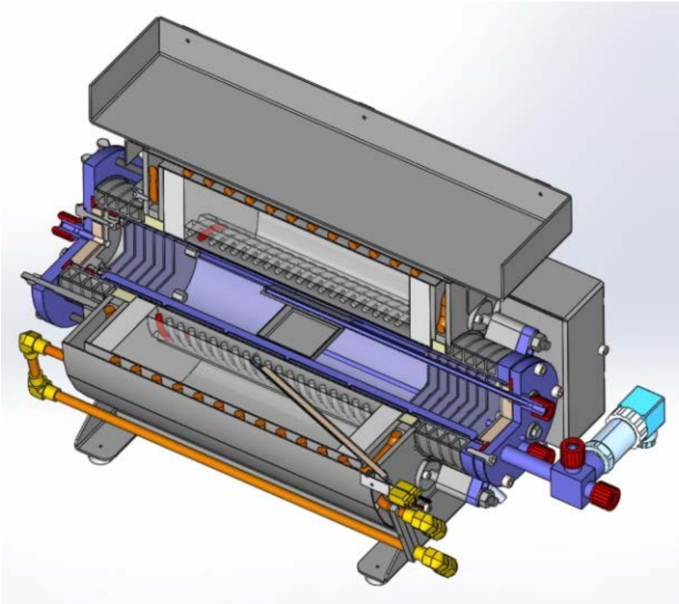
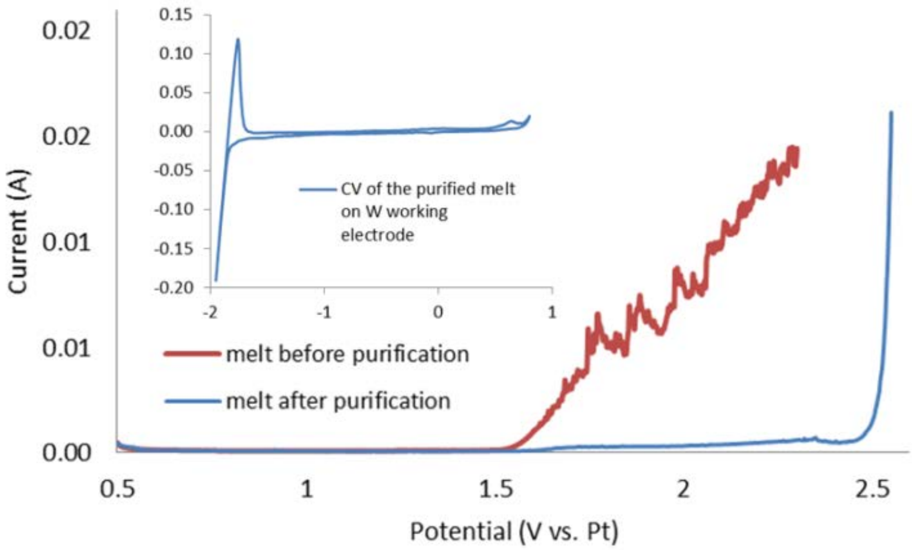


Figure 3.17: **Linear sweep voltammograms of the LiF-CaF_2 melt before purification (red line) and after purification (blue line) using HF gas (Au working electrode, PtO_2/O_2- quasi-ref. electrode, 10 mV/s, 850°C) and a cyclic voltammogram of the purified melt (insert) showing the melt is electrochemically clean (W working electrode, 200 mV/s, 850°C)**



Experimental and modelling developments at LPSC (Grenoble, France)

The SWATH experiment (Salt at Wall: Thermal Exchanges) is one of the research activities developed by the CNRS (Laboratoire de Physique Subatomique & Cosmologie [LPSC], Grenoble, France) within the European H2020 SAMOFAR Project.

The main objectives of SWATH are to improve molten salt numerical models for flow, turbulence and solidification descriptions using validation on experimental results. A critical point of SWATH is to ensure that the experimental data uncertainties are significantly lower than the effects of the thermal-hydraulic phenomena that are being studied. Since the possible accidental configurations of the MSFR reactor (geometries and conditions) are not completely known, SWATH is focused on understanding the underlying physical principles rather than developing experimental correlations.

The SWATH experimental salt set-up has two parts: i) the SWATH "facility" which allows for the establishment of a controlled flow and ii) the test sections (sections with very simple geometries that will be used to investigate the salt phenomena) where the measurements are made. The FLiNaK (LiF-NaF-KF eutectic) has been selected as working fluid for SWATH. It is a good compromise since it means achieving a similitude with respect to the phenomena encountered in a molten salt reactor, and it has a convenient Prandtl and a high enough melting point to allow for investigation of the effect of radiative heat transfer.

The operation of SWATH facility is based on a discontinuous working principle in which the flow is established by regulating the pressure difference between two tanks, rather than using a pump. Figure 3.19 presents a sketch of the salt facility, which is composed of two salt storage tanks, pipes and a glovebox filled with argon required to host the interchangeable test sections. The pressure control system is designed to maintain a stable flow during the operation by regulating opening and closing tank vanes. This control system uses information related to the flow rate such as the pressure drop measured at specific loop components and also the salt level in the tanks. Salt mass flow rate is calculated from the variation of the tank levels measured by two independent methods: a laser beam system and an electrical contact system.

As implementation of classical flow visualisation techniques such as particle image visualisation is not practical in the salt, a water model (Figure 3.20) is used to study the accuracy of the CFD models predictions regarding the flow velocity field. Instead, the salt experimental set-up will be focused on the investigation of the heat transfers aspects of the models. The SWATH water mock-up facility is also used to evaluate experiment design options before building or modifying the SWATH-Salt facility or the test sections. Furthermore it serves to confirm that the pressure control system allows obtaining adequate steady conditions in the test section. The proper functioning of some other facility components will also be checked in the water mock-up (for example the siphon pipes and level measurement system).

SWATH test sections geometries are designed to study different salt phenomena. Only simple geometries will be used to minimise possible sources of errors and uncertainties in the experiments and also in the numerical models employed to model the experiments. The test sections can be divided into two groups:

- test sections used to perform molten salt phase change experiments;
- test sections used to perform heat exchange experiments.

The goal of the phase change experiments is the evaluation of the accuracy of the microscopic and macroscopic salt solidification models. Data obtained from the test sections will be used to investigate the accuracy of the solidification, turbulence and radiative heat transfer models for the fuel salt and also to provide general recommendations on the salt thermal-hydraulic modelling.

Figure 3.18: **Sketch of Salt SWATH set-up**

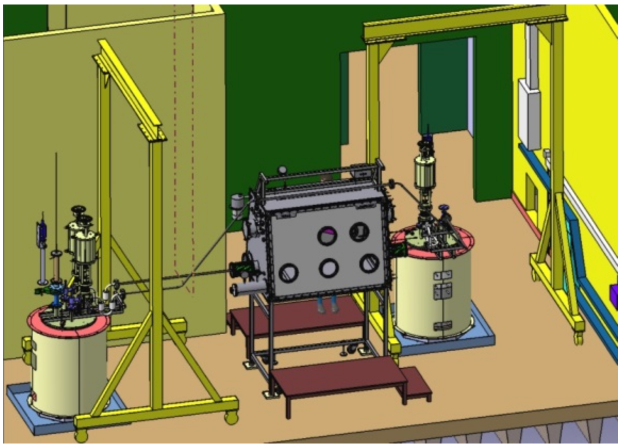


Figure 3.19: **SWATH water mock-up**

Recent development for MOSART concept (Russia)

"Rosatom" supported MSR activities continue to be limited to the 2 400 MWt Li,Be/F based MOSART design without and with U-Th support. No new significant R&D projects on the MOSART development under the agreement with "Rosatom" was started in 2016. Russian Foundation for Basic Research (RFBR) currently supports experimental studies in NRC "Kurchatov Institute" concerning fuel salt and material properties for MOSART design. Main results published in year 2016 by the NRC "Kurchatov Institute" within RFBR studies are given below:

Equilibrium distribution of lanthanium, neodimium, samarium and europium between molten flibe salt mixture and liquid bismuth

For the development of the fuel salt clean flow sheet for MSR design there are two main tasks, including i) multiple recycling of actinides with minimum losses to waste stream and ii) removal of soluble fission products (FPs), first of all lanthanides. The fuel salt clean-up scheme of the Li,Be/F MOSART is based on reductive extraction in liquid bismuth: i) in molten Li,Be/F – liquid Bi system An/Ln separation factors are quite high (see Table 3.3) and ii) lanthanides could be removed from LiF-BeF_2 based fuel salt by reductive extraction in liquid bismuth without any problem.

Table 3.3: **Separation factors of actinides and lanthanides relative to plutonium in the molten LiF-BeF_2/liquid Bi systems**

Element	LiF-BeF_2/Bi
Pu	1
Am	-
Cm	6
Nd	3 000
La	25 000

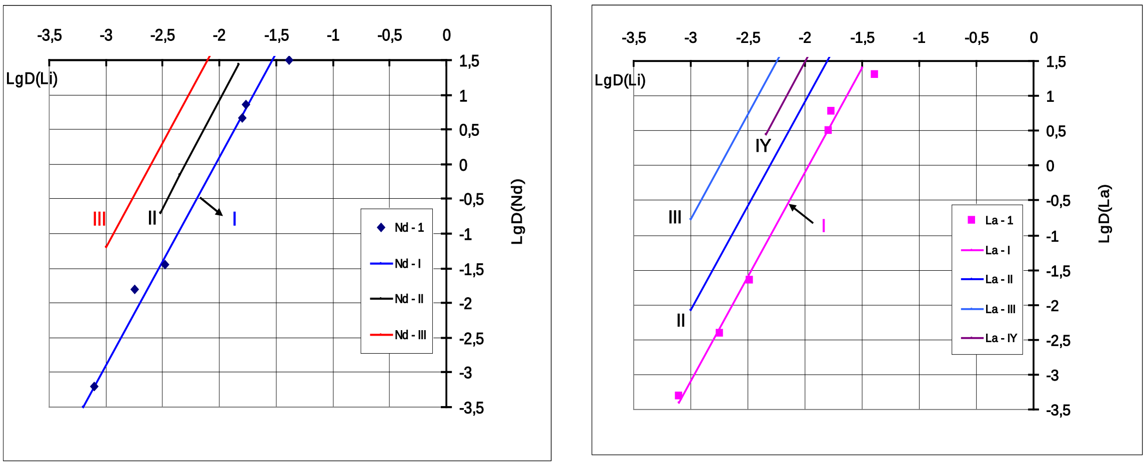
The extraction of lanthanum, neodymium and thorium from 15LiF-58NaF-27BeF_2, 60LiF-40NaF, 78LiF-22ThF_4, and 75LiF-5BeF_2-20ThF_4, (mole %) fluoride salt melts into liquid bismuth with a mixtures of lithium as a reducing agent was also studied at KI for temperature range of 580-750°C (Zagnit'ko, 2012). Equilibrium values of their distribution coefficients were measured. Straight lines in logarithmic co-ordinates (lgD(Nd,La,Th) vs.

lgDLi) describe our experimental data. Data from the experiments using mentioned above salts yielded a valence very close to 3 for the lanthanum and neodymium, as well as 4 for thorium. The measured distribution coefficients are consistent with the earlier data (Ferris, 1970) obtained for binary LiF-BeF_2 and LiF-ThF_4 systems as well as for ternary LiF-BeF_2-ThF_4 salt mixtures with account for the variation of temperature and salt composition. Beryllium contrary to Nd and La was almost not extracted into bismuth from beryllium containing salts. The distribution coefficients obtained for LiF-ThF_4 and LiF-BeF_2-ThF_4 salts with relatively high concentration of ThF_4 (about 20 mole%) cannot provide the effective separation between thorium and lanthanides in the fluoride salt/bismuth solutions. Excellent separation of thorium from lanthanides and alkaline-earth elements can be made by use of LiCl (Engel, 1979). The distribution coefficient for thorium is decreased sharply by addition of fluoride ion to the LiCl, although, the distribution coefficients for the rare earths are affected by only a minor amount.

The distribution coefficients XM(Bi)/XMFn of lanthanum, neodymium, samarium and europium were measured at KI depending on the lithium distribution coefficient D(Li) = XLi(Bi)/XLiF for a two-phase system "molten 73LiF-27BeF_2 salt mixture / liquid bismuth" at 600-610°C.

Figure 3.20 presents the measured values of the lanthanum and neodymium distribution coefficient D(La,Nd) = XM(Bi)/XMFn depending on the lithium distribution coefficient D(Li) = XLi(Bi)/XLiF for a two-phase system "molten 73LiF-27BeF_2 salt mixture/ liquid bismuth" at 600°C. Analysis of the presented data shows that the dependence of LgD(Nd,La) on LgD(Li) is linear and, accounting measurement errors, can be approximated as formulas: LgD(Nd) = 3LgD(Li) + 6.11 и LgD(La) = 3LgD(Li) + 5.91.

Figure 3.20: **Distribution coefficients in two-phase LiF-BeF_2/Bi system at 6 000°C for neodymium (left) and lanthanum (right)**

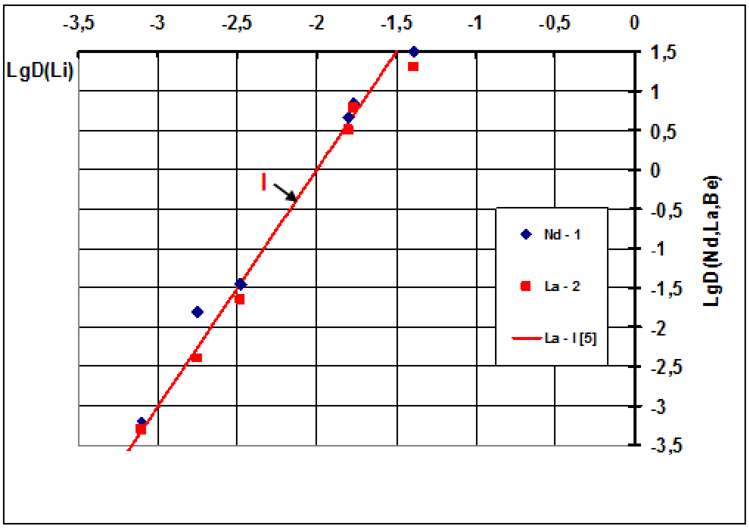


Left: I – current work; II – (Ferris, 1970; Ferris, 1971) for 66.7LiF-33.3BeF_2 eq. LgD(Nd) = 3LgD(Li) + 7.806; III – (Grimes, 1970) for 66LiF-34BeF_2 eq. LgD(Nd) ≈ 3LgD(Li) + 7.45

Right: I – KI data for 73LiF-27BeF_2 (Zagnit'ko, 2012); II – (Ferris, 1970; Ferris, 1971) for 66.7LiF-33.3BeF_2 eq. LgD(La) = 3LgD(Li) + 6.924; III – (Ferris 1970, Ferris 1971) for 56.9LiF-43.1BeF_2 eq. LgD(La) = 3LgD(Li) + 8.234; IV – for 66LiF-34BeF_2 eq. LgD(La) ≈ 3LgD(Li) + 8.04 (Grimes, 1970)

For comparison, our experimental results for Nd and La are given with the data obtained at 600°C for 66.7LiF-33.3BeF_2 and 56.9LiF-43.1BeF_2 (mole %) melts by Ferris et al. (Ferris, 1970; Ferris, 1971) as well with data received in 66LiF-34BeF_2 melt (mole %) by Grimes et al. (Grimes, 1970). As can be seen from Figure 3.21, the separation efficiency depends on LiF fraction in molten LiF-BeF_2 salt mixture. The values of D(Nd,La) decrease with the increase of LiF/BeF_2 mole ratio in salt composition at fixed temperature.

Figure 3.21: **Distribution coefficient of neodymium (1) and lanthanum (2) (I – calculation [Moriyama, 1983 and 1991] for lanthanium by eq. LgD(La) ≈ 3LgD(Li) + 6) in two-phase 73LiF-27BeF$_2$/Bi system at 6 000°C**

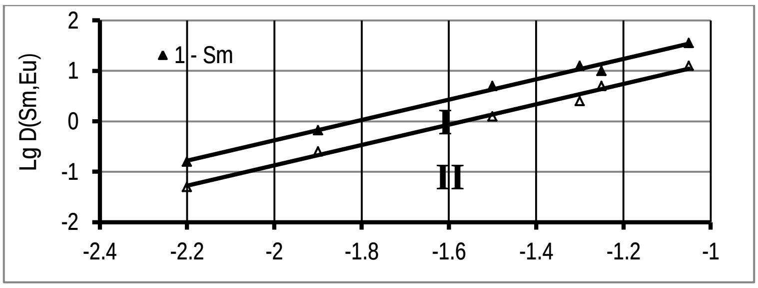


The valences of lanthanum and neodymium in the molten 73LiF-27BeF$_2$ salt mixture are 3. Neodymium is extracted more effectively compared to lanthanum. The separation factor is α = D(Nd)/D(La) = 1.58-1.60. It should be mentioned that unlike Nd and La, Be is not practically extracted into bismuth from melt 73LiF-27BeF$_2$ and its distribution coefficient makes up D(Be) <0.003 if D(Li) <0.04.

As it can be seen from Figure 3.21, our experimental dependences (Zagnit'ko, 2016) correlate with the ones calculated by Moriyama (Moriyama, 1983 and 1991) describing the dependence between the lanthanide equilibrium distribution coefficients and LiF molar fraction in the two-phase system "molten LiF-BeF$_2$ salt mixture/liquid bismuth". According to the Moriyama's calculations, theoretical curve "I" for lanthanum is plotted by equation LgD(La) ≈ 3LgD(Li) + 6 for molten 73LiF-27BeF$_2$ (mol.%) salt at 600°C.

Dependencies (see Figure 3.22) were also obtained between samarium and europium distribution and lithium distribution coefficient (Zagnit'ko, 2016): LgD(Sm) = 2LgD(Li)+ 3.65 and LgD(Eu) = 2LgD(Li)+3.15. Accounting for the effect of the LiF molar fraction in the molten salt, the measured coefficients of samarium and europium distribution correlate well with the data obtained by Ferris (Ferris, 1970; Ferris, 1971) and Grimes (Grimes, 1970) for composition 66LiF-34BeF$_2$ (mol. %).

Figure 3.22: **Measured at KI coefficients of samarium and europium distribution (points 1 and 2, respectively) for the two-phase system 73LiF-27BeF$_2$/Bi at 609°C**



Note: Lines I and II – calculations by equations for samarium and europium, respectively (Zagnit'ko2016).

Kinematic viscosity of molten salt fluoride mixtures

Kinematic viscosity of different molten salt fluorides mixtures was measured in the temperature range from liquidus to ~900°C by the oscillating cylinder method. The facility and measurement technique were described earlier (Merzlyakov, 2016). Table 3.3 presents the viscosity (ν, 10-6 m^2/s) vs. temperature (T, K) for the molten salts under study.

Table 3.3: **Viscosity vs. temperature for different melts (T, K).**

N	Composition, mol. %	$\nu \times 10^6$ (м^2/с)	RMS$\times 10^6$ (м^2/с)	Δt (^{0}C)
1	$73LiF - 27BeF_2$	2.773exp{3263.6*(1/T(K)-1/1003)}	0.059	609-900
2	99(1) + 1 CeF_3	2.15exp{1706*(1/T(K)-1/1009)}	0.066	600-934
3	98(2) +2 ZrF_4	1.943{1936.2*(1/T(K)-1/965.6)}	0.037	586-804
4	96(2) +4 ZrF_4	1.9506{2248.6*(1/T(K)-1/958.4)}	0.060	545-846
5	$85.5LiF – 14.5AlF_3$	1.2495exp{4852*(1/T(K)-1/1004)}	0.060	625 -846

In the temperature range, where the melts behave as ordinary liquid, the viscosity experimental values are approximated as: ν = A·exp [B/T]. The least square method was used to define the model parameters. The viscosity mean-square deviation estimated on the assumption of dispersion homoscedasticity makes up (0.06÷0.2) × 10-6 m^2/s. At temperatures higher than 800°C, the $85.5LiF-14.5AlF_3$ system behaves as ordinary liquid. Our data obtained on the kinematic viscosity of molten binary $73LiF-27BeF_2$ and $85.5LiF-14.5AlF_3$ systems are in the satisfactory agreement with the experimental data obtained by the Oak Ridge National Laboratory (ORNL) for close compositions.

Experiment on viscosity measurement with molten $73LiF-27BeF_2$ (mole %) mixture has been carried out again after addition of 1 mole % of CeF_3 and later 2 and 4 mole % of ZrF_4 in the same crucible. The procedure was managed approximately in the same conditions, as in the first case. Note, that addition to molten $73LiF-27BeF_2$ (mole %) mixture of 1 mole % of CeF_3 and later 2 mole % of ZrF_4 leads to some decrease of melting temperature and kinematic viscosity.

Online control of the molten salt state

The technique was developed on the basis of the reflectance spectroscopy to control online the composition and ion state of molten salt fluorides (Ignatiev, 2016). The technique allows not only measuring the concentration of target products and impurities in melts but also getting information about the structure and chemical nature of soluble ion forms. It also can be used as an independent method in the fundamental research. The technique has been tried out when measuring and studying soluble forms of the praseodymium in 46.5LiF-11.5NaF-42KF eutectics in the temperature range from 500°C to 750°C. The experimental data on the solubility of praseodymium trifluoride in 46.5LiF-11.5NaF-42KF eutectics measured by methods of isothermal saturation and reflectance spectroscopy are given in Table 3.4.

US MSR-related activities

US MSR activities now include both liquid and solid fuel (i.e. FHR) MSR activities. US MSR activity continues to be directed through the US Department of Energy's Office of Nuclear Energy Office of Advanced Reactor Technology. Notably, during 2016, the United States began sponsoring development of the molten chloride fast reactor (MCFR) through sponsoring a project lead by Southern Nuclear Services (in partnership with TerraPower Inc., ORNL, EPRI, and Vanderbilt University) to develop in integral effects test facility in support of an MCFR test and eventually a commercial reactor. Much of the US-related MSR activities are co-ordinated through the Gateway for Accelerated Innovation in Nuclear Program (https://gain.inl.gov/SitePages/Home.aspx). Notable Gateway for Accelerated Innovation in Nuclear activities have included forming an MSR technical working group that include representation from multiple private MSR developers and interested utilities, sponsoring an MSR workshop at ORNL (https://public.ornl.gov/conferences/MSR2016), and providing technology vouchers to both Terrestrial Energy USA and TransAtomic Energy to enable them to access national laboratory resources for MSR development. The US Department of Energy's Office of Nuclear Energy has also initiated support of commercialisation of a liquid salt environment mechanical creep testing system, MSR neutronics modelling and simulation tools, and commercialisation of new nickel-based alloys for liquid salt service through its office of technology transitions. The United States has also begun early-phase evaluation of the technologies required for safeguarding liquid fuel MSRs.

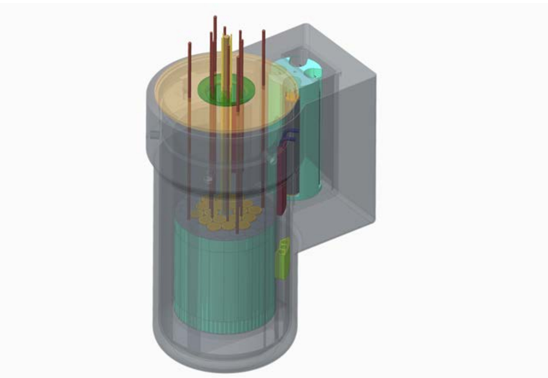
Table 3.4: **Experimental data on the solubility of praseodymium trifluoride in LiF-NaF-KF eutectics**

Temperature, °C	Solubility of PrF_3, mole %	
	Isothermal saturation	Reflectance spectroscopy
500	(8.87)	13.3±1.3
550	(13.35)	17.7±1.7
600	19.0±1.1	22.2±2.2
650	26.6±1.4	(26.7)
700	36.2±1.8	(31.2)
750	45.3±2.3	(35.6)

Note – in brackets are data, which were calculated and approximated by equations "solubility of praseodymium trifluoride vs. temperature".

Source: Ignatiev, 2016.

Figure 3.23: **Illustration of the FHR demonstration reactor configuration**



The US nuclear industry issued a number of reports related to MSR technology. The Nuclear Energy Institute released a strategic plan for advanced non-light water reactor development and commercialisation. The Nuclear Innovation Alliance released strategies for advanced reactor licensing, and the Electric Power Research Institute release a program on technology innovation: Scoping Study for an Owner-Operator Requirements Document (ORD) for Advanced Reactors. All of these industry planning documents prominently feature MSRs.

The US Nuclear Regulatory Commission (NRC) has also undertaken a joint initiative with the US Department of Energy on advanced reactor licensing that includes developing advanced reactor design criteria which would provide the minimum criteria necessary for licensing MSRs. The NRC has also recently published its Vision and Strategy for developing non-LWR regulation. Terrestrial Energy USA has notified the NRC that it intends to submit either a design certification or construction permit application for its integral molten salt reactor no later than October 2019 (www.nrc.gov/docs/ML1633/ML16336A508.pdf). Moreover, based upon positive evaluation of Terrestrial Energy USA's first phase loan guarantee application, Terrestrial Energy USA has been invited to submit part II of its application of a loan guarantee to support its initial deployment efforts.

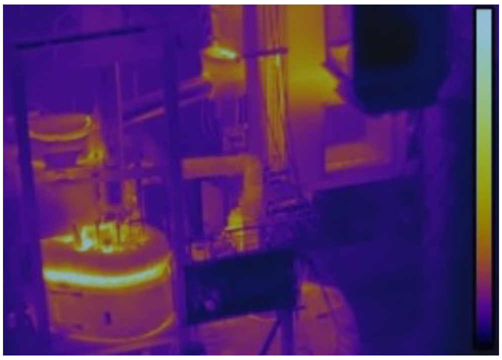
The United States has both national laboratory and university led FHR projects. The university projects are co-ordinated through the DOE Nuclear Energy University Program. The United States is currently sponsoring two large university FHR projects. The first is a collaboration between the Massachusetts Institute of Technology (MIT), the University of California at Berkeley, the University of Wisconsin and the University of New Mexico. The second collaboration is between the Georgia Institute of Technology, The Ohio State University, and Texas A&M University. The MIT lead project has included developing an advanced FHR reactor concept featuring a pebble-bed core with online refuelling coupled to an open-air Brayton power cycle and firebrick resistance-heated energy storage. The project also includes materials compatibility and irradiation testing.

A new DOE university award supports the development of a high-quality benchmark to benefit the MSR nuclear community. The effort is led by the University of California at Berkeley with support from ORNL and the Grenoble Institute of Technology. The purpose of the project is to examine the legacy data of ORNL's Molten Salt Reactor Experiment to develop a benchmark for the International Reactor Physics Benchmark Experiment Evaluation Project Handbook.

National laboratory-led efforts during 2016 have included developing an FHR demonstration reactor (FHR-DR) design concept (http://info.ornl.gov/sites/publications/Files/Pub61577.pdf), as well as initiating a project to experimentally evaluate tritium management technology options. The United States has also been co-operating with the Czech Republic to assess the sensitivity/uncertainty of molten salt element cross sections to MSR design evaluation efforts.

Also, ORNL and the Shanghai Institute of Applied Physics (SINAP) have previously 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. The most significant US development resulting from this collaboration in 2016 was completion of a FLiNaK liquid salt test loop, a thermal image of its operation is shown as Figure 3.24.

Figure 3.24: **Thermal image of the FLiNaK liquid salt test loop in operation**



Developing FHR industry consensus standards also is also continuing. American Nuclear Society standards on the design safety of both FHRs (ANS 20.1) and liquid-fuelled MSRs (ANS 20.2) are currently under development. Also, both American Society for Testing and Materials 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.

Additional information on MSR technologies is available on ORNL MSR web pages. www.ornl.gov/msr.

FHR and MSR in China

In the year of 2016, the Chinese Academy of Sciences' Thorium Molten Salt Reactor Program has significantly accelerated their development of liquid fuel molten salt reactors. The programme has developed a "three-step" roadmap of utilising thorium fuel in China. A modular liquid fuel molten salt demonstration reactor was designed to support the thorium roadmap. Research and development work, especially the physical and chemical property measurement of the fuel salt with different elemental mixing, has been initiated to form a basis for the design of reactors and the reprocessing techniques. The program was also evaluating the technologies that have been developed in the molten salt-cooled reactor development, in order to adapt those technologies in the liquid fuel environment.

The Thorium Molten Salt Reactor (TMSR) team has been trying to design and construct molten salt-cooled and thorium-fuelled molten salt test reactors, plus a simulator of the molten salt-cooled reactor. Major efforts consist of identifying candidate site for the test reactors, completing engineering design, applying for licence from the Chinese government, and developing and obtaining components and materials.

The TMSR team has reached an agreement with the State Power Investment Corporation (SPIC), one of the three major nuclear power developers and operators in China, identifying Haiyang in Shandong province as the candidate site for the molten salt test reactors. SPIC owns the Haiyang site and two AP1000 units are currently under construction at the site. The joint design team of TMSR and Nuclear Power Institute (NPI) has completed the preliminary engineering design of the nuclear island, except for the instrument and control and cover gas systems, of the 10 MWth molten salt-cooled test reactor (TMSR-SF1). The TMSR team and Shanghai Nuclear Engineering Research and Design Institute (SNERDI) have signed a contract to create a joint design team working on the preliminary engineering design of the instrument and control and cover gas systems of the TMSR-SF1. The TMSR design team has completed the engineering design of the TMSR-SF0, which is an electric powered simulator of the TMSR-SF1. The TMSR-SF0 will be used to conduct experiments to validate codes and to verify the performance of the system. The TMSR design team has also completed the design of a FLiNaK test production unit, which is capable of producing ten metric tons of FLiNaK salt per year once constructed. The salt can be used for TMSR-SF0 and other experiments.

In the second half of 2016, the TMSR team has been collecting information of the Haiyang site and preparing for the environmental impact assessment of building the molten salt test reactors at Haiyang. The TMSR and SNERDI joint team plans to complete the preliminary engineering design of the instrument and control and cover gas systems of the TMSR-SF1 by early 2017. By the end of 2016, the TMSR licensing team has completed a draft preliminary safety analysis report of the TMSR-SF1 based on the current preliminary engineering design and prepare for the application of the construction permit. The TMSR design team plans to release conceptual designs of a 2 MWth thorium-fuelled molten salt test reactor (TMSR-LF1) and a small modular thorium-fuelled molten salt demonstration reactor (TMSR-LF2) in the first half of 2017.

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3.4. Supercritical water-cooled reactor (SCWR)

Main characteristics of the system

The supercritical water-cooled reactor (SCWR) is a high-temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point (374°C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: pressure vessel concepts proposed first by Japan and more recently by a Euratom partnership, and 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 pressures 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 three Project Management Boards (PMBs) within the SCWR System: System Integration and Assessment (provisional), Materials and Chemistry, and Thermal-hydraulics and Safety. The Fuel Qualification Testing (provisional) PMB, which was identified in previous reports, has been included as a component in the System Integration and Assessment (provisional) PMB currently being managed by the System Steering Committee. Table 3.5 lists the members and shows the status of these PMBs.

Table 3.5: **Status and memberships of SCWR System Arrangement and Project Arrangements**

SCWR System Arrangement and Project Arrangements	Signatories	Date of signature
System Arrangement	Canada, Euratom, Japan, Russia China	November 2006 (renewed 2016 by Canada and Japan) July 2011 (renewed 2016) July 2014 (renewed 2016)
Thermal-Hydraulics and Safety Project Arrangement	Canada, Euratom, Japan	October 2009 (Japan withdrawn 2016)
Material and Chemistry Project Arrangement	Canada, Euratom, Japan	December 2010 (Japan withdrawn 2016)
System Integration and Assessment Provisional Project Arrangement	Managed by the System Steering Committee	-

Canada, China and Euratom are in the process of signing the extension of the Project Arrangements for Thermal-Hydraulics and Safety as well as the Materials and Chemistry. Regretfully, Japan decided not to sign the extension at this point. Russia has expressed interest to participate in the Thermal-Hydraulics and Safety Project and is undergoing their internal process before deciding to sign the Project extension.

Prior to the inclusion of the fuel qualification testing (provisional) PMB into the System Integration and Assessment (provisional) PMB, Canada and Euratom are collaborating informally to pursue in-reactor irradiation of SCWR fuels at supercritical pressures in the Řež research reactor in Czech Republic. China has also expressed interest to participate in the second phase of testing.

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. An important collaborative R&D project is to design and construct an in-reactor fuel test loop to qualify the reference fuel design. As a SCWR has never been operated before, such generic testing is considered to be mandatory before a prototype reactor can be licensed.
- Thermal-hydraulics and safety – Gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed for validating thermal-hydraulic codes. The design-basis accidents for a SCWR have some similarities with conventional water reactors, but the difference in thermal-hydraulic behaviour and large changes in fluid properties around the critical point compared to water at lower temperatures and pressures need to be better understood.
- Materials and chemistry – Qualification of key materials for use in in-core and out-core components of both pressure tube and pressure vessel designs. Selection of a reference water chemistry which minimises materials degradation and corrosion product transport will also be sought based on materials compatibility and an understanding of water radiolysis.

Main activities and outcomes

Significant R&D achievements have been accomplished in the three PMBs through strong collaboration between participants. In addition to the key institutes responsible for developing the SCWR concepts, academia and partner institutes have contributed to the success. Furthermore, a number of highly qualified personnel have been trained, benefiting nuclear and non-nuclear industries.

System integration and assessment

The System Integration and Assessment Provisional PMB covers three main activities:

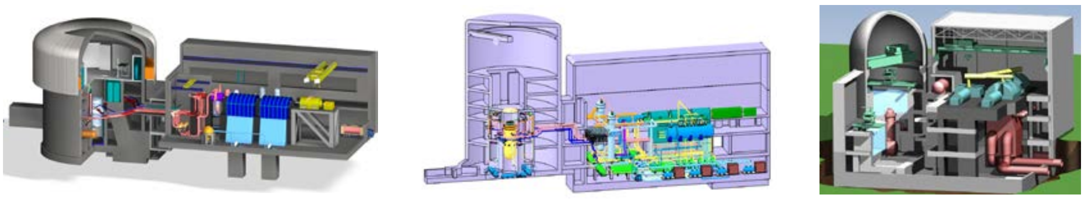
- review and assessment of SCWR concepts;
- fuel qualification testing;
- SCWR physics.

Canada, Euratom and Japan have successfully completed the development of their SCWR concepts, which were reviewed by international peers. Figure 3.25 shows the proposed SCWR plant concepts. The core of Canada's concept is the pressure tube type, while those of Euratom's and Japan's concepts are the pressure vessel type. Both Canada's and Euratom's core concepts are developed for the thermal spectrum. On the other hand, Japan has developed two core concepts with one at the thermal spectrum and another at the fast spectrum (both having the same plant concept).

China and Russia are continuing their development. Both core concepts are the pressure vessel type. However, China is developing a thermal spectrum core concept while Russia is working on a fast spectrum core concept. China is planning to host a review meeting of their SCWR concept with international peers in 2018.

Figure 3.25: **SCWR plant concepts**

Canada Euratom Japan

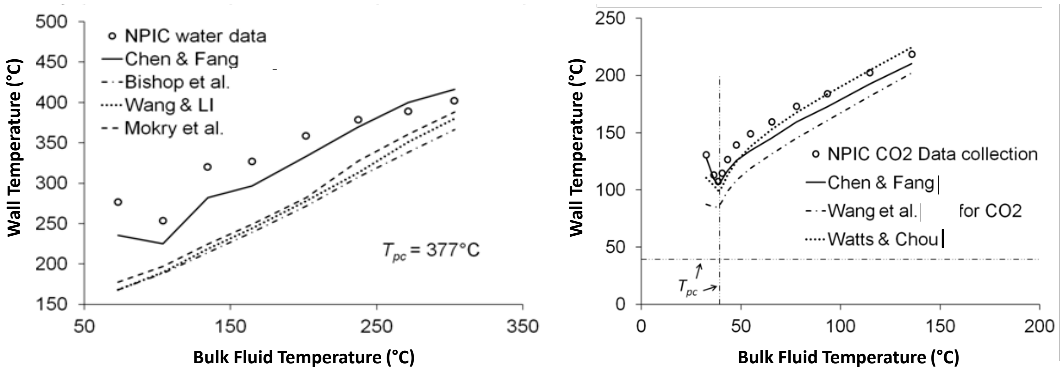


All proposed SCWR core concepts (other than Japan's super-fast reactor) have been developed to generate electric powers at or greater than 1 000 MW, which are considered excessive for small remote communities. The flexibility of adjusting the SCWR core size for meeting local deployment needs has prompted the interest in developing small SCWR concepts ranging from 10 to 300 MW in electric power. A project plan is being established to pursue the development.

Thermal-hydraulics and safety

The supercritical heat transfer database has been expanded to include experimental data of Nuclear Power Institute of China (NPIC), which were obtained with water or carbon dioxide flow in tubes. Including these data, the database contains over 26 000 data points for water flow and close to 20 000 data points for carbon dioxide flow in tubes. It has been applied in assessing the prediction accuracy of 18 heat transfer correlations (Figure 3.26). Overall, the correlation of Chen and Fang provides the best prediction accuracy for both water and carbon dioxide flows. Figure 3.27 compares predicted wall temperatures of various correlations against experimental data of NPIC, which were obtained with water flow in a 6-mm ID tube and carbon dioxide flow in an 8-mm ID tube. Similar trends were observed for carbon dioxide data. The correlation of Chen and Fang agrees closely with the experimental data.

Figure 3.26: **Comparisons of predictions against heat transfer data for tubes**



CNL collaborated with Xi'an Jiaotong University in investigating the heat transfer characteristics in a 2x2 rod bundle with and without the wire-wrapped spacers. Non-uniform circumferential wall temperature distributions were observed around the uniformly heated rods. Peak temperatures were measured at the narrow-gap region between the rod and the unheated ceramic square enclosure and minimum temperatures at the central sub-channel between all four heated rods. Differences between peak and minimum wall temperatures depend on the flow conditions. Heat transfer enhancement was observed for the bundle equipped with the wire-wrapped spacers. It is, however, relatively small.

The effect of spacer configuration on heat transfer has been examined in a three-rod bundle cooled with carbon dioxide at supercritical pressures. Wire-wrapped or grid spacers were installed onto the heated rods. Figure 3.28 illustrates axial wall temperature distributions along one heated rod. Overall, there are not much differences in wall temperatures between wire-wrapped and grid spacers. However, the variation in wall temperature at various circumferential positions of the heat rod is smaller for the wire-wrapped spacer than the grid spacers at the subcritical temperature region. This may be attributed to the improved mixing between subchannels of the wire-wrapped spacer reducing the flow and enthalpy imbalances. Localised heat transfer enhancement has been observed at the grid spacer locations but decayed over a short distance from the spacer.

A reduction in wall temperature has been observed at the inlet region of the bundle with the wire-wrapped or grid spacer. This is attributed to the development of the thermal-boundary layer. Similar trend was shown for the 8-mm tube at the same flow conditions. The region of wall temperature reduction appears to be shorter for the wire-wrapped spacer than the grid spacer. This could be attributed to the improved mixing that enhances the development of the thermal-boundary layer.

A series of heat transfer experiments were conducted with supercritical water in a vertical upward bare tube to i) provide relevant experimental data; ii) obtain a better understanding of heat transfer deterioration and the criterion for the onset of heat transfer deterioration; and iii) provide guidance for the system design and safety operation. The test section was constructed with a 1Cr18Ni9Ti tube of 25-mm outer diameter and a wall thickness of 2.5 mm. It was heated over a length of 6 metres with joule heating. The heated section was preceded with a 1-m long unheated section for flow development. The NiCr-NiSi thermocouples were inserted at the inlet and the outlet of the test section to measure the bulk fluid temperatures. K-type thermocouples were attached along the test section for measuring the outer-wall temperatures.

Figure 3.27: **Comparison of heat transfer characteristics between (a) wire-wrapped spacers and (b) grid spacers**

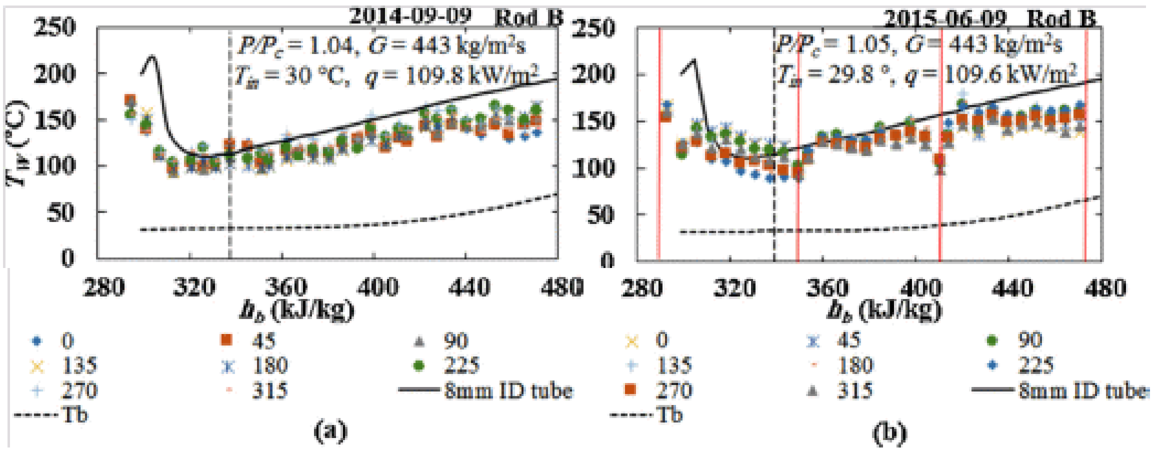
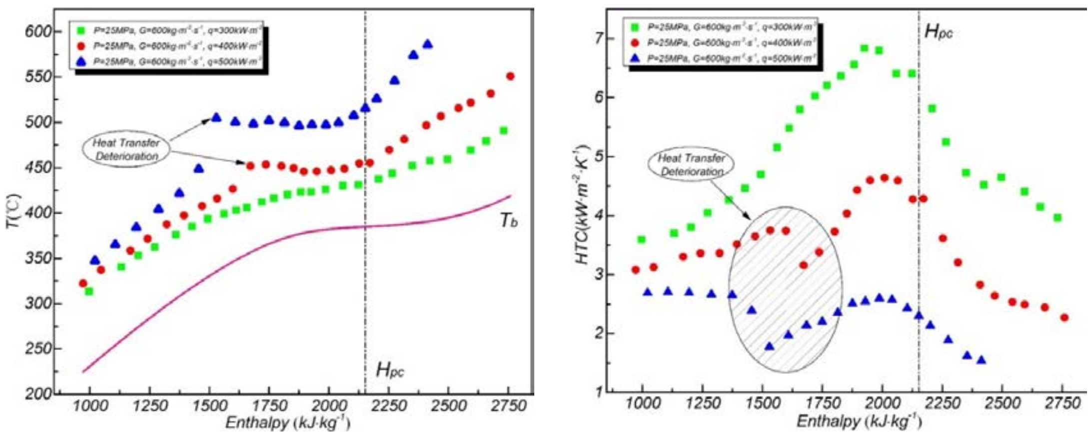


Figure 3.29 illustrates variations of the experimental inner-wall temperature (T), and corresponding heat transfer coefficient, as functions of bulk fluid enthalpy and heat flux (q) at the pressure (P) of 25 MPa and mass flux (G) of 600 kg·m-2·s-1. The inner-wall temperature increases gradually with increasing fluid enthalpy along the tube at the heat flux of 300 kW·m-2. The corresponding heat transfer coefficient has a peak value before the pseudo-critical point. At heat fluxes of 400 and 500 kW·m-2, the wall temperatures exhibit peaks values at enthalpies of about 1 625 kJ·kg-2, and the corresponding heat transfer coefficients reach localised minimum values signifying the occurrence of heat transfer deterioration. The peak wall temperature increases with increasing heat flux.

Figure 3.28: **Variations of wall temperature and heat transfer coefficient as function of enthalpy and heat flux**



The instability of supercritical water flow inside two parallel channels was investigated experimentally. Two types of instabilities with various oscillation periods occurred at different power regions during the heating process, indicating the existence of two types of dynamic instabilities. The Type-I instability occurred when the outlet temperature of the test section surpassed the pseudo-critical temperature. A sudden increase in pressure drop caused a big disturbance to the system which resulted in a periodic feedback between the pressure drop and the mass flow rate. Large oscillations were observed in the entire system at low heating powers during which the two parallel channels oscillated in phase with the whole system over a long oscillation period (20-300 s). The type-II instability occurred only in the heated section at high powers at which the two parallel channels oscillated 180 degrees out of phase while the rest of the system was stable. The period of Type-II instability is relatively small. Flow maldistribution occurred during experiments. Small differences were generated during the fabrication of two subsonic-sonic venturi nozzles. This asymmetry (K=8.76 and K=6.52) between the two venturi nozzles was not apparent at the initial stage of heating, but became large with increasing heating power (Figure 3.30).

Experiments on heat transfer characteristics of supercritical R134a in a tube have been conducted with upward and downward flows. The test section is a circular tube of 10-mm inner diameter. In addition, heat transfer experiments simulating pressure transients, which may be encountered during start-up, shutdown and postulated accident scenarios of SCWR, have been carried out Figures 3.29 and 3.30 compares experimental heat transfer coefficient as a function of fluid enthalpy (h) in upward and downward flows at steady state. The heat transfer coefficient increases with enthalpy and reaches a peak before the pseudo-critical point. It decreases with increasing enthalpy beyond the pseudo-critical point. Heat transfer is more effective for downward than upward flow at the mass flux of 400 kg/m²s. However, the effect of flow direction appears to be small at the mass flux of 1 000 kg/m²s.

Figure 3.29: **Transition region and Type-II instability region**

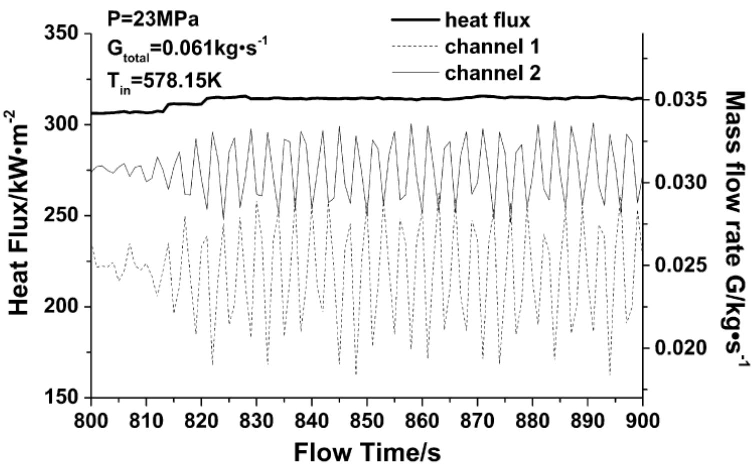


Figure 3.30: **Effect of flow direction on heat transfer at supercritical pressures**

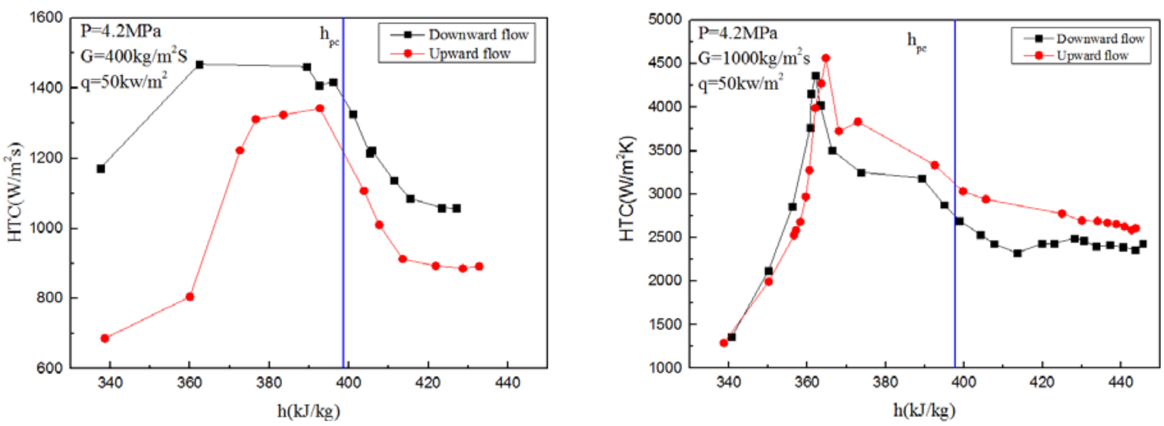
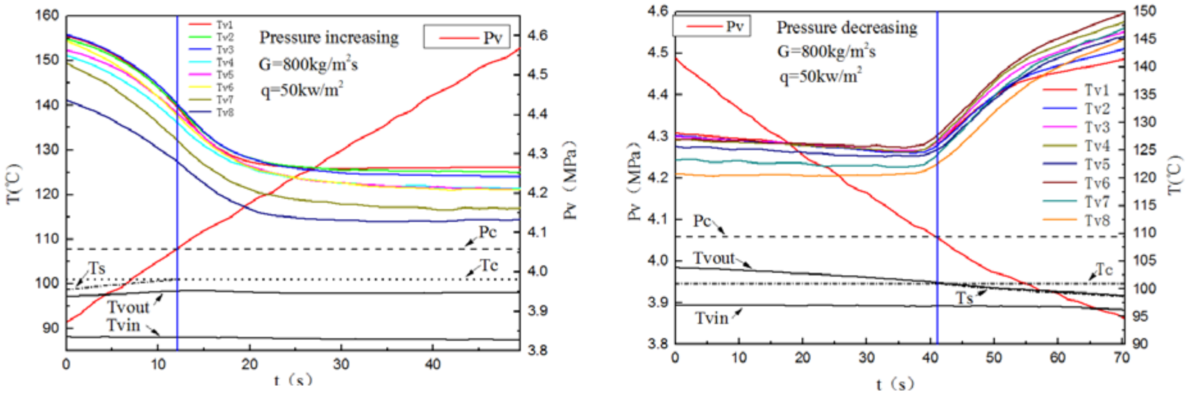


Figure 3.31: **Wall temperature responses during pressure transients**



Heat transfer experiments were carried out with upward refrigerant R-134a flow inside the vertical 10-mm tube under pressure transients. Figure 3.32 illustrates the wall temperature responses to increasing and decreasing pressure transients. Wall temperatures decrease with increasing pressure. The decrease is much more rapid as the pressure approaches the critical pressure and become gradual after passing the critical pressure. On the other hand, wall temperatures decrease with decreasing pressure but increase rapidly as the pressure crosses the critical point.

While supercritical water is a perfect coolant with excellent heat transfer, a temporary decrease of the system pressure to subcritical conditions, either during intended transients or by accident, can easily cause a boiling crisis with significantly higher cladding temperatures of the fuel assemblies. Such situation is planned to be tested with a small fuel assembly of four rods, to be operated in a critical arrangement with supercritical water inside a research reactor in the Czech Republic. First out-of-pile tests of this experiment have recently been performed in the Supercritical Water Multipurpose Loop (SWAMUP) facility at Shanghai Jiao Tong University in China.

Some of the transient tests have been simulated at KIT with a one-dimensional MATLAB code, assuming quasi-steady state flow conditions, but time-dependent temperatures in the fuel rods. The new method reproduced well the boiling crisis during depressurisation from supercritical to subcritical pressure, including rewetting of the hot zone within some minutes, as shown in Figure 3.33, but the peak temperature was somewhat under-predicted. Tests with a lower heat flux, which did not cause such phenomena, could be predicted as well. In another test with increasing pressure, however, a boiling crisis was also observed at a heat flux, which was significantly lower than the predicted critical heat flux.

Experimental studies on critical heat flux (CHF) were carried out at the KIMOF test facility which is a closed loop with forced circulation of R134a. The test section consisted of a stainless steel tube of 3 015 mm long with inner and outer diameters of 10 and 12 mm, respectively. It was installed vertically with an upward flow of R134a. To ensure the full development of the turbulent flow the heating length of 2 495 mm started after 500 mm (i.e. 50 times the tube diameter). The tube outer-wall temperatures were measured using 35 thermocouples. Experiments of CHF were conducted at pressures of 2.8, 3.3, 3.8 and 4 MPa, mass fluxes of 500-4 000 kg/m²s and inlet temperatures of 23-81°C. The effect of various parameters on CHF was investigated. Figure 3.33 shows the effect of pressure and mass flux on CHF, at a constant inlet sub-cooling of 30 K.

Figure 3.32: **Transient wall temperatures of a test fuel assembly predicted for a depressurisation from 25 to 17 MPa with 1 MPa/min at a heat flux of 640 kW/m²**

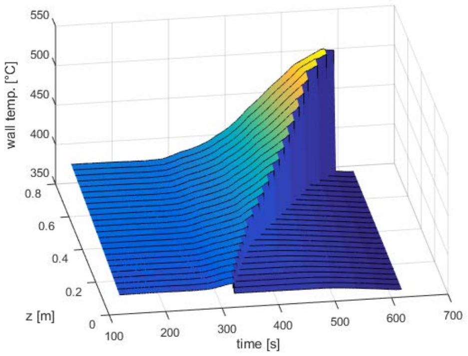
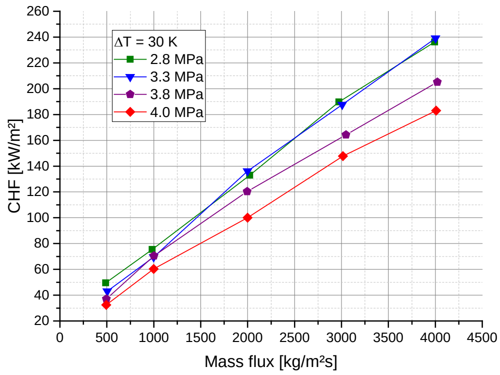
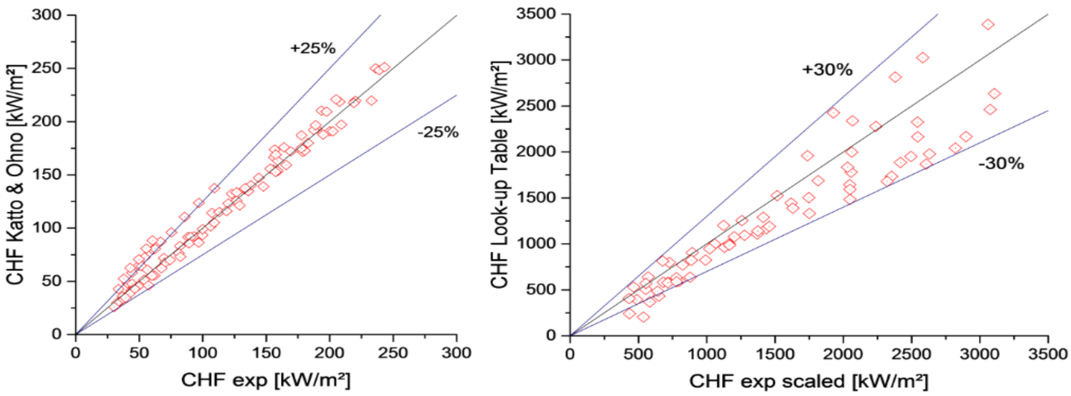


Figure 3.33: **Influence of mass flux and pressure on CHF at 30 K sub-cooling**



The experimental results were compared with the correlation of Katto & Ohno. It was found that the correlation can predict the present experimental results well except for low mass fluxes. The mean relative deviation of all data is 10.3% and the standard deviation is 11.6%. Moreover, scaling laws were applied to convert the experimental results to water equivalent conditions, which were then compared with the standard CHF look-up table. Figure 3.34 compares experimental CHF values to predictions of the Katto & Ohno correlation and the CHF look-up table. Good agreement has been observed.

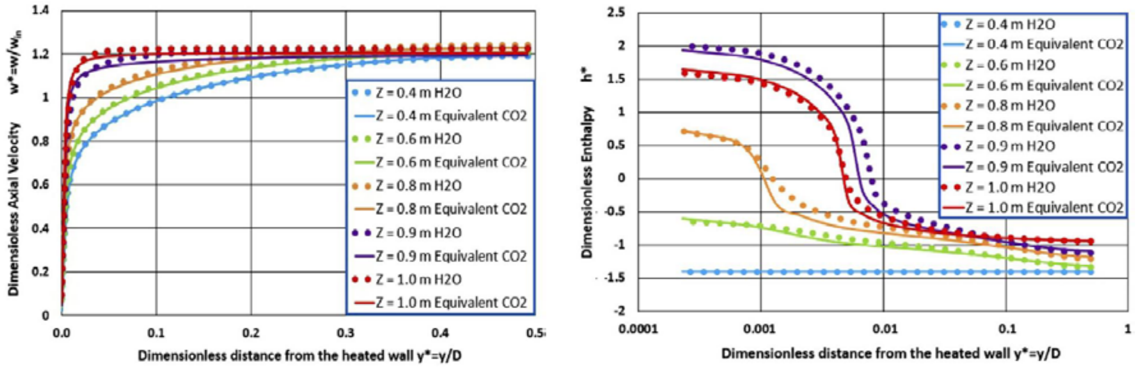
Figure 3.34: **Comparison of experimental CHF values to predictions**



CFD simulation of heat transfer at supercritical pressures was conducted at different institutions to assess the feasibility of various turbulence models for their application to supercritical fluids, investigate the mechanistic behaviour of heat transfer, to enhance the physical understanding and to provide information for the development of new heat transfer correlations, and study the similarity between different fluids.

Figure 3.35 shows the distribution of axial velocity and enthalpy for supercritical water and CO_2 flowing in a vertical oriented tube. The thermal-hydraulic parameters in both fluids are determined with the fluid-to-fluid scaling method. As recognised, good agreement is achieved in the velocity distribution as well as in the enthalpy distribution in both fluids. This indicates indirectly the feasibility of the fluid-to-fluid scaling method applied.

Figure 3.35: **Axial velocity (left) and enthalpy (right) distribution for supercritical water and CO_2**



In CVR, the verification of the experimental code ATHLET 3.1A (Analysis of Thermal-hydraulics of Leaks and Transients) was performed. The main source of data was IAEA´s CRP results summarised in report "Heat Transfer Behavior and Thermohydraulics Code Testing for Supercritical Water Cooled Reactors (SCWRs)" (IAEA, 2014). Two cases have been modelled and the obtained results have been confronted with experimental data showing the capability of the code to follow the selected heat transfer using the selected correlations. For assessing the heat transfer behaviour for SCW, the code has implemented six correlations: Watts–Chou, Mokry et al., Gupta, Cheng et al. (acceleration parameter), Cheng et al. (buoyancy parameter), Yang (normal heat transfer), and Yang (deteriorated heat transfer).

The first test benchmark: Code Test Benchmarking 1: "Steady state Flow in a Heated Pipe", aimed at assessing codes capabilities in prediction of heat transfer against experimental data. The case itself is divided in case 1 and case 2.

The first reference test case is represented by a steady state flow of water at supercritical conditions that runs upward in a heated pipe. The measurements for the heat transfer (wall temperature) were collected from a vertical pipe with uniform heating along the axial direction. The second case uses the same modelling approach as for the case 1, however the exercise also captured data points where deteriorated heat transfer (DHT) was observed in the upward direction. The experiment was therefore more difficult for the participants to evaluate. The cases were subdivided in variant 1 and 2. Variant 1 is of a high interest, the parameters of the experimental conditions were similar to the ones proposed for SCWR, and in particular its results can be utilised for thermal-hydraulic prediction for core parameters. Variant 2 incorporated a more defined DHT zone in the upward direction compared to downward direction. Variant 2 was mostly prepared for analysing the flow direction effect on heat transfer in a vertical tube, for example, in the case of steam generators. The comparison of the experimental data and ATHLET correlations are on Figures 3.36 to 3.41.

ATHLET 3.1A has proved to have good capabilities of assessing heat transfer in SCW. Codes that can calculate SCW conditions are still to some extent limited. Their feedback showed that the code has a great potential in the supercritical field. Based on these studies, GRS kept on improving the SCW module and adding the state-of-the-art correlations. The newest version, 3.1A, assessed in this report, showed to capture even DHT zone. DHT still remains a complex phenomenon and more studies and experiments need to be performed for a better evaluation. Generally, in the case 1, Watts-Chou showed a good agreement with the experimental data the first 2 m of the pipe length. In the last 2 m, Gupta has an excellent agreement with the experimental results. In the case 2, Gupta is over predicting the wall temperature and the heat transfer coefficient for all Variants. Watts-Chou and Mokry have a good agreement for the Case 2 Variant 1, although both the correlations they were not able to simulate the DHT peak showed in the Case 2 Variant 2 upward flow. Only Gupta was able to simulate the position of the DHT peak. In the Case 2 Variant 2 downward flow, only Watts-Chou was able to simulate temperature. In general, the results obtained in this exercise were similar with the benchmark participants in Feuerstein et al. (2016).

Figure 3.36: **Bulk temperature distribution**

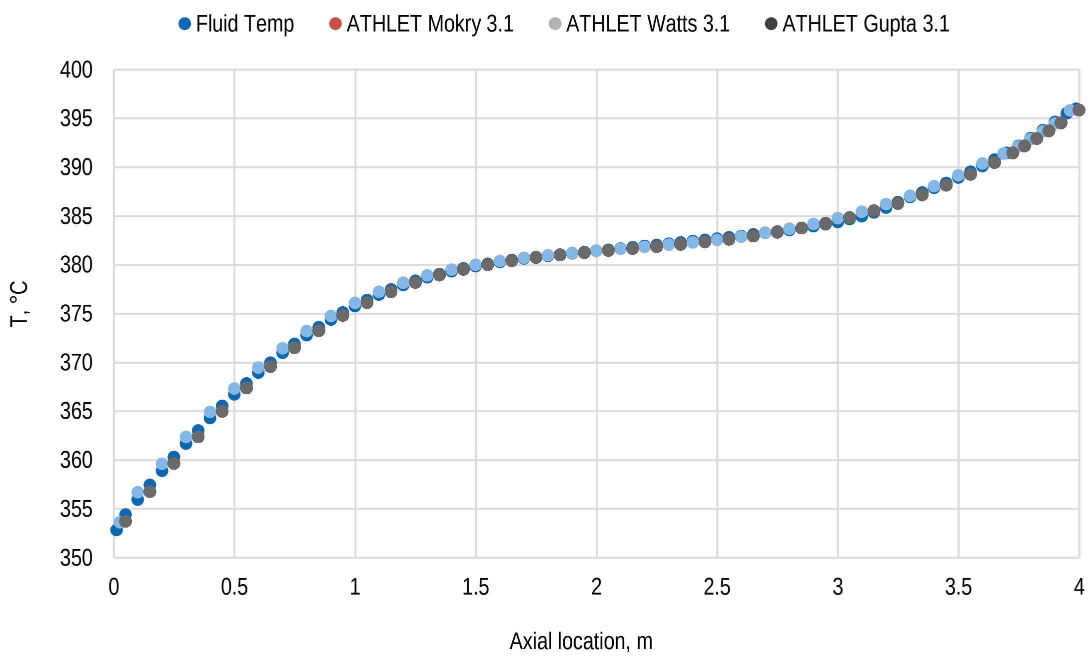


Figure 3.37: **Heat transfer coefficient Case 1**

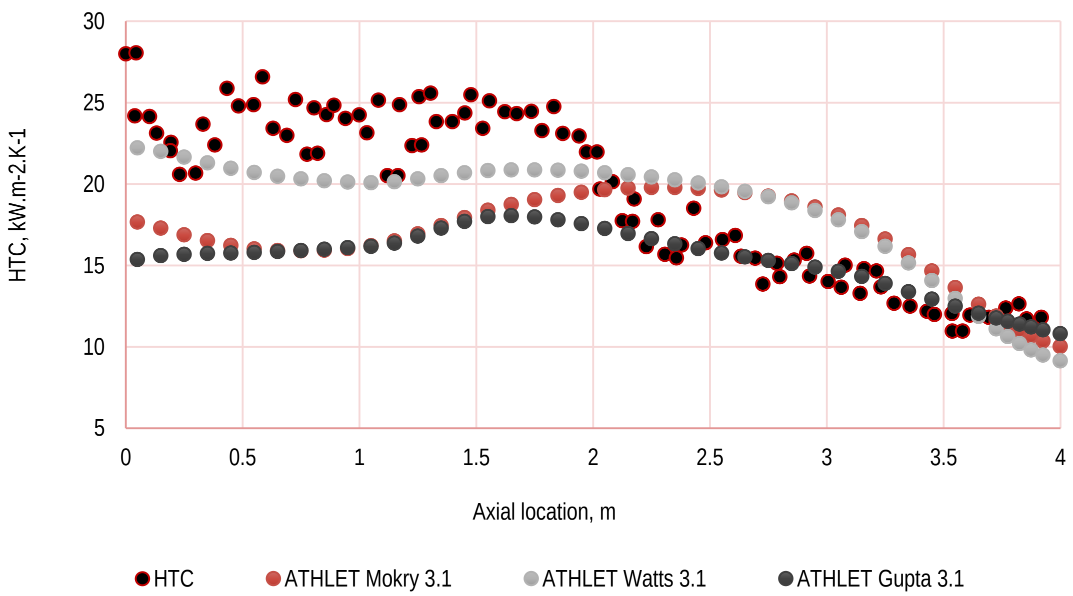


Figure 3.38: **Bulk temperature distribution Case 1**

a) upward flow b) downward flow

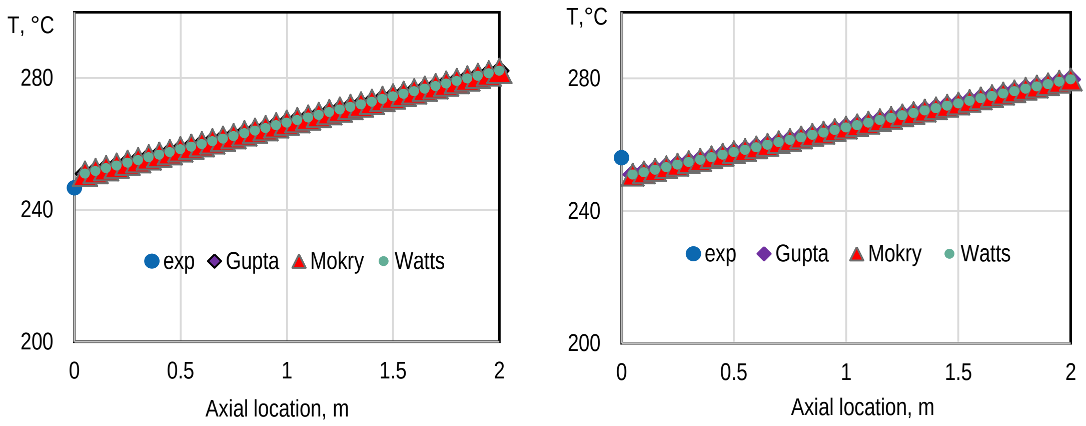


Figure 3.39: **Wall temperature distribution for Case 2 Variant 1**

a) upward flow b) downward flow

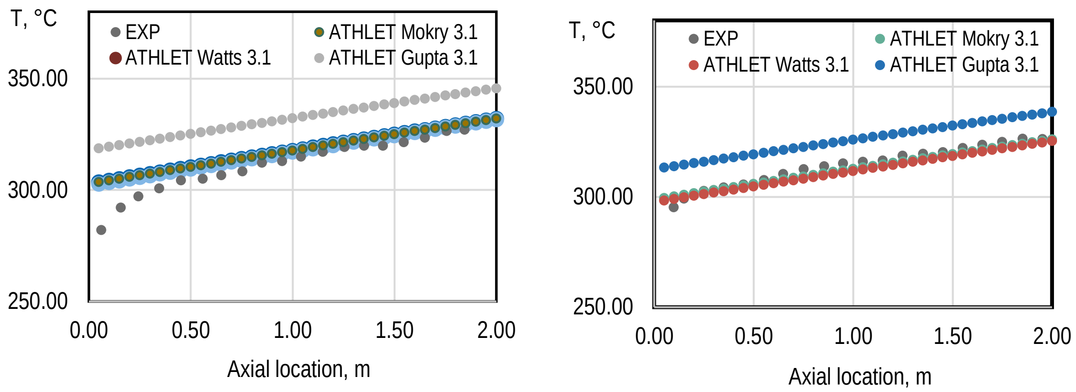


Figure 3.40: **Bulk temperature distribution**

a) upward flow b) downward flow

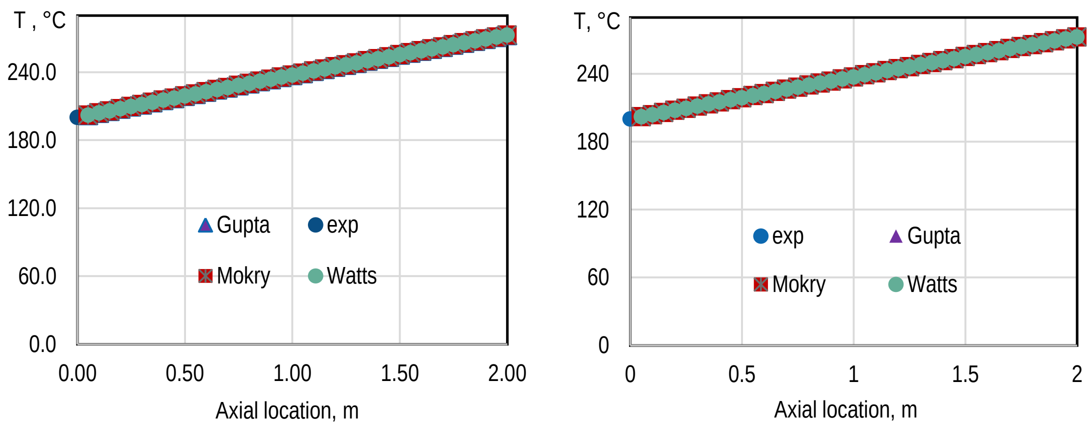
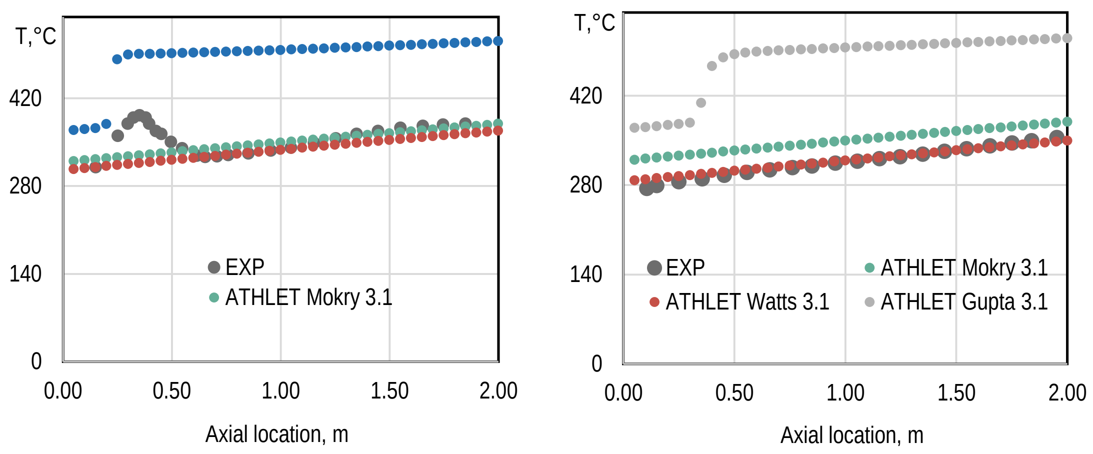


Figure 3.41: **Wall temperature distribution Case 2 Variant 2**

a) upward flow | b) downward flow



Materials and chemistry

In 2016, the M&C PMB has continued working on evaluation of candidate alloys for all key components in the SCWR designs. This includes general corrosion and stress corrosion cracking tests in autoclaves connected with water recirculation loops, as well as development work on state-of-the-art test facilities and measuring equipment. In addition to this, the ultimate goal has been to promote activities towards in-pile tests both in Europe and China in the near future. Supporting actions on water chemistry studies and modelling efforts has been performed as well to better understand the component's anticipated service environment.

The successful round-robin exercise on corrosion was finished in 2015 and results were published in peer-reviewed journal paper. In 2016, the M&C PMB initiated the follow-up campaign on round-robin corrosion exercise between all partners (Canada, China and Euratom). The idea in this follow-up campaign is to compare the corrosion test results by using more detailed test parameters than in the previous one in order to further diminish differences in test results between laboratories. Preparation of a first round-robin exercise on stress corrosion cracking testing was also initiated in 2016 by the partners and is anticipated to start in the second half of 2017.

One significant knowledge in regard to the stress corrosion cracking (SCC) behaviour of austenitic stainless advanced in 2016 was the finding of "threshold strain" for SCC in supercritical water. The initial SCC tests at CANMETMaterials in SCW of 500°C found no crack initiations when the alloys (310s, 800H and 347) were strained to 5%. However, the alloys did show SCC when strained to failure. This result suggested that there was a threshold strain (which is somewhere greater than 5% strain) for SCC, at least in slow strain rate testing (SSRT). This finding was confirmed at JRC (Netherlands) in their tests in SCW of 650°C, in which no SCC was found even with more than 300 hour static loading period added to the 5% SSRT test period. A new SCC round-robin test programme is proposed (to be led by CANMETMaterials) that will use tapered test samples to define and compare the threshold SCC conditions of different alloys.

Corrosion resistance of austenitic stainless steels (SS) 310, 304, and Ni and Fe based A286 exposed to various pressures of 0.1 MPa, 8 MPa, and 29 MPa at 625°C for 1 000 hours has been investigated. Extensive microscopy investigations revealed a single-layer oxide formed at 0.1 MPa and dual-layer oxides at 8 MPa and 29 MPa, followed by a Cr depleted region into the austenite substrate. The compositions of the inner oxides at 8 MPa and 29 MPa are Cr rich and largely similar to those of the single-layer oxides at 0.1 MPa exposures. This similarity suggests that corrosion testing in superheated steam will give results in qualitative agreement with those expected at 25 MPa (SCWR operating pressure), and confirms that superheated steam at 0.1 MPa is a suitable surrogate for SCW corrosion testing.

Two superalloys, A286 and Alloy 625, were tested in supercritical, high-pressure, and low-pressure steam at 625 °C for 1 000 h; each represents a pressure of 29 MPa (SCW), 8 MPa (high-pressure steam), and finally 0.1 MPa (low-pressure steam). Results from this study show a higher oxidation rate, in terms of weight gain as shown in Figure 3.42, for A286 under all conditions primarily due to its low Cr content. Alloy 625, on the other hand, exhibits much more oxidation resistance under all conditions due to the formation of protective Cr-containing surface oxide(s). Weight changes of A286 and Alloy 625 in supercritical water and low-pressure steam are comparable, while high-pressure steam exposure leads to considerable weight gain of A286 samples and weight loss and pitting on alloy 625.

FeCrAlY and NiCrAl coating samples, were tested in superheated steam (SHS) at 800°C for up to 600 hours. The FeCrAlY was covered with surface scale more rapidly than that on NiCrAl. Figure 3.43 shows that the weight change results also suggest that more oxide formation took place on FeCrAlY than NiCrAl. For FeCrAlY, grain boundary oxide (Al_2O_3) formed rapidly upon exposure to SHS for 300 hours. Further exposure caused more intragranular Al_2O_3 to form, in addition to magnetite formation on the grain boundaries. For NiCrAl, NiO seems to have formed initially upon SHS exposure due to the high Ni content in the alloy. Spinel and $(Cr,Al)_2O_3$ also formed after 300 hours with limited amount of Al_2O_3. After 600 hours Al_2O_3 became well developed and the coverage of spinel and Cr_2O_3 on the surface reduces.

Figure 3.42: **Weight change vs. test condition after 1 000 h**

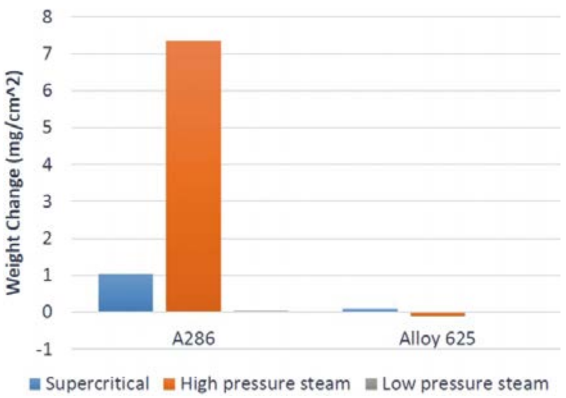
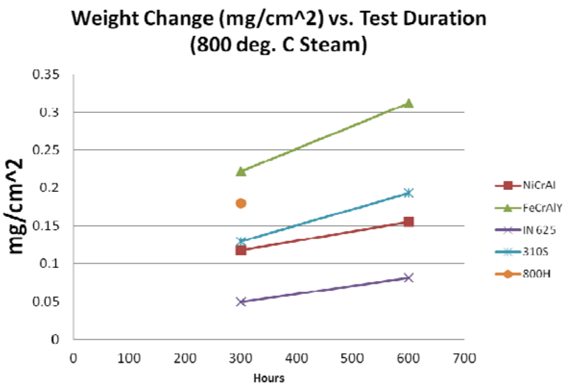
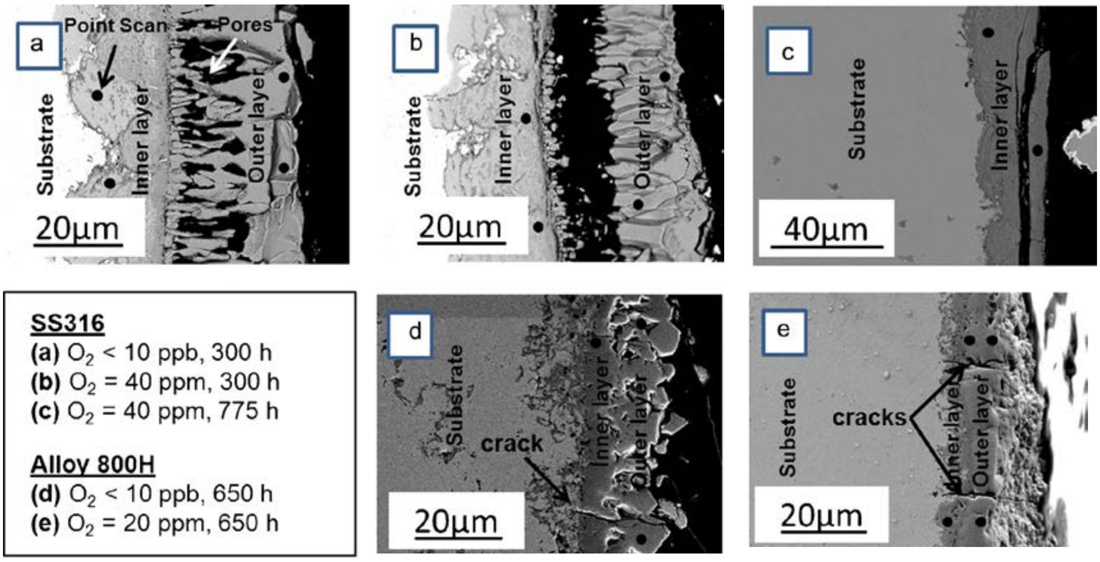


Figure 3.43: **Weight change as a function of exposure time in SHS**



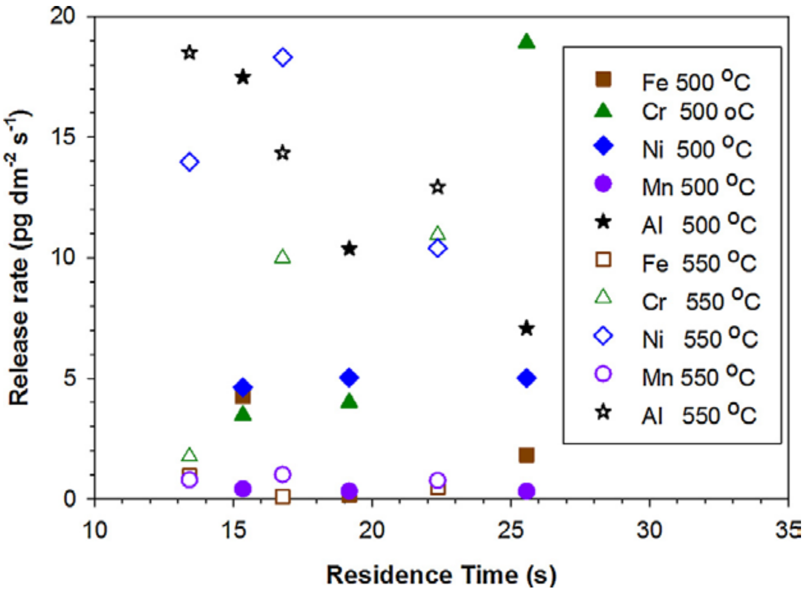
An investigation into the morphology of oxide layers formed on stainless steel SS316 and Alloy 800H in continuous supercritical water flow conditions was performed. Figure 3.44 shows a cross-sectional view of the oxide layers formed on the SS316 and alloy 800H tubes in SCW (650°C and 25 MPa). Duplex oxide layer structures were observed. In deoxygenated water, an outer magnetite layer is formed on both materials, with an inner layer on alloy 800H containing chromium oxide. In oxygenated water, only magnetite is seen on SS316, while both hematite and magnetite are observed on alloy 800H. Alloy 800H forms a thinner scale, in which the presence of hematite is thought to provide better corrosion protection. Numerous cracks are formed in the oxide layers after oxygenated treatment.

Figure 3.44: **Cross-sectional view of the oxide layers formed on the SS316 and Alloy 800H tubes in SCW (650°C and 25 MPa)**



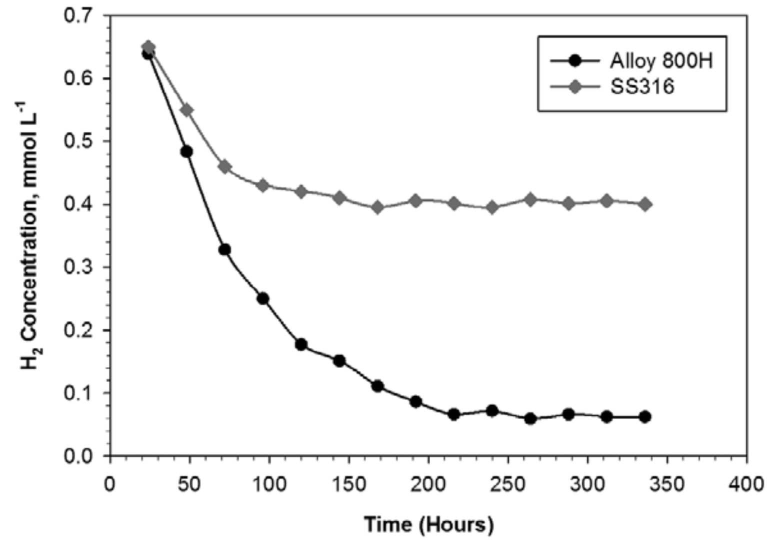
The experimental set-up was used to monitor concentrations of dissolved metals, oxygen and hydrogen at the exit of an alloy 800H tube exposed to oxygenated supercritical water to study the initial stages of oxidation (Choudhry et al., 2016). Initially, both water and oxygen are oxidants, generating H_2, releasing low concentrations of Fe, Al, Ni and Mn. All added oxygen is consumed during this period (Figure 3.45). No Cr release is observed during these initial stages, attributed to Mn incorporation in the oxide. Rapid reappearance of oxygen in the water is attributed to thickening of the surface oxide. Manganese dissolution results in Cr release and formation of an outer iron oxide layer.

Figure 3.45: **Release rates for Fe, Ni, Cr, Mn and Al as a function of residence time in the test section**



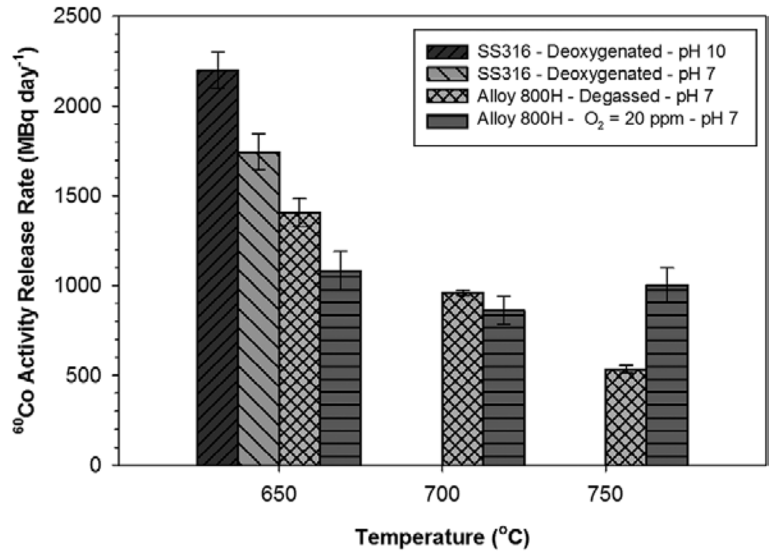
Hydrogen evolution and activity release rates for stainless steel SS316 and nickel-based alloy 800H at the SCWR conditions were estimated, together with amount of oxygen absorbed into the passivation oxide layers formed on both materials. Hydrogen evolution from an oxidised SS316 surface was found to be about six times larger than from alloy 800H at constant temperature and pressure of 650°C and 25MPa, shown in Figure 3.46. The largest 60Co activity release rate due to corrosion was estimated to be for SS316 at 650°C and at pH10, shown in Figure 3.47. Because of a single loop design, the radioactive corrosion products of in-core surfaces of the SCWR are expected to be carried out-of-core with the supercritical coolant.

Figure 3.46: **Hydrogen evolution in the reactor versus exposure time**



Note: Temperature, pressure and at-pump volumetric flow rate were held constant at 650°C, 25 MPa and 0.1 mL/min.

Figure 3.47: **Activity release rate of 60Co in MBq/day at a constant pressure of 25 MPa and a volumetric flow rate of 0.1 mL/min**



The SCC test was conducted on a 12.7 mm thickness compact tension (CT) specimen with 5% side grooves. The SCC test was performed in a 4 L, alloy 625 autoclave system equipped with a servo tensile machine, a reversed DCPD (direct current potential drop) crack length measurement system and a recirculating water loop. The CT specimen was loaded and immersed in high purity water at the pressure of 25 MPa. Pt wires were spot welded to the CT specimen to apply current and measure DC potentials at CT specimen notch opening mouth.

Before the SCC test in SCW, a fatigue pre-crack was extended by ~1 mm in room temperature air at a Kmax = 25 MPa√m, load ratio R = 0.3, and frequency f = 1 Hz. The pre-cracked specimen was then loaded in the autoclave, filled with water and raised to the testing temperature. Transitioning from fatigue crack to corrosion crack was performed to provide an opportunity for the crack to respond as if it had always been an SCC crack, and propagate along the most susceptive path. During transitioning, R was increased from 0.1 to 0.7, and subsequently the loading frequency f was decreased from 1 Hz to 0.001 Hz in several steps. A hold time of 3 000~9 000 seconds at Kmax was then introduced before changing to constant stress intensity factor (K) conditions for SCC evaluation. The transition period is one of the most important parts during a successful SCC crack growth rate test, and failure to transition has often led to the erroneous conclusion that the material was immune to SCC. The duration of SCC test in SCW was about 3 700 hours in total and the crack length versus time was recorded. Figure 3.49 shows the SCC growth rate behaviour at different temperatures and DO concentrations.

Usually, the fatigue crack growth rate (CGR) will drop by ~10 times when decreasing the frequency from 0.01 Hz to 0.001 Hz in an inert environment. However, in supercritical water, the fatigue crack growth rate dropped from 9.4×10-7 mm/s to 3.3×10-7 mm/s (less than three times lower) as the frequency was decreased by 10X, indicating a large accelerating effect of SCW on fatigue crack growth.

To confirm the creep contribution to SCC growth rate, a slow strain rate tensile test was conducted on a bar specimen at the strain rate of 1×10-7/s at 500°C in an inert gas environment (Ar). The specimen had a gage section of 20 mm length and 4 mm diameter, and was loaded at constant load for 360 hours after being strained to 3.37%, as shown in Figure 3.49. After the creep test, the specimen was examined for evidence of creep cracking on the gage surface and the cross section. These inter-granular (IG) cracks on the surface and inside the specimen indicated that this 310S material has high creep cracking susceptibility at 500°C, which definitely contributed to the overall growth rate in SCW environment.

Figure 3.48: **SCC growth curve of the specimen after transition in SCW environment at the temperatures ranging from 400 to 500°C**

(a) (b)

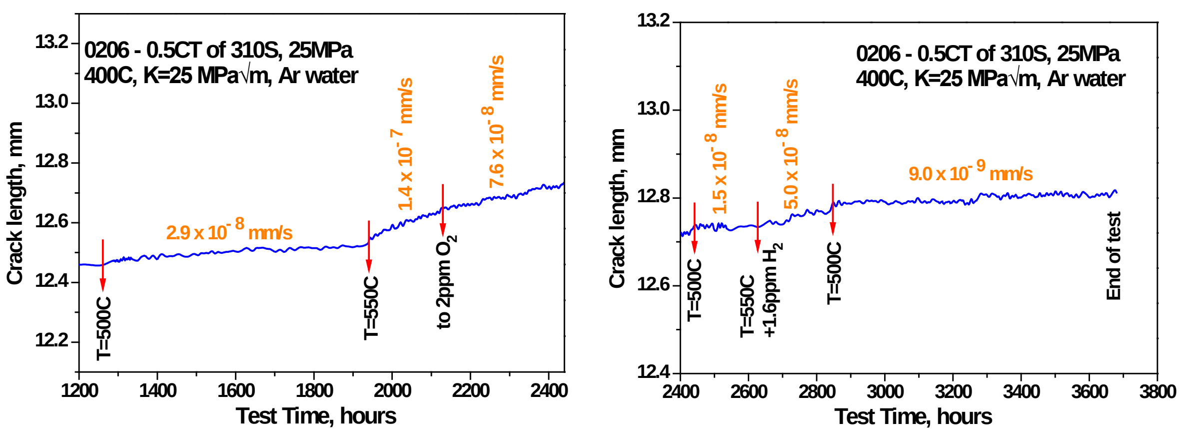
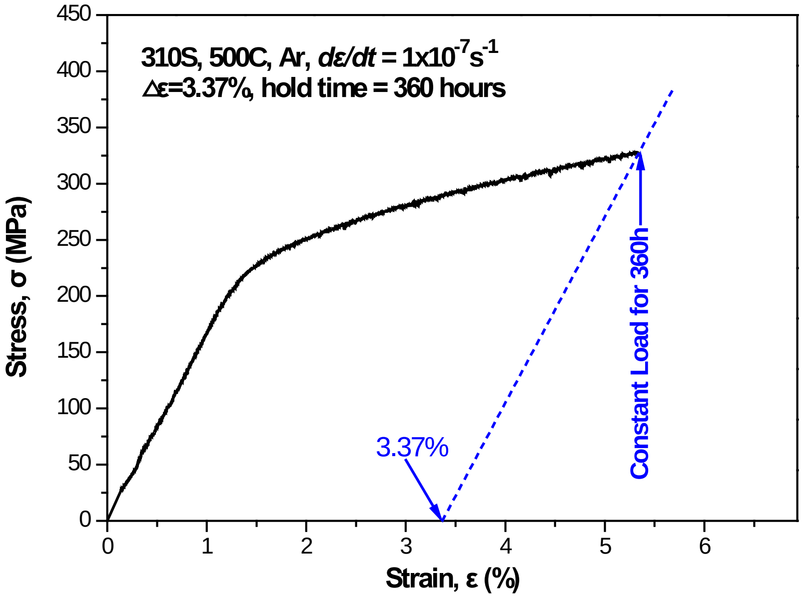


Figure 3.49: **Stress-strain curve of creep test at 500°C**



Tests have been performed to obtain crack growth rate in inert gas environment by direct current potential drop using cold-worked alloy 690. The final heat treatment was mill annealed at 996°C for 20 minutes followed by air cooling. The material was forged to 30% thickness reduction at room temperature. Figure 3.50 shows the detailed microstructures the grain boundaries. A semi-continuous distribution of carbides precipitate at grain boundaries along with small isolated TiN particles, and a few large carbides. No micro-cracks or voids are observed at the interfaces between matrix and carbides. Figure 3.51 shows the electron backscatter diffraction (EBSD) images of as-received and 30% cold-worked alloy 690 materials. The EBSD inverse pole figure maps reveal that the grain was highly compressed for 30% cold-worked material, as shown in Figure 3.51(c), compared with the as-received material in Figure 3.51(a). High degree of misorientation was observed for 30% cold-worked material, especially near grain boundaries, indicating a high residual strain along the grain boundaries after cold work. This high IG residual strain may raise the SCC susceptibility, leading to high intergranular stress corrosion cracking (IGSCC) growth rates.

The curves of crack length versus time obtained on 30% cold-worked specimen are shown in Figure 3.52, with the CGRs and "on the fly" changes marked. Figure 3.52(a) and (b) shows the curves at pre-crack and transition stages, respectively. Figure 3.52(b) and (c) show the SCC growth curves at the temperatures of 450, 500 and 550°C, with CGRs of 5.3×10-7, 6.0×10-6 and 5.9×10-5 mm/s, respectively. Figure 3.52(c) and (d) shows the creep growth at 450 °C, 500°C and 550°C, with the CGRs of 3.9×10-7, 5.1×10-6 and 4.7×10-5 mm/s.

Figure 3.50: **Microstructure of the 30% cold-worked Alloy 690 with IG carbides along the grain boundaries**

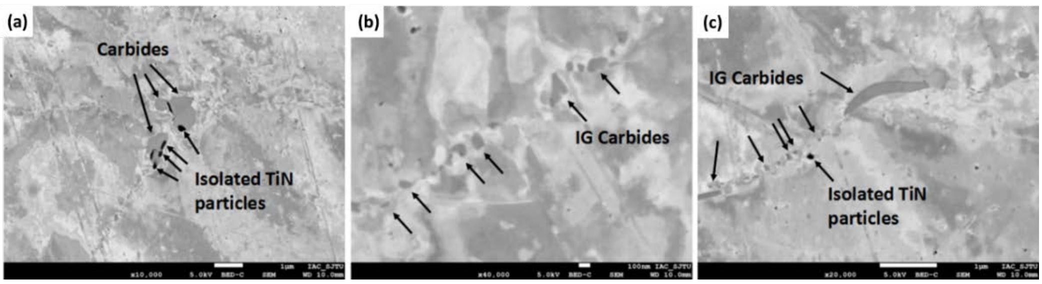


Figure 3.51: **EBSD inverse pole figures (a, c) and local misorientation maps (b, d) for as-received (a, b) and 30%CW (c, d) Alloy 690 materials in the compression section**

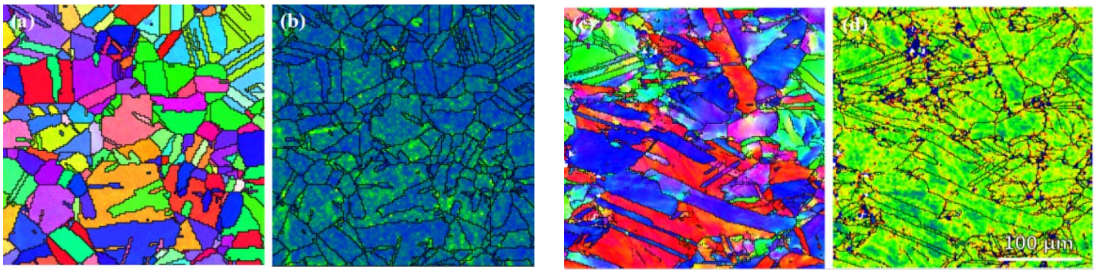
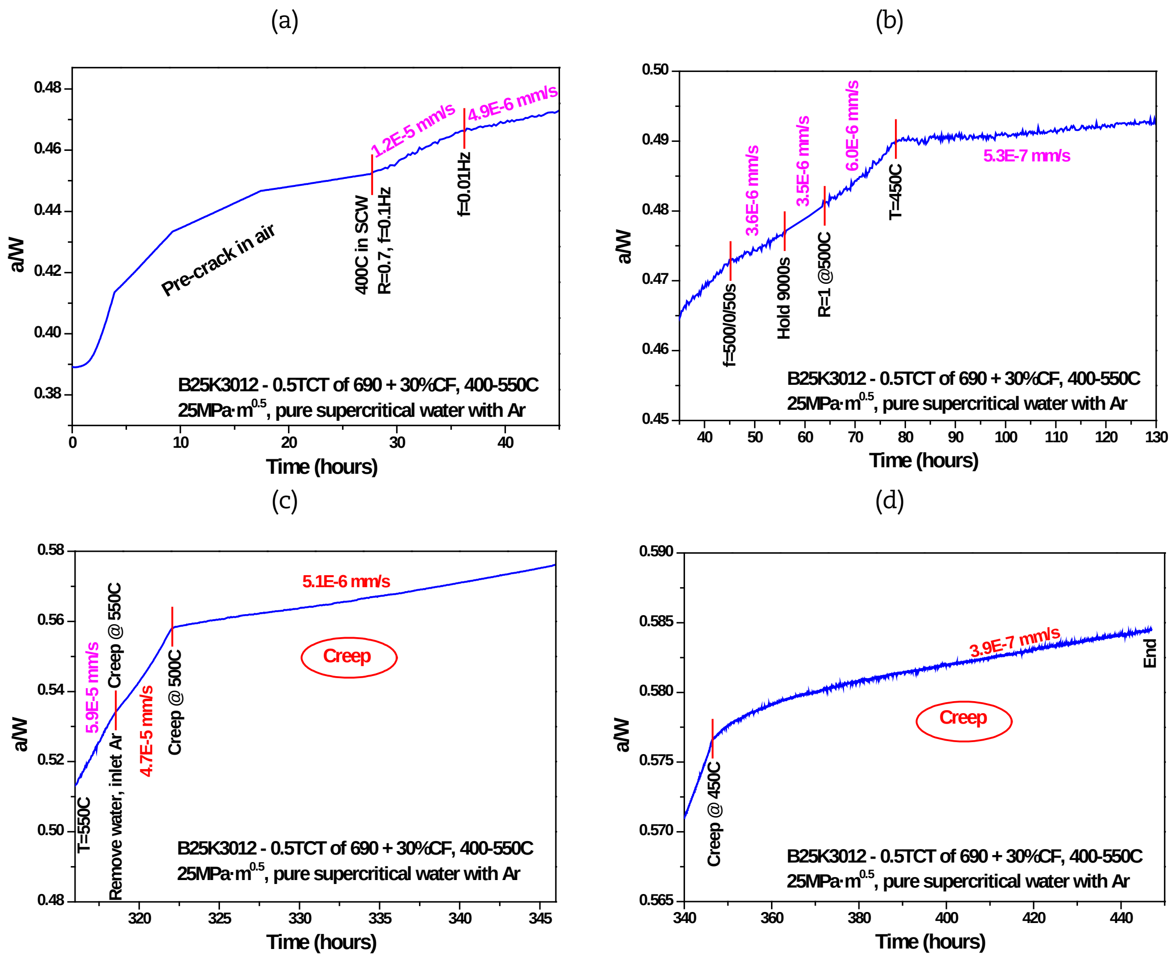


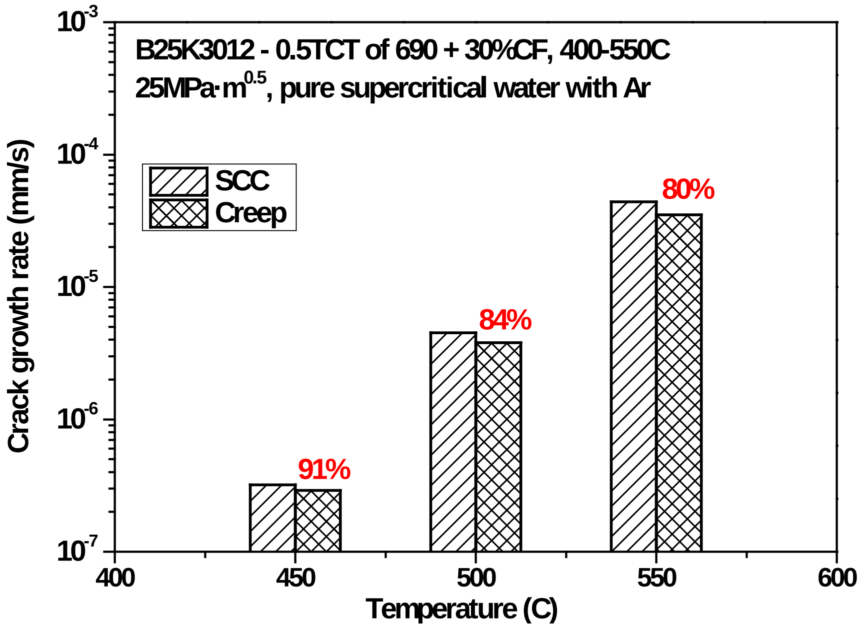
Figure 3.52: **The crack growth curve of 30% cold-worked specimen, showing (a) pre-crack, (b) transition, (c) SCC growth and (d) creep crack growth**



The creep induced CGRs cold-worked alloy 690 has been confirmed to be a comparable level with the SCC under SCW environment. Thus, we can conclude that the SCC CGR is the total CGRs induced by corrosion and creep. For 30% cold-worked alloy 690, the creep contributes more than 80% of the crack growth rate in SCW environment at the temperatures between 450 and 550°C, as shown in Figure 3.53. The results indicate that the creep is the major cracking mechanism that contribute to the cracking of highly cold-worked alloy 690 in SCW.

At VTT, the Academy of Finland Project Interactive Modelling of Fuel Cladding Degradation Mechanisms (IDEA, 2012-2016) finished during the first half of 2016. The ultimate goal of the project was to assess candidate materials performance in SCW conditions in terms of general corrosion and environmentally assisted cracking. In addition to this, major task was to validate and develop further new oxide film modelling procedure applicable for SCW conditions which was introduced originally for LWR conditions. Based on the development work, this model is able to predict oxide film thickness as a function of exposure time in SCW conditions with certain limitations. General corrosion and SCC tests revealed the fact that there is still a knowledge gap in materials selection for SCWRs. It is evident that either development work on existing materials (e.g. coating or surface modification) or completely new material is needed (e.g. ODS-alloys). Besides materials testing itself, also development work on testing facilities was part of the project and significant progress was achieved in mechanical testing by using miniature autoclave equipped with pneumatic based bellows device. This testing system was designed so that it can be utilised in-pile as well.

Figure 3.53: **Comparison of SCC and creep CGRs of 30% cold-worked alloy 690 specimens**



At JRC, the research on SCWR continued within institutional project IntAg-LWR. The main objectives of the project was to assess candidate materials performance in terms environmentally assisted cracking, in particular, acceleration of crack growth rate due to exposure in SCW compared to subcritical water. Furthermore, an iron/iron oxide reference electrode developed by the IFE OECD Halden Reactor Project for corrosion potential and electrochemistry based measurement in sub-and supercritical water was assessed. First two prototypes were installed in the JRC SCW autoclaves. The first tests were focused on assessment of in situ electrochemical potential measurements of AISI 316L in subcritical and supercritical up to 600°C, as well as long-term reference electrode stability and sensitivity to dissolved oxygen content. Following that a dedicated electrochemical cell consisting of iron/iron oxide reference electrode, a Pt-basket counter electrode and a 316L cylindrical working electrode was installed in one of the SCW autoclaves. In situ electrochemical impedance spectroscopy (EIS) measurement has been conducted to investigate three objectives: i) effect of temperature, in particular close to critical point of water; ii) effect of exposure time and iii) effect of pressure. EIS spectra were measured from 230°C up to 604°C to fulfil first objective using both two and three electrode set-up. As the objective two, effect of exposure time was investigated during more than 2 000 h long exposure of AISI 316L in 500°C/25MPa SCW. Effect of pressure was then examined by conducting EIS measurements while gradually decreasing the autoclave water pressure from 25 to 10 MPa. JRC also co-ordinated the 2nd GIF SCWR round-robin exercise on corrosion of candidate materials in SCWR conditions with stress on clarifying soma of the findings of 1st round-robin Inter-laboratory comparison. The objective of the 2nd round-robin exercise was to identify the reasons for the observed differences in the first round-robin exercise. For this purpose, the round-robin exercise has been performed on identical coupon specimens of alloy 800H and 310S austenitic stainless steels in supercritical water. In order to avoid differences in the initial surface conditions of the 2nd round-robin test specimens, JRC Institute for Energy and Transport prepared all test specimens according to the coupon preparation guideline described below. Similar test conditions have been used in each laboratory to assess the reproducibility of the results and the reliability of the test facilities. JRC conducted two tests within the 2nd round-robin exercise in 2016 and the evaluation of the test has been under way.

One of the objectives of Ciemat activities in the field of SCW during 2016 was to study the oxidation and SCC behaviour of nickel base alloy 690 in this special environment. The intergranular carbides have played an important role in the resistance to corrosion of the A600. Nevertheless, the role of these intergranular carbides in the A690 is not well known yet. Corrosion tests performed in the Ciemat supercritical water loop with two kind of A690 specimens (without intergranular carbides [Solution annealed] and with intergranular carbides [thermal treated]) have pointed out a better response of the material to SCC in the SA condition. However, more work is needed in order to complete these results and to gain more in-depth knowledge into the effect of intergranular carbides in SCW.

In addition to this, in 2016 the Structural Materials Division of Ciemat has continued the study about the effect of pressure and temperature on the physicochemical properties of water in the supercritical zone. Moreover, Ciemat is involved in the second international round robin and tests were started in late 2016.

At Research Centre Řež, Czech Republic in co-operation with the University of Chemistry and Technology (UCT), Prague within the Project ARMAT korundum and mullite ceramics and graphite sealing samples were tested in demineralised supercritical water at parameters 570-600°C and 23-24 MPa (for photographs of the samples see Figures 3.56 and 3.57). After the exposure, weight change, flexural strength test and SEM test were performed.

Within the SUSEN Project Phase II, construction of the supercritical water loop and Ultracritical water loop continued. Construction of the S-CO_2 loop was finalised and first commissioning experiments were performed.

Figure 3.54: Korundum and mullite ceramic specimens in holder

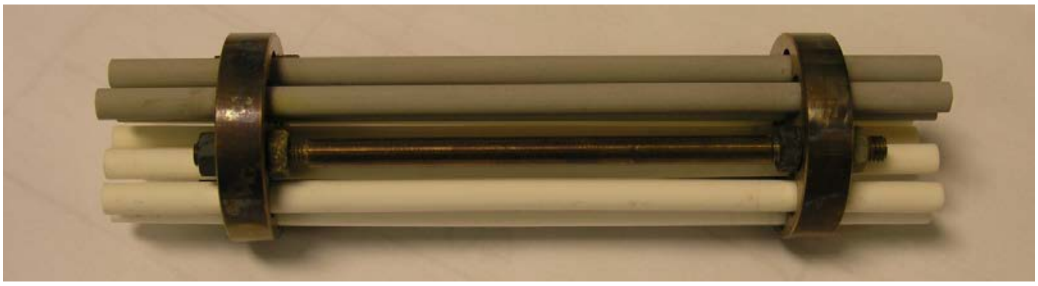
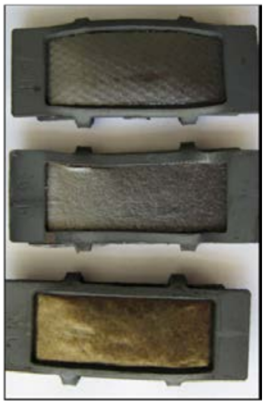


Figure 3.55: **Graphite sealing before exposure**

Figure 3.56: **Graphite sealing after exposure**



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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 Phenix end-of-life tests, and will be continued with the ASTRID Project in France, the restart of Joyo in Japan, the lifetime extension of BN-600 and the start-up of the BN-800 in Russia, and of the China Experimental Fast Reactor.

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

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

- A large size (600 to 1 500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially minor actinides, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors as shown in Figure 3.58.
- An intermediate-to-large size (300 to 1 500 MWe) pool-type reactor with oxide or metal fuel as shown in Figures 3.57 and 3.58.
- 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 3.61.

Figure 3.57: **Japanese sodium-cooled fast reactor (loop-configuration SFR)**

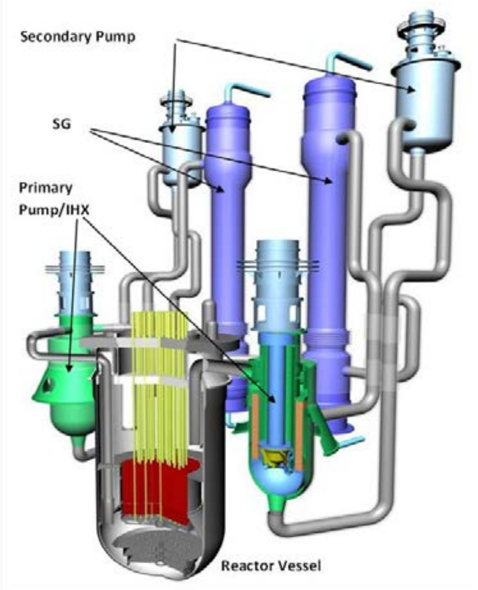


Figure 3.58: **Example sodium fast reactor (pool-configuration SFR)**

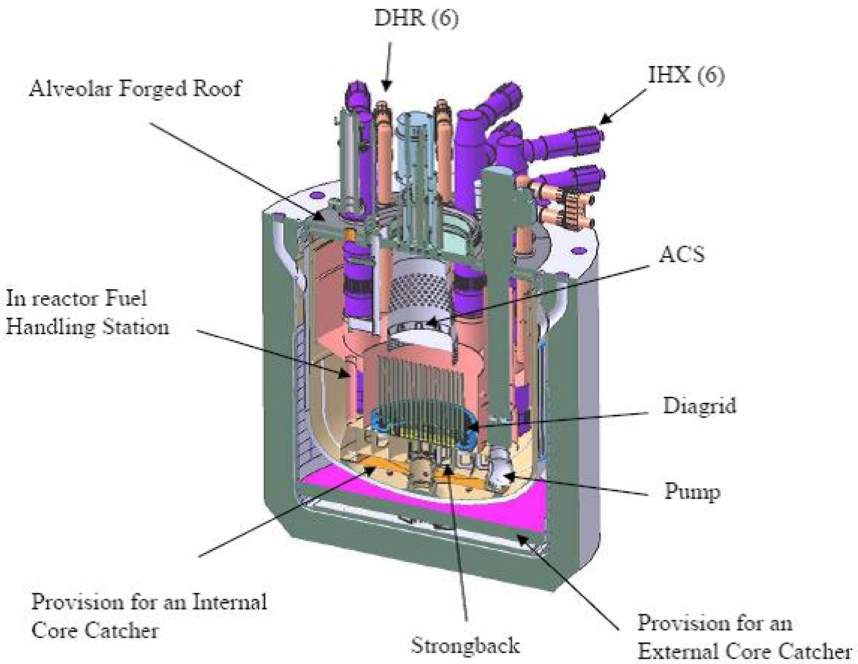


Figure 3.59: **Korea advanced liquid metal reactor (pool-configuration SFR)**

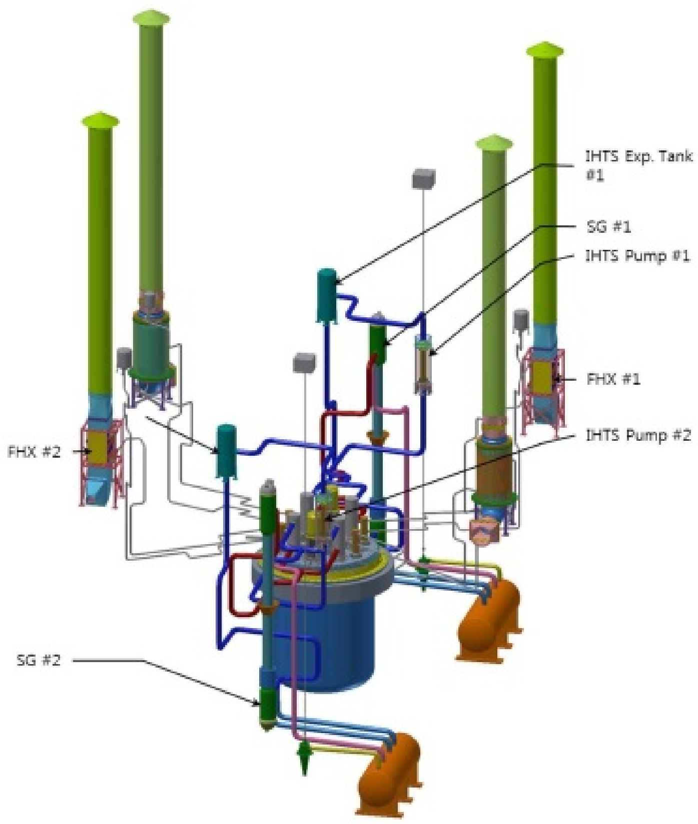
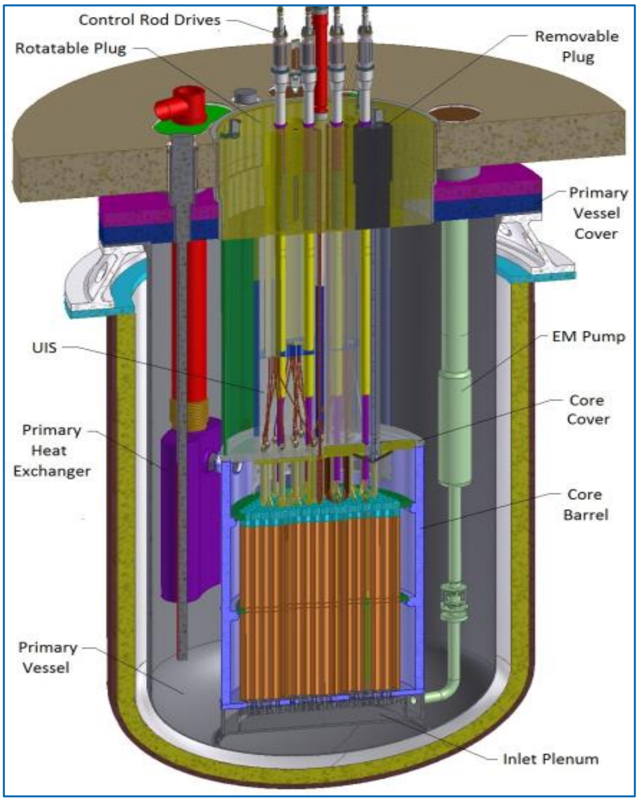


Figure 3.60: **AFR-100 (small modular SFR configuration)**



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.

Status of co-operation

The first system arrangement (SA) for the international R&D of the SFR nuclear energy system became effective in 2006 and extended for another ten years in 2016, the present signatories are:

- Commissariat à l'énergie atomique et aux énergies alternatives, France;
- Department of Energy, United States;
- Joint Research Centre, Euratom;
- Japan Atomic Energy Agency, Japan;
- Ministry of Education, Science and Technology, Korea;
- China National Nuclear Corporation, China;
- Rosatom, 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 Project Arrangement of AF was amended to include the contributions of China and Russia in 2015. The Project Arrangement of GACID 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 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 October 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 SFRs (e.g. Monju, Phenix, BN-600 and CEFR), and WP SO 3 “Studies of innovative design and safety systems” related to the safety technology for the Gen IV reactors such as passive safety systems.

Advanced Fuel Project (AF)

The Advanced Fuel Project aims at developing minor actinide-bearing (MA-bearing) high burnup fuel for sodium-cooled fast reactors to satisfy the Generation IV criteria regarding safety, economy, sustainability and proliferation resistance and physical protection. The R&D activities of the Advanced Fuel Project include fuel fabrication, fuel irradiation and core materials (e.g. cladding materials) development. The advanced fuel concepts include non-minor actinides-bearing (MA) driver fuels for reactor start-up as well as MA-bearing fuels as driver fuels and targets dedicated to transmutation, in order to address both homogeneous and heterogeneous ways of MA transmutation as a long-term goal. Fuels considered include oxide, metal, nitride and carbide. Currently, cladding/wrapper materials under consideration include austenitic as well as ferritic/martensitic steels but aim to transition in the longer term to other advanced alloys, such as oxide dispersion-strengthened steels (ODS).

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

Research on component design and balance-of-plant covers experimental and analytical evaluation of different domains. In order to improve availability of the reactor an important work has been undertaken on advanced in-service inspection and repair technologies, with in particular sensors development and data treatment for example to detect defects under the sodium surface. Some other important topics are investigated such as leak-before-break (LBB) assessment, steam generators and development of alternative energy conversion systems, e.g. using Brayton cycles. Such a system, if demonstrated to achieve the expected economic and efficiency benefits, would reduce the cost of electricity generation significantly. The primary R&D activities related to the development of advanced BOP systems are intended to improve the capital and operating costs of an advanced SFR. The main activities in energy conversion systems 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:
  - 2012: Compile existing self-assessment results for SFR design tracks;
  - 2012: Solicit economics assessment using EMWG methodology;
  - 2013: Solicit proliferation assessment using PRPPWG methodology;
  - 2014: Solicit safety assessment using RSWG methodology.

SO Project:

- Methods, models and codes:
  - 2008-2011: Research collaboration on methods, models and codes for safety technology and evaluation among four countries of France, Japan, Korea and United States.
  - 2012: Research collaboration between China, France, Japan, Korea, Russia, United States and Euratom.
- Experimental programmes and operational experience:
  - 2008-2011: Research collaboration on the experimental programmes and operational experience including the operation, maintenance and testing experience in the existing SFRs (e.g. Monju, Phenix, BN-600 and CEFR) between France, Japan, Korea and United States. (Collaboration with Korea started in 2009).
  - 2012: Research collaboration between China, France, Japan, Korea, Russia, United States and Euratom.
- Studies of innovative design and safety systems:

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

AF Project:

- 2007-2012: Viability study of proposed concepts;
- 2009-2015: Performance tests for detailed design specification;
- 2014-2016: Demonstration of system performance;
- 2017-2027: Evaluation, optimisation and demonstration.

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.

Note: Amendment No.2 of Project Arrangement was approved in 2014.

Main activities and outcomes

System Integration and Assessment (SIA) Project

The SIA Project of the sodium-cooled fast reactor system was started on 22 October 2014 when the Project Arrangement was signed by the representatives of CIAE/China, CEA/France, DOE/United States, JRC/Euratom, JAEA/Japan, KAERI/Korea, and Rosatom/Russia. The Project Plan in the Project Arrangement structures the work scope into several WPs as follows:

- WP 1.1.1: SFR system options definition;
- WP 1.1.2: Contributed trade studies;
- WP 1.2.1: SFR R&D needs;
- WP 1.3.1: General assessment and integration;
- WP 1.3.2: Contributed assessment studies.

Given the nature of work in the SIA Project, specific contributions are only expected for trade studies and self-assessment contributions. The other integration and assessment activities will be conducted directly as part of the Signatory's responsibilities for preparation and consultation at the SIA PMB meetings.

At each SIA PMB meeting:

- the list of major system options and design tracks is updated (WP 1.1.1);
- the comprehensive list of R&D needs (WP1.2.1) is reviewed;

- the recent R&D results of each SFR Technical Project are reviewed to assure consistency with Generation IV system options and R&D needs.

The current roster of SFR system options includes loop, pool and small modular SFR types. For these System options, the current four design tracks are: JSFR (JAEA, loop), KALIMER (KAERI, pool), the Example Sodium Fast Reactor (ESFR, Euratom, pool), and AFR-100 (DOE, modular). These tracks cover a broad range of SFR design characteristics. New design track contributions are expected from several Project Members, for example, the Russian BN-1200 conceptual design and the China CFR-1200 may be proposed as design tracks in future.

A comprehensive list of R&D needs was updated by the SIA PMB members at the last PMB meeting. The revised R&D needs list was approved by the SFR System Steering Committee.

Procedures for SIA review of the technical projects continue to evolve. The current approach is to have project members from the host country provide technical updates at the SIA PMB meeting. This approach was still quite effective to provide a good overview of the complete set of Generation IV R&D activities, and to stimulate discussion regarding the impact and integration of recent accomplishments.

In 2016, the following trade studies were contributed within WP 1.1.2:

- Scenarios for ESFR deployment in Europe (Task 1.1.2.EU1);
- Inlet temperature study for CFR-1200 (Task 1.1.2.CH1);
- Comparison between nitrogen conversion system and water steam conversion system for ASTRID (Task 1.1.2.FR1);
- Impact of outlet temperature on SFR performance (Task 1.1.2.US1);
- Safety self-assessment of JSFR track (Task 1.3.2.JP1) was contributed within WP 1.3.2.

Safety and Operation Project

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

WP SO 1: Methods, models and codes

CIAE (China) is developing the FASys system analysis code to analyse the system response in a wide range of SFR transients. The development plan includes two phases: Phase I (2012~2016) devoted to main models development, main function definition and implementation, preliminary validation and preliminary applications. Phase II (2017~2019) focuses on all models development, friendly interface development and detailed validation and verification. The major code models include sodium loop thermal-hydraulic model (pipes, pumps, IHX, pools), core analysis model (point kinetics and reactivity feedback, pin heat transfer and single-phase coolant thermal-hydraulic model) and plant protection and control system models. The models mentioned above are divided into three types: hydraulic model, thermal model and neutron kinetics model. The major modules in the code include pre-processor module, system geometry building module, steady state calculation module, transient calculation module, postprocessor module and output module. The FASys code is architected in the FORTRAN 95 language with good coding style, and is developed by mainly adopting the structured programming method.

The validation of the code was initiated by comparing to the CEFR commissioning tests. In the case of the hydraulic model validation, Figure 3.62 shows the results on primary pumps flow during coast down. In this experiment, both of the CEFR primary circuit pumps are coasting, and the pump flow meter value is recorded. As seen in the figure, the agreement is quite close, with FASys predicting the pump flow with time, as compared to the experimental data. Another example of the validation is presented in Figure 3.9 with the comparison between the loss of station power test data and calculation results in IHX first side inlet temperature and outlet temperature. It can be indicated that the calculation results were matched well with the test data.

Figure 3.61: **CEFR Primary pumps flows**

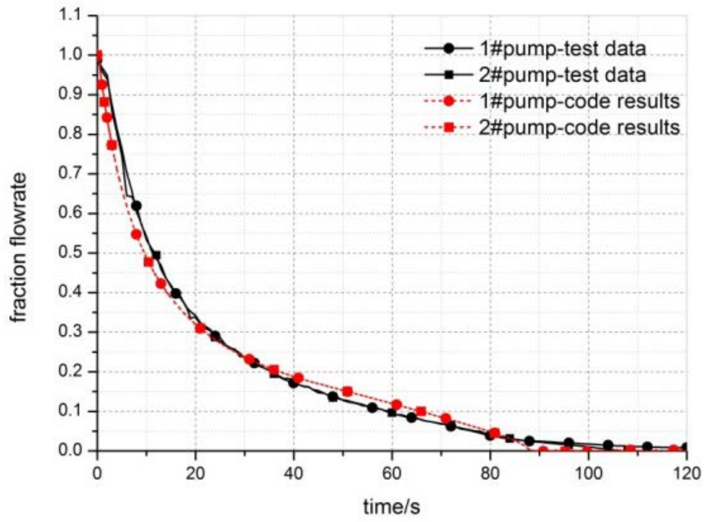
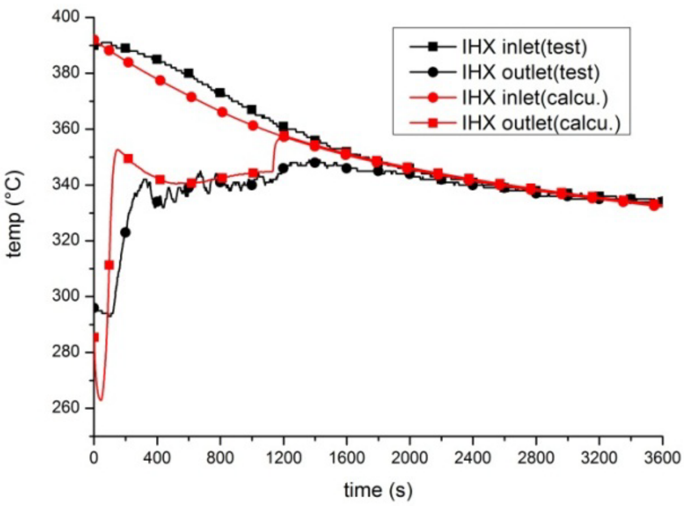


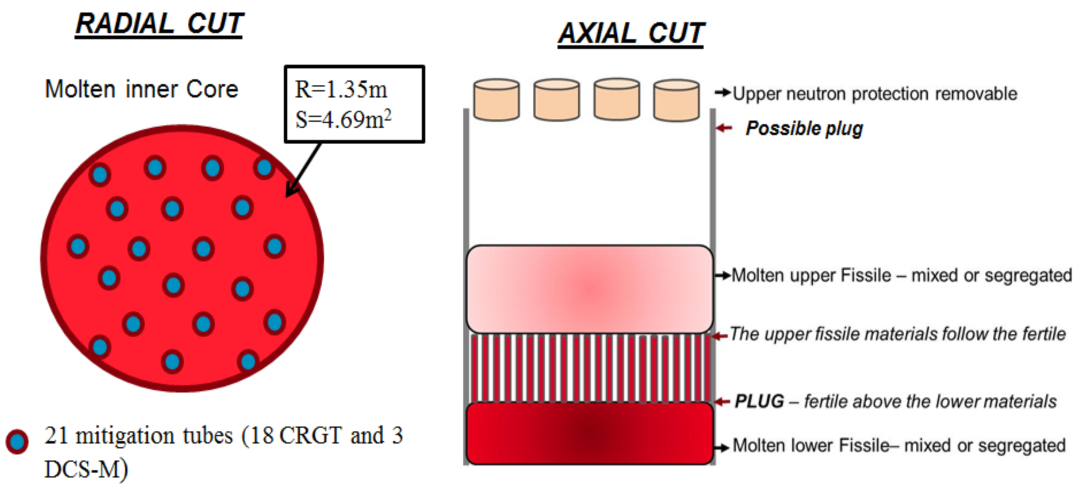
Figure 3.62: **IHX temperature during a loss of station power**



The CIAE (China) is also developing the hypothetical core disruptive accident (HCDA) analysis code. Core disruptive accident analysis is important for fast reactor severe accident analysis. The major code models include: neutron dynamics model, reactivity feedback model, core thermal model, core disruptive model.

The CEA is developing a physical-probabilistic tool dedicated to molten material core discharge during postulated severe accidents (Figure 3.64)

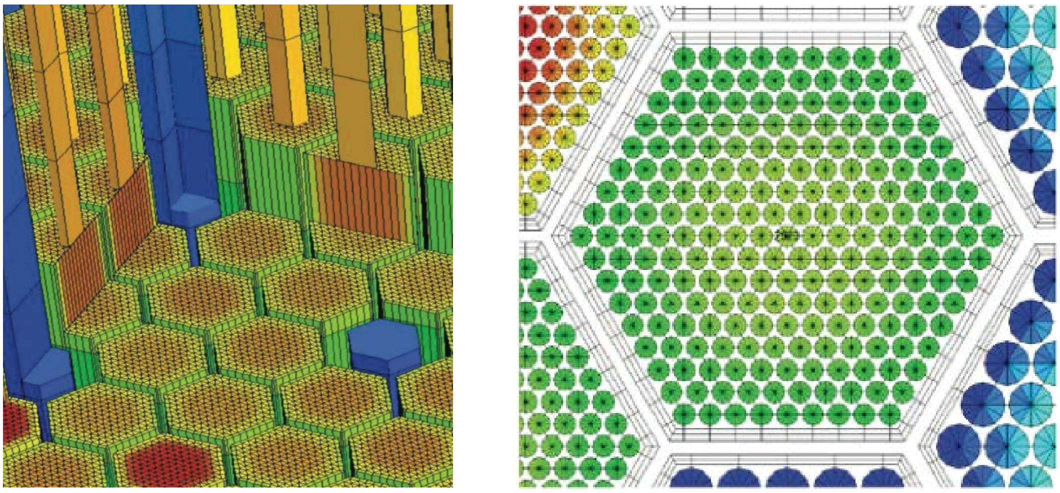
Figure 3.63: **Analytical tool for severe accidents simulation**



The study deals with the assessment, against SIMMER results. This 0D tool handles heat transfers from molten, possibly boiling, pools to mitigation tube walls, fuel crust evolution, segregation/mixing of fuel/steel pools, radial thermal erosion of mitigation tube wall, and discharge of molten material with axial thermal erosion of the transverse tube, coupled with neutronic evolution of the fuel power. This very low time consuming tool enable large sensitivity studies on different physical and design parameters. The physical models and the calculation scheme of the physico-statistical (also called analytical) tool are generic to the treatment of various material molten pool of constant radius. This tool is parametrised to facilitate sensitivity evaluations (such as initial reactivity, wrapper thickness of the mitigation tubes, initial material masses, fuel power…). This tool couples the temporal evolutions of materials located inside the upper and lower fissile zones to the evolution of the global core neutronics. The considered molten pools are composed of steel and fuel which could be mixed or segregated (steel layer above a fuel lower pool). The spatial distribution of materials between these two pools evolves during the transient depending on material temperature. Various configurations are treated: totally segregated materials, partially segregated configuration where the steel mass is distributed between a steel layer which is above and a lower mixed steel/fuel pool or totally mixed configuration where the pure steel layer has disappeared. The lower mixed pool is considered homogeneous with physical properties dependent on the proportions of the various materials inside the pool. For the reference case, the core is assumed at residual power and at initial time. The reactivity is null. It is assumed also no reactivity supply during the transient (caused for example by a sodium return inside the plenum). The transient evolutions calculated with the simplified analytical tool and SIMMER are similar and the same reactivity contributions are observed. The material ejection in the analytical tool takes few tenth of seconds where as it is instantaneous in SIMMER. This behaviour has been explained and seems realistic according to some past experimental results. Finally, it has been demonstrated that this analytical code will be a valuable tool to perform sensitivity studies and highlights the most influent parameters. It will be used in support to the design of mitigation devices and will enable to perform large statistical treatment of uncertainties.

The CEA R&D Programme on Generation IV SFRs includes the study of reactor behaviour during accidental transients possibly leading to sodium boiling, such as an unprotected loss of flow (ULOF). To that end, two existing CEA thermal-hydraulic codes are being extended to model two-phase sodium flows: CATHARE (system dynamics) and Trio_U MC (sub-channel) (Figure 3.64).

Figure 3.64: **Trio_U MC model of SFR core**



CATHARE can be used to model a complete experimental loop or reactor circuit with a 1D representation of the test section; this model may be replaced with a Trio_U MC model of the test section itself in order to account for 3D boiling effects. In the case of the 6-equations CATHARE code, the applicability of its current closure laws to two-phase sodium flow must be evaluated; for the 3-equations MC code, new numerical methods are needed as well.

Simulations with these codes were done recently of the GR19 sodium boiling experiments done at CEA Grenoble in the 1980s. These tests were performed on a 19-pin out-of-pile mock-up of the SUPERPHENIX fissile subassembly installed on the CFNa III experimental loop: they include static and quasi-static boiling tests, as well as slow and fast loss-of-flow transients. These tests enabled key advances in the understanding of the phenomenology of sodium boiling, including the possibility of stable boiling. Simulations were undertaken in order to provide a preliminary validation of the two codes and to guide future code improvements.

The results presented show that the CATHARE code can achieve good agreement with the experimental data: however, some closure laws (especially for heat transfer) will still have to be revised to better fit sodium boiling physics. Meanwhile, Trio_U MC is capable of predicting correctly the occurrence and extension of local boiling in boiling steady states: however, further improvements will be needed to correctly model unstable and transient boiling (as well as the coupling to CATHARE).

The need of new experimental programmes is considered to account for the design innovations of Gen IV SFR subassemblies (such as the inclusion of a sodium plenum above the fuel pins), as well as to better improve and validate the codes' physical models. The results presented show that code validation in two-phase sodium is highly sensitive to several experimental parameters: mastering these parameters should be paramount to the usefulness of new experimental programmes for reliable code qualification.

The JAEA has been developing probabilistic risk assessment (PRA) methodologies against various external hazards, such as snow, tornado, strong wind, rainfall, volcanic eruption and forest fire. The PRA methodology consists of external hazard curve evaluation and event sequence analysis methods to estimate a core damage frequency. To develop the forest fire PRA methodology, JAEA has evaluated an external hazard curve of the forest fire based on a logic tree. The logic tree consists domains of “forest fire breakout and spread conditions”, “weather condition”, and “vegetation and topographical conditions”. A location nearby a typical nuclear power plant site in Japan was selected for our studies. The frequency of a large forest fire of the location is approximately 1/5 of the average in Japan. A number of forest fire simulations were performed to obtain a response surface for a frontal fireline intensity at different combinations of wind speed and humidity. The hazard curve has been successfully evaluated by Monte Carlo simulations where one sample gave a unique intensity from the response surface and its frequency was given by the combination of the branching probabilities in the logic tree.

In JAEA, fundamental experiments of sodium-concrete reaction (SCR) were performed by thermal analytical techniques for developing the reaction model. As a series of possible reactions on concrete ablation in SCR, kinetic behaviour of Na2O-SiO_2 and Na2O-concrete aggregate reactions were investigated. Kinetic parameters were obtained by using kinetic laws such as Kissinger and Freedman methods.

In the United States, construction and operation of a nuclear power installation requires licensing by the US Nuclear Regulatory Commission (NRC). A vital part of the licensing process is the analysis of the source terms that represents the potential release of radionuclides during normal operation and accident sequences. Historically, source term analyses have utilised deterministic, bounding assessments of radionuclide release to the environment. Significant advancements in technical capabilities and knowledge have enabled the development of more realistic analyses such that a mechanistic source term assessment is now expected to be a requirement for advanced reactor licensing. Argonne National Laboratory has assessed the state of development of a mechanistic source term for SFR and qualitatively identified and characterised the major sources and transport processes. Due to common design characteristics among current US SFR vendor designs, a metal fuel, pool-type SFR was selected as the reference design for this work, which allows gaps and uncertainties in the current knowledge base to be identified. Radionuclides originate both in-vessel and ex-vessel, including in-core fuel, primary sodium and cover gas clean-up systems, and spent fuel movement and handling. Transport phenomena affecting various release groups include fuel pin and primary coolant retention and behaviour in the cover gas and containment. Radionuclides released from a primary sodium fire are also considered as potential sources. Available experimental data relevant to the aforementioned phenomena and operating incidents at domestically operated facilities have been reviewed. Following this initial assessment, Argonne developed estimates for the release fraction of radionuclides from metal fuel pins to the primary sodium coolant during fuel pin failures at a variety of temperature conditions. Release estimates were based on the findings of an extensive literature search that included past experiments and reactor fuel damage accidents. Data sources for each radionuclide of interest were reviewed to establish release fractions, along with possible release dependencies, and the corresponding uncertainty levels. Considering the extensive range of phenomena affecting the release of radionuclides, the existing state of knowledge generally appears to be substantial, and may be sufficient in most areas. For core damage accidents, high retention rates can be expected within the fuel matrix and primary sodium coolant for all radionuclides other than the noble gases. These factors greatly reduce the magnitude of possible radionuclide release to the environment. Although the current knowledge base is substantial – and radionuclide release fractions were established for the elements deemed important for the determination of off-site consequences the following gaps were identified.

- There is uncertainty regarding the transport behaviour of iodine, barium, strontium, tellurium and europium during metal fuel irradiation to high burnup levels. The migration of these radionuclides within the fuel matrix and bond sodium region can greatly affect their release during pin failure incidents. Post-irradiation examination of existing high burnup metal fuel can likely resolve this knowledge gap.
- Data is sparse regarding radionuclide release from molten high burnup metal fuel in sodium, which makes the assessment of radionuclide release from fuel melting accidents at high fuel burnup levels difficult. This gap could be addressed with fuel melting experiments using samples from the existing high burnup metal fuel inventory.

- The available thermodynamic data regarding the behaviour of lanthanides and actinides in liquid sodium is limited. However, a determination of the data requirements for mechanistic source term development should be formally made prior to the expenditure of significant research efforts to expand that data.

In Korea, an effort to expand the SAS4A models for the analysis of metal fuel cores has been performed in KAERI in the framework of a collaboration with ANL. The SAS4A safety analysis code, originally developed for the analysis of postulated severe accidents in oxide fuel SFR, has been significantly extended to allow the mechanistic analysis of severe accidents in Metallic Fuel SFRs. The new SAS4A models track the evolution and relocation of multiple fuel and cladding components during the pre-transient irradiation and during the postulated accident, allowing a significantly more accurate description of the local fuel and cladding composition. The local fuel composition determines the fuel thermo-physical properties, such as freezing and melting temperatures, which in turn affect the fuel relocation behaviour and ultimately the core reactivity and power history during the postulated accident. The models describing the fission gas behaviour, fuel cladding interaction, clad wastage formation and cladding failure models have been also significantly enhanced. The paper provides on overview of the SAS4A key metal fuel models emphasising their new capabilities, and presents results of SAS4A whole core analyses for selected PGSFR postulated severe accidents.

EURATOM investigated the capability of modern system codes to simulate the ASTRID core behaviour in the most representative design-basis accident, the ULOF, before the onset of sodium boiling. Several organisations participated in the investigation with various system codes including CATHARE-2, ATHLET-3.1A, SIM-SFR, SAS-SFR, SAS4A, SPECTRA and TRACE. Using neutronic and thermal-hydraulic specifications and calculated safety coefficients assuming end of equilibrium core conditions, ASTRID core models and point kinetics models were developed by the participants. A common ULOF transient specification for the simulation was set up and relevant reactivity feedbacks were identified. Prior to the ULOF simulation, comparisons of the steady state analyses for selected ASTRID parameters were performed including fuel centreline temperature, fuel pellet outer surface temperature, clad inner and outer surface temperatures, coolant temperature, fuel-clad gap size and gap conductance, as well as fuel ΔT and fuel-clad gap ΔT. The agreement between the steady state results was found satisfactory. For the ULOF simulation, the failure of the reactor shutdown system and all primary pumps is assumed whereas the secondary system was assumed fully functional. All the participants simulated the ASTRID ULOF transient up to the onset of sodium boiling with their system codes and the results were then compared and analysed in order to identify sources of discrepancies. The results of the simulation show good agreement for the pump coast-down flow rate characteristics, the reactor power, core outlet coolant temperature, reactivity feedbacks, core coolant inlet and vessel lower plenum temperatures, Core coolant outlet and vessel upper plenum temperatures, vessel wall temperature, maximum fuel and clad temperature. Significant discrepancies in the calculated ULOF sodium boiling onset transient time between the various system codes were noticed. However, a second round of iteration of the simulation succeeded in reducing the maximum differences to an acceptable level.

WP SO 2: Experimental programmes and operational experiences

The JAEA has investigated the capability of natural circulation for core cooling in Monju during a station blackout (SBO) induced by an earthquake and a subsequent tsunami. The plant dynamics analysis code Super-COPD was used for the investigation, which was validated by using the preliminary natural circulation test data in Monju. As a result, it was concluded that the decay heat can be safely removed by natural circulation under such an SBO condition.

The US Department of Energy completed testing at the air-based Natural Convection Shutdown Heat Removal Test Facility (NSTF) at Argonne National Laboratory. After nearly four years of scaling studies, project preparations, and construction of the 1/2 scale test facility, experimental operations on the NSTF began in late 2013 with the initial fire-on and bake-out of the 220 flat plate resistance heaters. Shakedown and scoping activities were then completed in early 2014. The project team began data quality testing shortly after and has since completed over 1 300 hours of active test operations. Results comparing computational fluid dynamics simulations with high-fidelity thermal measurements. Throughout the 20-month testing window, performance metrics were observed by varying parameters of integral power, power profile, single and dual chimneys, reduced discharge elevations, inclement weather, prototypic decay heat curves, blocked riser ducts, and adjacent inlet/outlet ports. At steady state conditions and prototypic power levels, the system performs its cooling related function well and is able to maintain safe limits on the reactor vessel walls regardless of design-basis faults. Disruptions due to minor blockage in riser channels and chimney ductwork (50% induced) introduce minimal rises in-reactor vessel temperature (≤10°C observed) and do not pose a severe safety hazard. The system is robust to meteorological perturbations including wind excursions (24.5 m/s observed) and temperature fluctuations (-6°C to 32°C observed). However, as with any natural circulation or chimney based system, the facility exhibits certain sensitivity to a subset of scenarios dependent on meteorological and operating conditions. Analogous to priming a chimney flue on a cold winter day, appropriate engineering controls must be made during sensitive operating times to prevent system wide instabilities that degrade heat removal performance. These sensitive operational windows have been found to be limited to: i) start-up periods when the system is still thermally and hydraulically developing; and ii) periods of low power removal by the reactor vessel cooling system. The controls used to mitigate meteorological perturbations during the test series centred on actuator valves along the chimney duct work. By introducing flow resistance on the outlet of either both or a single chimney loop, the NSTF is successfully able to overcome start-up and wind induced instabilities. Work is in progress towards examining passive alternatives to these active actuator valves and has encompassed separate effects studies on a reduced scale test facility. Several weather cap designs were tested for their effectiveness in preventing down-draft phenomena (e.g. flow reversals). The planned objectives were achieved during the project period and resulted in a high-quality experiment test facility that is supported by a strong administrative programme. The programme has followed NQA-1 standards in all aspects of operation with compliance assessed during multiple audits. The test facility has successfully generated data which quantifies the heat removal performance during a wide range of operating conditions. The archived data suites are suitable to support efforts in ascertaining the viability of air-based reactor vessel cooling system concepts as a decay heat removal system for future reactor designs.

The Institute of Physics and Power Engineering has performed experiment with modelling fuel pin failure under ULOF accident conditions in SFR. The experiment was carried out on 19-rod model assembly at the PLUTON test facility with sodium coolant. Main purposes of the experiment were as follows: Identification of principal mechanisms of degradation of fuel pin simulator claddings; evaluation of axial material distribution in final state of the model assembly; evaluation of blockage phenomena of the model assembly cross sections; estimation of material ejection outside of the model assembly. A specific thermal effect of thermite reaction was equal to 1.6 MJ/kg, temperature of the thermite reaction was about 3 100K. Sodium temperature in reaction zone of test section was preliminarily increased by heaters up to 550°C. Initiation of thermite reaction in fuel pin simulators of model assembly was provided by voltage supply to ignition system. Three basic mechanisms of cladding degradation were identified:

- temperature stresses in cladding material;
- melting cladding material;

- dynamic effects caused by fast conversion of thermal energy of fuel pin simulator corium into mechanical work during thermal interaction of corium with sodium.

Basing on the experimental results, it was evaluated that the coefficient of conversion of thermal energy of fuel pin simulator corium into mechanical work was equal to 0.115%. Zone of global fuel pin simulator claddings degradation made approx. 65% of the model assembly height and it was localised mainly in area of rod bundle with increased density of thermite load. Total amount of products of the thermite reaction ejected outside of the model assembly borders made 75-80% of initial mass of thermite mixture. Almost total blockage of cross section of the model assembly in its lower part was revealed.

Advanced Fuel Project

The first project period, from 2007 to 2016, has been structured in three stages: evaluation of advanced fuel and material options, minor actinide-bearing fuels evaluation, and assessment of high burnup capability of advanced fuels and materials. During the first stage, fuels under consideration were mixed uranium-plutonium-based driver fuels with a minor actinide content of a few percent in accordance with the so-called homogeneous recycling path. During the second stage (minor actinide fuels evaluation) the scope covered homogeneous and heterogeneous recycling paths, with higher minor actinide concentrations in dedicated fuels located throughout the core and only at the core periphery, respectively. The final selection of fuels will be dependent upon multiple domestic factors for each country. Nevertheless, the evaluation pointed out that experience on oxide and metal fuels is highest, while nitride and carbides are still at an early stage of development. Austenitic as well as ferritic/martensitic materials were recommended as starting options for core materials with the aim of transition to other advanced alloys, such as oxide dispersion-strengthened (ODS) steels, in the longer term. The findings of the first project period were merged into a recommendation report on advanced sodium fast reactor fuel types.

In 2016, developments on fuels, core materials and preparation processes have continued. Post-irradiation examinations (PIE) of irradiated fuels as well as physical properties determination on fresh fuels have continued regarding oxide, metallic and nitride fuel-based systems. Recently completed irradiation tests have been analysed and new irradiation tests have been started or are under preparation.

In particular, PIE and performance analysis for minor actinide oxide fuels have continued on AFC-2C and AFC-2D, which were irradiated in the Advanced Test Reactor (ATR), and after successful completion of the Dispositif d'Irradiation d'Actinides Mineurs dans Osiris (DIAMINO) irradiation in the OSIRIS reactor the pins are now being prepared for PIE in the Fuel Examination Laboratory (LECA) facility.

Analysis of minor actinide and rare earth containing metal fuels irradiated in the ATR proceeded in 2016 with the PIE of AFC-2E, and the ATR irradiation test series on metal based fuels is now continuing with the irradiation of AFC-3F. After successful completion of PIE on the U-Zr based fuel from the first HANARO irradiation test, a second irradiation test up a medium burnup of circa 6 at% was prepared and is now ready to start in the HANARO reactor.

The investigation of corrosion resistance of minor actinide-bearing oxide and inert matrix fuels in liquid sodium has been continued, and progress has been made on the development of more advanced and simplified americium bearing oxide fuel fabrication routes, like the weak acid resin conversion process.

Regarding cladding development, fabrication and characterisation of ferritic/martensitic cladding tubes have continued while preparation for evaluation of the materials irradiation tests advanced. The development and testing of ODS steels is advancing, and in 2016 two Material Test Assemblies (MTA) with ODS steels samples have been loaded into the BN-600 reactor for irradiation testing up to 145 dpa.

Component Design and Balance-of-Plant Project

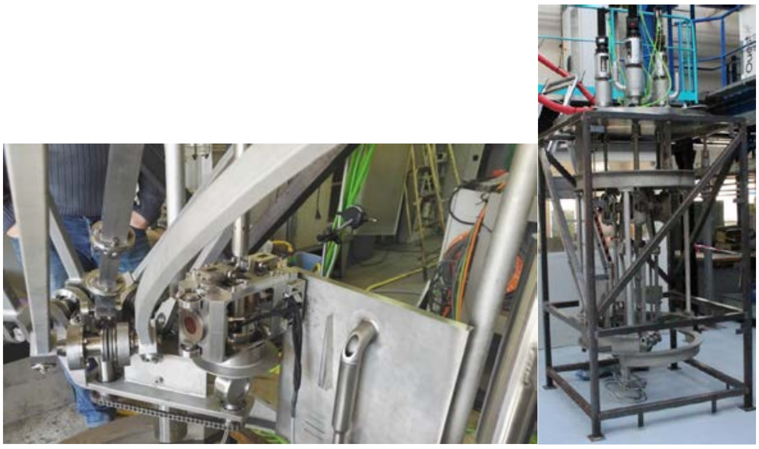
The CD&BOP Project started in October 2007 when the Project Arrangement was signed by the members of CEA/France, DOE/United States, JAEA/Japan and KAERI/Korea. The CD&BOP activities include in-service inspection and repair technologies, LBB assessment technology and sodium-heated steam generators. Supercritical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

Inspection technologies

This topic has largely been studied during 2016 with several work axes.

The first one concerns the development of a new device needed to make the demonstration of the techniques devoted to under sodium viewing. Indeed as the sodium is opaque a large improvement of inspection techniques that rely on the ability to “see” under the sodium surface is important. The retained technique uses ultrasonics. The tracks of development are focused on sensors development and data treatment (images reconstruction). During past years some demonstrations of feasibility were achieved in simulant fluid (water). To go further, it is mandatory to have a demonstration in liquid sodium; it means that the sensors must be available and also the system able to provide under sodium movement of these sensors. In 2016 the tool needed to obtain such demonstration (i.e. a robot able to operate under sodium) has been realised. Figure 3.65 shows a picture of the robotic arms called VENUS. Its first operations started in representative conditions (hot liquid sodium) in December 2016.

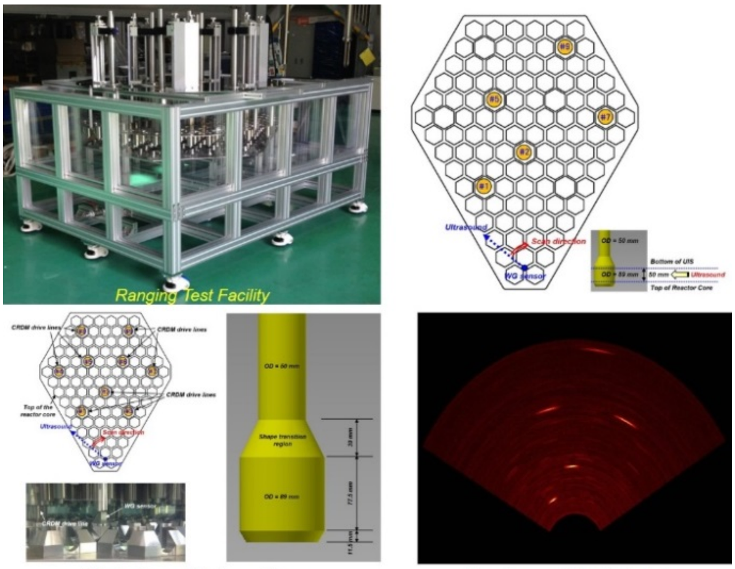
Figure 3.65: **VENUS robotic arm for under sodium viewing**



In parallel, investigations of the inspection methods were pursued: surface imaging by tomographic approach (experimental results), proposal of non-destructive imaging of thick welds based on an adjoint method (simulation results), and future tasks (application to thick welding).

An alternative to the use of immersed ultrasonic sensors could be, for some applications, the use of waveguide sensors. For the performance demonstration of the ranging waveguide sensor, a crucial test facility simulating real-scaled obstacles which might exist between the top of the reactor core and the bottom of the upper internal structure was manufactured and several tests in water conditions were conducted for various arrangements of obstacles (Figure 3.68). FEM models for the optimisation of the ranging waveguide sensor have been developed. The recent design of the waveguide sensor and the key parameters with objective values for optimisation have been discussed within the framework of collaboration between two members of the CD&BOP group.

Figure 3.66: **Ranging test facility and performance test result**



Obstacles and their positions

Ranging result for 4 obstacles

LBB assessment technology

In this field, the philosophy of Leak Break Assessment has been refined and applied to MONJU SFR.

Moreover, to perform the LBB assessment of the SFR pipes made of Mod.9Cr-1Mo steel, a fracture assessment method was developed taking stiffness evolution into account. The method estimates the critical crack size using two-parameters method, synchronising with FE analyses. The rotational stiffness evaluation method was also proposed. It predicts plastic collapse stress conservatively.

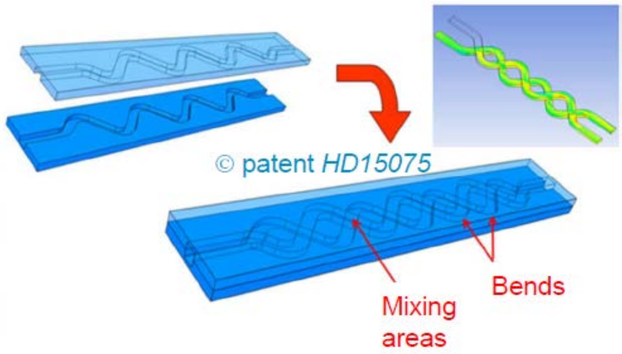
High-temperature crack growth models of Mod.9Cr-1Mo steel associated with defect assessment have been studied. Fatigue Crack Growth and Creep Crack Growth (CCG) tests on P91 specimens were conducted. Mathematical models of (fatigue and creep) crack growth were obtained from tests. Conservatism of the crack growth models was quantified.

Advanced energy conversion systems Brayton cycle

Concerning the energy conversion systems, a key component is the heat exchanger between the secondary and tertiary fluids. Part of the 2016 studies were focused on the way to monitor this component during operation, in particular techniques to detect any leak between the secondary and tertiary fluids were investigated. Passive acoustic leak detection for sodium fast reactors has been studied based on Hidden Markov models. The preliminary results showed good performance in term of detection rates and low false alarm rate. However, consolidation of the study must be done with new experiments with more representative background noise and difference orifice shapes, and in realistic geometry with representative materials.

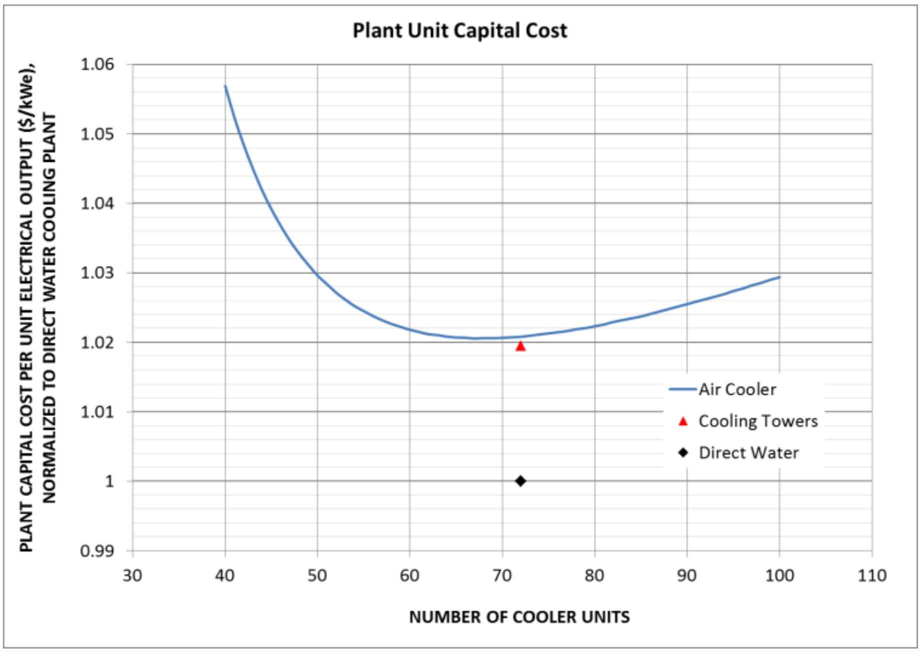
Concerning the design of compact heat exchangers, the retained option is a plate heat exchanger with small channels. However, this technology is not industrially developed at the industrial scale, with the retained materials and with adequate operating conditions. A specific development for heat exchangers between sodium and nitrogen has been accomplished. At first, optimisation of channel design of the sodium/gas heat exchanger is done with the study of different heat exchange patterns (example given in Figure 3.67). Evaluation of the resulting performances was done numerically and experimentally.

Figure 3.67: **Example of improved channels scheme**



The possible use of commercially available, cost effective, finned tube air coolers for direct heat rejection to the air atmosphere heat sink in the case of the sCO_2 Brayton cycle has been studied. A re-optimisation of cycle conditions for dry cooling has been performed, preserving the gross efficiency of the cycle (slightly increased) and limitation of capital cost increase (2% see Figure 3.70).

Figure 3.68: **Optimisation of number of cooler units and impact on the NPP capital cost: comparison of air cooler, cooling towers and direct water**



This preliminary and simple examination shows that dry air cooling can be practically utilised with both the sCO_2 Brayton cycle with finned tube air coolers and the superheated steam cycle with finned tube air-cooled condensers, provided that the cycle conditions are selected appropriately.

The Plant Dynamic Code was also used in 2016 to study different transients. For a 1 000 MWt SFR with a sCO_2 Brayton cycle power converter, double-ended guillotine rupture of one of four large diameter CO_2 pipes between the sodium-to-CO_2 heat exchanger and turbine results in rapid cycle depressurisation and release of CO_2 inventory in about one second. The depressurisation timescale is sensitive to the assumed break flow area. For a one-way break with an equivalent diameter equal to half that of one large diameter pipe, the sCO_2 Brayton cycle depressurises and loses CO_2 in 22 seconds – Sodium in the sodium-to-CO_2 heat exchanger is overcooled for the first four seconds due to a greater than normal CO_2 flowrate and undercooled thereafter.

Steam generators

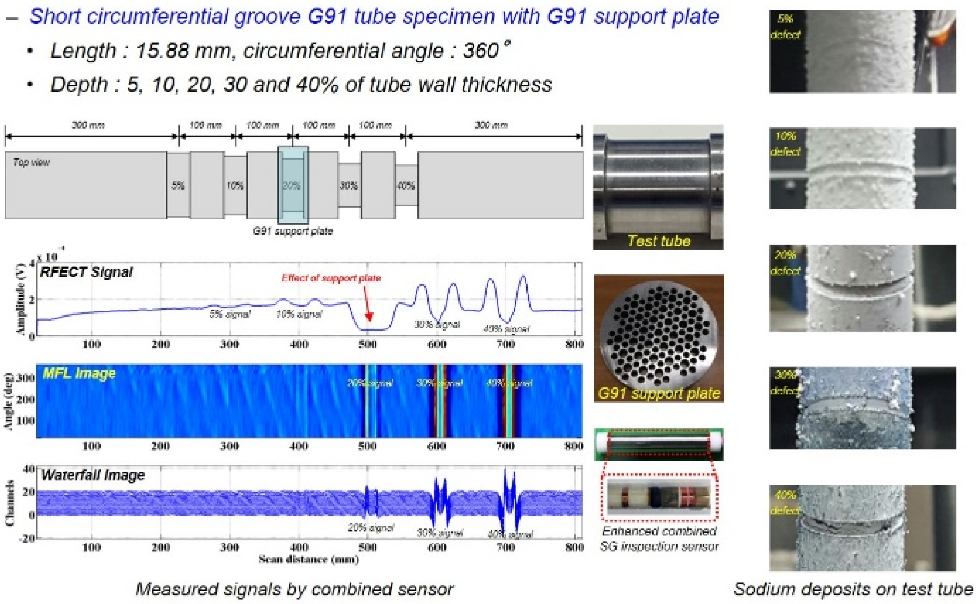
SWAT-3R tests were carried out to validate the applicability of new wastage correlations in a vertical straight tube bundle system for simulated practical SG operation conditions. In the reacting zone, the wastage rate can be somewhat conservatively predicted by using correlations of composite oxidation-type corrosion with flow (COCF) and liquid droplet impingement erosion (LDIE) with liquid film. Out of reacting zone, COCF correlation for NaOH environment can appropriately predict the experimental data of wastage rate

The mechanistic computer code SERAPHIM has been developed to evaluate the wastage environment during the tube failure accident in the SG of the sodium-cooled fast reactors. The numerical models for compressible multiphase flow, chemical reaction, and liquid droplet entrainment and its transport were incorporated into the SERAPHIM code. In this study, applicability of the SERAPHIM code was investigated through the analysis of the experiment on water vapour discharging into liquid sodium. It was demonstrated that the SERAPHIM code could predict the temperature distribution and the environment of the LDIE under the practical SG condition.

In order to limit the occurrence of sodium water reaction in SG after tube failure, remote inspection could be used. To reach this goal a combined Remote Field Eddy Current Testing (RFECT) + Magnetic Flux Leakage (MFL) testing probe has been developed and its performances are evaluated.

The performance of the combined SG tube inspection sensor has been improved and the detectability of the sensor was evaluated through several damage detection tests in air. Preliminary tests for evaluation of the effect of sodium deposits on the measured signals of the sensor were also conducted in a newly constructed sodium deposit test facility (Figure 3.69). The sodium deposit affects the RFECT signal but not significantly the MFL signal.

Figure 3.69: **Example of damage detection results obtained by the combined sensor and photos of sodium deposits on the test tube exposed in 350°C sodium**



New sodium facilities in support of sodium components and technology development

MECANA, STELLA1&2 and the Sodium Thermal-hydraulic Experiment Loop for Finned-tube Sodium-to-Air Heat Exchanger (SELFA) have been designed and/or constructed.

Global Actinide Cycle International Demonstration Project

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

The irradiation behaviour of the Am-1 test in the Joyo reactor, such as americium migration, was analysed and investigated in detail based on the PIE results for irradiation behaviour modelling. The Joyo irradiation experiment is currently suspended. The irradiation experiment will resume after the safety examination for the new regulatory requirements.

R&D on fabrication is in progress and the specifications of (U, Pu, Am, Np)OX, have been established at CEA. The overall programme on property measurements was defined and split between several laboratories. Figure 3.70 is a photograph of a uranium-americium oxide pellet fabricated by CEA.

The availability of americium is limited. Figure 3.71 shows a photograph of a new glovebox at Idaho National Laboratory that will be utilised to obtain and process americium supply.

Figure 3.70: **Photograph of CEA fabricated (U,Pu, Am, Np)O_2 pellet (LEFCA facility)**

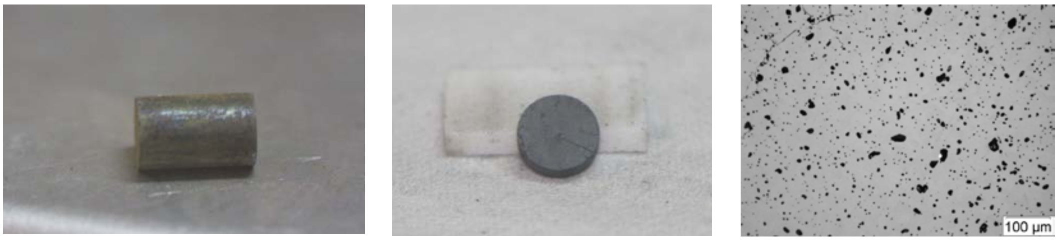


Figure 3.71: **Photograph of a new glovebox at Idaho National Laboratory**

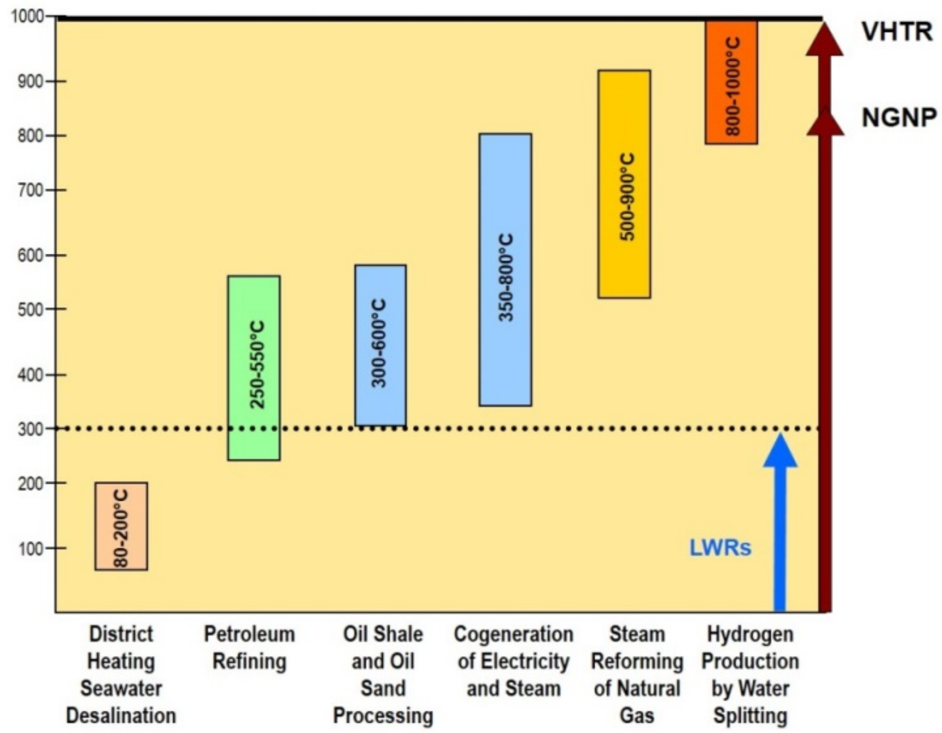
3.6. Very-high-temperature reactor (VHTR)

Main characteristics of the system

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

Use of helium as coolant and ceramics as core structure material allows operation temperature at core outlet as high as 1 000°C allowing for hydrogen production using processes that do not emit greenhouse gases, such as thermochemical cycles (iodine sulphur) or high-temperature steam electrolysis (HTSE). Even at moderate temperatures, VHTRs can be useful for hydrogen production with processes such as the Copper Chlorine cycle. Beyond electricity generation and hydrogen production, high-temperature reactors can provide process heat for use in other industries currently served by fossil fuels (Figure 3.72).

Figure 3.72: **High-temperature heat applications**



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

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

In the current projects of VHTR, the first candidate for electric power conversion unit is an indirect Rankine cycle applying the latest technology of conventional power plants, as this technology is available. However, direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles are also considered to be applied in the near future.

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

The technology is being advanced through near and medium-term projects, such as HTR-PM, NGNP, GT-MHR, NHDD and GTHTR300C, led by several plant vendors and national laboratories respectively in China, the United States, Korea and Japan. The construction of HTR-PM demonstration plant (two pebble-bed reactor modules with one super heat steam turbine generating 200 MWe) started in China (Figure 3.75) on 9 December 2012. Each reactor module will have a power of 250 MWth. The coolant gas temperature will be 750°C, which represents the current state-of-the-art for materials and the requirement of high-temperature steam generation. High-quality steam of 566°C will be fed into a common steam header. HTR-PM demonstration plant will be connected to the grid in 2018, which will represent a major step towards the deployment of Generation IV technology.

Status of co-operation

The VHTR SA 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 had 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. Extension of the system arrangement became effective on 30 November 2016 with the signature of China, Japan and the United States. Other implementing agents are in the process of concluding the arrangement.

The fuel and fuel cycle Project Arrangement (PA) became effective on 30 January 2008, with implementing agents from Euratom, France, Japan, Korea and the United States. The PA has been extended to include input from China and was amended in 2013. It went into effect in January 2014.

Figure 3.73: **HTR-PM first reactor vessel Installation (March 2016)**

The materials PA, which addresses graphite, metals, ceramics and composites, was signed by implementing agents from Canada, France, Japan, Korea, South Africa, Switzerland, the United States and Euratom by 16 September 2009, and is effective since 30 April 2010. China initiated the process for joining the project in 2010. South Africa's withdrawal from this PA became effective as of 21 November 2013. Canada withdrew from the materials PA at the end of 2012. The details to amend the PA to reflect China's (INET) joining, and to extend the duration of the incorporated Program Plan until 2015 have been finalised and approved by the VHTR SSC in 2014. The amended PA is expected to be signed by the signatories in early 2017 following signing of the updated Framework Agreement. A further extension of the PA through 2018 that would also add Australia (ANSTO) as a new member was also begun in 2016.

The hydrogen production PA became effective on 19 March 2008 with implementing agents from Canada, France, Japan, Korea, the United States and Euratom. In 2010, China expressed its wish to join this PMB. As a result, an amended Project Plan incorporating Chinese contributions and other countries' updated contributions was prepared under the consensus of the PMB and submitted for approval to the System Steering Committee in 2011 October. Preliminary updating of the Project Plan was achieved following the 14th meeting of the Hydrogen PMB in China 2014 December. However, finalising the PA is pending the next Hydrogen PMB meeting in early 2017.

The Computational Methods Validation and Benchmarks (CMVB) PA remains provisional. In discussions during the 13th and 14th CMVB PMB meetings in 2016, the PA, Project Plan (PP) and work plan for the first year were finalised. The CMVB PA is expected to be effective in early 2017.

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 SRP 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 VHTR SRP is structured into six projects; only three projects are now effective, and one is provisional, as discussed below:

- Fuel and fuel cycle (FFC) investigations are focusing on the performance of the TRISO-coated particles, which is the basic fuel form of the VHTR. R&D aims to increase the understanding of standard design (UO_2 kernels with SiC/PyC coating) and examine the use of uranium-oxicarbide (UCO) kernels and ZrC coatings for enhanced burnup capability, reduced fission product permeation and increased resistance to core heat-up accidents (above 1 600°C). This work involves fuel characterisation, post-irradiation examination, safety testing, fission product release evaluation, as well as assessment of chemical and 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 minor actinides (MA) in support of a closed cycle.
- Materials (MAT) development and qualification, design codes and standards, as well as manufacturing methodologies, are essential for the VHTR system development. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in the operating environments. For core coolant outlet temperatures up to around 950°C, it is envisioned to use existing materials; however, the goal of 1 000°C, including safe operation under off-normal conditions and involving corrosive process fluids, requires the development and qualification of new materials. Improved multi-scale modelling is needed to support inelastic finite element design analyses. In addition to other high-temperature heat exchangers, additional attention is being paid to the metal performance in steam generators, which reflects the current interest in high-temperature steam-based process applications. Structural materials are considered in three categories: graphite for core structures, fuel matrix, etc.; very/medium-high-temperature metals; and ceramics and composites. A materials handbook has been developed and is being used to efficiently store and manage VHTR data, facilitate international R&D co-ordination and support modelling to predict damage and lifetime assessment.
- For hydrogen production (HP), two main processes for splitting water were originally considered: the 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.
- CMVB focuses on subjects needed for the assessment of the reactor performance in normal, upset and accident conditions. This encompasses the construction of a phenomena identification and ranking table, computational fluid dynamics, reactor core physics and nuclear data, chemistry and transport and reactor and

plant dynamics. Code validation needs to be carried out through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported by HTR-10 and HTR-PM 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 2018. 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 Very-High-Temperature Reactor (VHTR) Fuel and Fuel Cycle (FFC) Project is intended to provide demonstrated solutions for the VHTR fuel (design, fabrication, and qualification) and for its back-end management, including novel fuel cycle options.

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

Fuel cycle back-end encompasses spent fuel treatment and disposal, as well as used graphite management. An optimised approach for dealing with the graphite needs to be defined. Although a once-through cycle is envisioned initially, the potential for deep burn of plutonium and minor actinides in a VHTR, as well as the use of thorium-based fuels, will be accounted for as an evolution towards a closed cycle. The task structure is shown in Figure 3.76.

Status of ongoing FFC activities

During 2016, significant work was accomplished in the areas of irradiation and PIE, characterisation, safety testing and back-end fuel cycle issues.

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

Figure 3.74: **FFC project 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 Characterization Techniques
- Task 2.4 Fuel Performance Modeling

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 Behavior and Waste Package

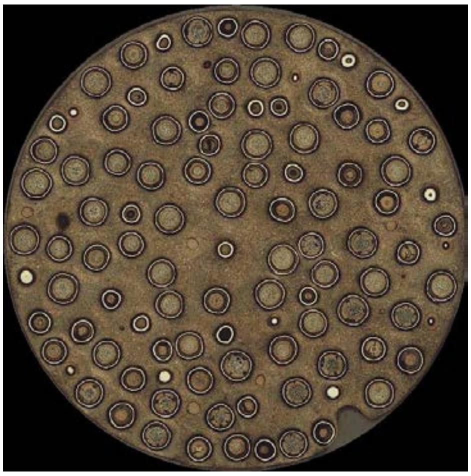
WP6 Other Fuel Cycle Options
- Task 6.1 Transmutation
- Task 6.2 Thorium Cycle

Irradiation and PIE

In the United States, AGR-1 post-irradiation examination (PIE) is complete and a final report has been issued. AGR-2 and AGR-3/4 PIE are underway at INL and ORNL. Capsule disassembly and dimensional measurements, non-destructive gamma spectrometric analysis of the empty graphite fuel holders and the fuel compacts, and optical microscopy of numerous compact cross sections have been completed. Figure 3.75 shows a cross section of an AGR-2 UCO fuel compact irradiated to a burnup of 11.0% FIMA at a time-peak temperature of 1 305°C. Analysis of fission product inventory in capsule components (to help assess fission product retention by the particles) and safety testing and destructive analysis of compacts is ongoing. Compact destructive examinations include deconsolidation-leach-burn-leach analysis, gamma counting of individual particles, finding and analysing particles with failed SiC, non-destructive particle X-ray analysis and particle microanalysis.

AGR-3/4 irradiation was completed in April 2014 and PIE is currently in progress. This includes capsule disassembly and dimensional measurements of the components, non-destructive gamma scanning of the fuel compacts and the matrix and graphite rings, and analysis of fission products on the capsule components to help quantify total fission product release from the fuel. Heating tests of fuel compacts, fuel bodies (i.e. intact capsule internals consisting of fuel compacts surrounded by matrix and graphite rings), and individual matrix/graphite rings are also planned, as well as destructive analysis of fuel compacts by stepwise, radial deconsolidation. An apparatus to deconsolidate AGR-3/4 compacts in radial sections is being developed and should be installed in a hot cell in 2017.

Figure 3.75: **Cross section of an AGR-2 UCO fuel compact irradiated to a burnup of 11.0% FIMA at a time-peak temperature of 1 305°C**



The PIE of High-Flux Reactor (HFR)-EU-1 containing Chinese and German fuel irradiated at typical pebble-bed conditions is also completed.

Chinese pebbles in HFR-EU1 were transported to ITU this year, and the PIE of Chinese pebbles is anticipated to begin in 2017, together with the HTR-PM irradiated pebbles at ITU. High-temperature test (one), deconsolidation (two), and coated particles examination will be performed on the HFR-PM pebbles.

In Korea, an irradiation of TRISO fuel began in the high-flux advanced neutron application reactor (HANARO) in July 2013 and was completed in March 2014. After five irradiation cycles in HANARO, the maximum burnup was 37 344 MWd/MtU. The evaluation of TRISO fuel service condition at HANARO is now completed. Non-destructive examination on irradiated rods (measurement of the rod diameters, γ-scanning, X-ray CT inspection, laser piercing, collection and analysis of fission gas), fuel compacts and graphite specimens (dimensional measurement, measurement of weights and densities, deconsolidation of fuel compacts, X-ray inspection, measurement of thermal diffusion coefficients of graphite disks) was performed. Destructive examination was also carried out on TRISO fuel particles (optical inspection, EPMA).

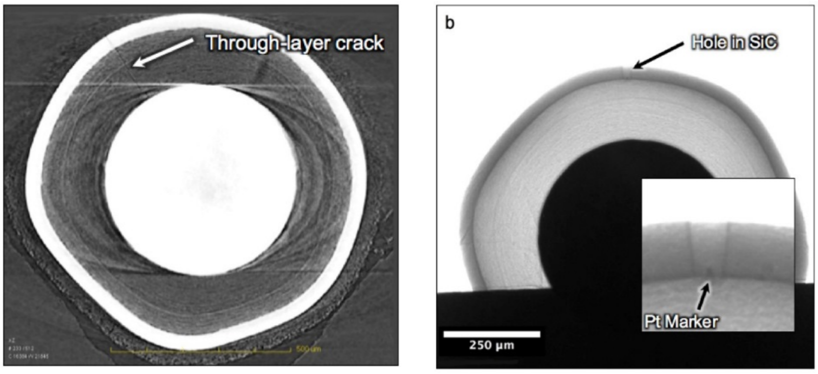
Fuel attributes and material properties

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 and the work will be reported soon. Plans have been established for Korean surrogate particles but are awaiting funding decisions in the parties.

In China, extensive characterisation of an oxidised SiC layer on TRISO fuel between 800 and 1 600°C was completed. 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.

Discussions were held at the June 2016 PMB meeting to share progress and plans for completion of the round-robin experiment to benchmark the leach-burn-leach (LBL) process. The United States presented a final report on the fabrication, analysis, and testing of simulated LBL defects containing depleted uranium (DU). Pre-burn leach defects were created by impact fracture of all the TRISO layers and post-burn leach defects were created by focused ion beam (FIB) milling a hole through the SiC. The impact-cracked defect structure resulted in an exposed kernel that would be leached by nitric acid in the first leach stage of the LBL process, while the FIB-milled defect structure resulted in exposure of an intact inner pyrocarbon layer that would only result in leaching in the second stage of the LBL process (after the burn). Non-destructive X-ray tomography was used to verify defect microstructures in every particle to be used for the round-robin experiment (Figure 3.76) and LBL testing was performed to verify the simulated particles performed as intended. The United States also reviewed the experimental plan for doping China TRISO particles containing natural uranium (NU) with the simulated defects, and presented statistical analysis of the expected outcome of the round-robin experiment for various native defect fractions in the NU-TRISO to highlight the need for a high-quality material with a low native defect fraction.

Figure 3.76: **a) X-ray tomograph of a simulated pre-burn leach LBL defect and b) X-ray radiograph of a simulated post-burn leach LBL defect.**



China informed the PMB that the TRISO particles that they had available for the experiment contained DU rather than NU as previously planned. This necessitated a revamping of the round-robin experimental plan, because there would not be sufficient difference in enrichment between the United States and China DU TRISO to differentiate between simulated LBL defects and any native defects that might be present in the China DU TRISO. The United States proposed that the United States and China samples be tested separately to ensure a successful and unambiguous experiment. The United States issued a revised report in December of 2016 that documented the preparation of US samples according to a new plan for the round-robin experiment (ORNL/TM-2015/722, Revision 2).

The US test samples consisted of TRISO particles containing ZrO_2 kernels doped with one, two, or four simulated LBL defects (one set with pre-burn defects and one set with post-burn defects for each participant), and a seventh sample for each participant containing no simulated defects as a control. Each US sample also contained a known mass of a coal impurity standard that would provide data on the accuracy of the LBL analysis for trace impurities. The United States packaged the samples prepared for Korea and China and sent shipping requests to KAERI and INET. Shipment to KAERI was completed in February of 2017, and shipment to INET is awaiting final authorisation from China.

China is currently working on authorisations to send samples of their DU TRISO to the United States and Korea. These samples will be analysed with LBL separate from the US samples. While the US samples will test the accuracy of the LBL analysis at each facility, the China samples will provide verification that each laboratory's LBL process does not introduce spurious defects that could result in a false rejection of a fuel lot. The China samples will also provide additional comparative results of impurity analysis on a representative TRISO material to explore the possible impact of impurity composition on the measurement.

Korea will be responsible for collecting and assembling all the round-robin experiment data from participants and writing the final report. A meeting is planned in association with the 2017 FFC PMB to present and discuss results.

Plans have also been made to hold the 4^{th} International Workshop on HTGR SiC Material Properties in conjunction with the 2017 FFC PMB meeting. The Technical Chair for the workshop (Gerczak, United States) sent invitations to prospective presenters and 12 abstracts have been received. Workshop attendees will also have the opportunity to join FFC PMB representatives on a tour of the Baotou pebble fuel fabrication facility.

Modelling of fission product release during heating tests (accident fuel performance benchmark)

This task started in September 2015 and is planned through September 2017. Calculations of a numerical calculation case were carried out by the different participants (INL, JAEA and KAERI). Preliminary results were sent to INL and compared. Results are overall in good agreement but large discrepancies were found at low release fractions (<10-5). Calculations on AGR-1 and HFR-EU1bis were performed and preliminary results were discussed at the PMB meeting held in Idaho Falls in June 2016. Final results on AGR-1 and HFR-EU1bis as well as additional results on AGR-2 will be discussed at the 2017 PMB meeting. Based on the preliminary results, Ag and Sr (and Cs at high safety test temperature) show good agreement between the three different codes. Larger discrepancies were observed for Kr, but calculated release fractions for this element are very low. Benchmark results still need to be compared to experimental data, but a large over-prediction is expected. The plan is to finalise the calculations by May 2017 and share the results at the next meeting in June 2017. The final report should then be completed by the end of September 2017.

Safety testing

The EU finished the accident safety testing of HFR-EU1 pebbles in 2015, and further reports are waiting for the new representative to bring them to PMB.

In China, the conceptual design of accident heating furnaces is underway but has been delayed somewhat 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, simulated heat-up test equipment was constructed for a simulated heating test in a laboratory. A specimen of Ag in a graphite container was tested at a maximum temperature of 1 700°C under Ar atmosphere.

In the United States, an additional AGR-1 transient temperature safety test has been completed. This test involved heating three AGR-1 fuel compacts (approximately 12 000 particles) to a peak temperature of nearly 1 700°C, using a temperature profile that mimics the peak core temperature during a depressurise loss of forced cooling event. No TRISO failures were observed during the test. The results have been presented at HTR2016 in November 2016. AGR-2 heating tests are in progress.

The following activities are scheduled to take place over the next several years:

- complete fabrication of final qualification fuel for AGR-5/6/7 campaign;
- fabricate AGR-5/6/7 irradiation test train and initiate irradiation of qualification fuel;
- complete PIE and safety testing of AGR-2 industrially produced TRISO particles;
- complete PIE and safety testing of AGR-3/4;
- develop accident testing furnace system to simulate air/moisture ingress events.

Beyond 2019, the AGR programme will complete the AGR-5/6/7 irradiation, complete PIE and safety testing on the AGR-5/6/7 experiment, including moisture and air ingress effects tests.

In Japan, oxidation tests with SiC-TRISO are being carried out. Oxidation testing furnace was built in 2015. Oxidation test is currently underway (using dummy SiC-TRISO particles with/without OPyC layer at ~1 600°C under 20 ppm to 20% of O_2 atmosphere). Results are expected to be obtained in December 2017.

Enhanced and advanced fuel

In the area of advanced fuel, both Korea and China are continuing to develop production routes for UCO, based in large part on the successful performance of this advanced high burnup fuel in the AGR-1 experiment. In Korea, UCO fuel kernel fabrication is ongoing. The dispersion of carbon black in the broth solution was studied through a combination of ultrasonic and high shear mechanical mixing with cooling. Thermal treatment experiments have been carried out using a newly built furnace system based on an established heating programme. China is interested in developing UCO ZrC-TRISO and has been evaluating ZrC coating layers. A research project on coated particles with UCO kernel and ZrC coating layer is ongoing. The first stage on UCO kernel is now completed. Two different carbon blacks were used to study the influence on the performance of UCO microspheres. The report is completed and was given to the Technical Secretary to be uploaded on the website. The first stage on Zr coating layer is also completed

Waste management and other fuel cycle options

This area covers three issues:

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

Some documents concerning fuel storage in the framework of Advanced High-Temperature Reactors for Cogeneration of Heat and Electricity R&D (ARCHER) will be made available to the PMB. No activity has been started yet within the FFC Project regarding the assessment of the VHTR thorium fuel cycle.

Project management

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

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

The next five-year Project Plan (2018-2023) is being prepared.

FFC conclusions

With the completion of the second five-year plan of collaborative work pending, the FFC Project of the VHTR is producing many positive results. The success has led to an ambitious third five-year plan (2018-2023).

Materials

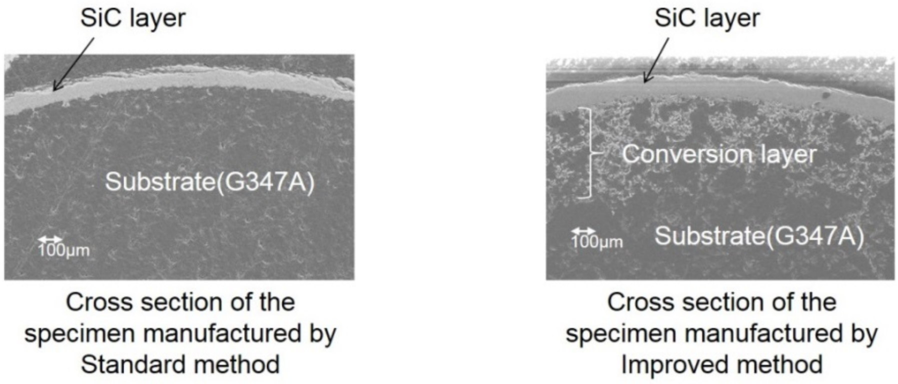
Although the term of the original Materials Project Plan (PP) was completed in 2012, the Materials Project Arrangement (PA) continued through 2014 while simultaneously pursuing an initial extension of the PP through 2015 and an additional extension through 2018. Changes in participation of the PMB are reflected in the new PPs and PAs. Canada withdrew unconditionally from the PA, effective 31 December 2012, at its own request, reflecting changes in its internal programmatic priorities. The conditional withdrawal agreement for Pebble Bed Modular Reactor LTD (PBMR) from the PA became effective on 21 November 2013, when it was signed by the final Signatory of the PA. Contributions for the extension of the PP through 2015 were developed by the remaining six signatories (European Union, France, Japan, Korea, Switzerland and United States), as well as China that will be joining the PA. The extended and augmented contributions were compiled into a revised PP and unanimously recommended by the PMB for approval by the VHTR System Steering Committee, which was received on 18 February 2014. Final approval of the extended PA is expected early in 2017 following signing of the updated Framework Agreement.

As part of the development of the revised PP, a thorough review was made of all the high-level deliverables (HLDs), which were consolidated, added, deleted or clarified to enhance accountability. All HLDs scheduled for completion prior to the end of 2014 were completed. Additionally, by the end of 2016, over 360 technical reports describing contributions from all signatories had been uploaded into the Gen IV Materials Handbook, the database used to share materials information within the PMB. This is well over twice as many reports as originally scheduled within the PA, reflecting the outstanding technical output of the membership. Uploads of the supporting materials test data are proceeding well for metals and are now in progress for graphite.

In 2016, research activities continued to focus on near- and medium-term project 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 multiple 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. One area of significant multi-signatory interest was in examining complementary approaches for improving the oxidation resistance of graphite. JAEA, KAERI and DOE are examining varying ways of applying SiC, boron and B4C coatings to graphite. An example of microstructures resulting from alternate coating methods used by JAEA for SiC coatings to test for improved oxidation resistance is shown in Figure 3.79 (Fujitsuka et al., 2015). Data to support graphite model development was generated in the areas of microstructural evolution, irradiation damage mechanisms and creep. Support was provided for both the American Society for Testing and Materials (ASTM) and ASME development of the codes and standards required for use of nuclear graphite, which continue to be updated and improved. Multiaxial fracture testing, at both the laboratory and component scale, as well as analysis of graphite was performed.

Figure 3.77: **Comparison of microstructures of graphite coated with a SiC layer by different methods to improve oxidation resistance.**



Examination of high-temperature alloys (800H and 617) provided very useful information for their use in heat exchanger and steam generator applications. These studies included an evaluation of the existing data base and an extension of it through creep, creep-fatigue and creep crack growth rate testing to 950°C. The most significant outcome of this work was the development and submission of an ASME Code Case for the use of alloy 617 as a new construction material for high-temperature nuclear components at temperatures to 950°C for 100 000 hrs. Data for the Code Case was contributed from multiple signatories (DOE, KAERI and CEA). An example of the details involved in establishing the Code Case is illustrated in Figure 3.80 that depicts which criterion was used at which combination of time and temperature to establish St, the maximum allowable general primary membrane stress intensity (Wright, 2015). The relatively simple-looking table contained in Figure 3.77 required detailed examination of data records and microstructures from hundreds of specimens of 617 from many sources and heats of material to establish which criterion would govern fracture under which condition.

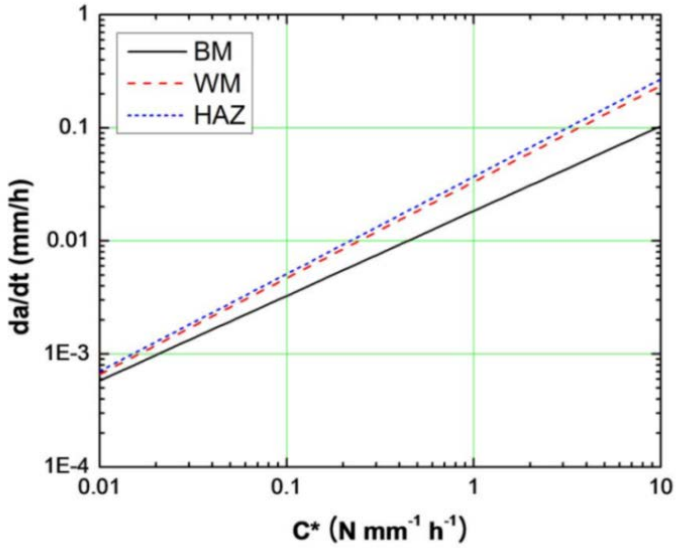
Other metallic materials were also examined as part of the PA. Irradiation and irradiation creep was studied on 9Cr-1Mo ferritic-martensitic steels and oxide-dispersion-strengthened steels, plus creep behaviour was examined in 2.25Cr-1Mo steel for steam generator applications. An example of work on creep crack growth testing of Grade 91 (9Cr-1Mo steel) is provided in Figure 3.79 (Kim et al., 2013), showing the difference in growth rates between base and weld metal. It is important to understand such behaviour, since advanced pressure vessels of VHTRs may be constructed of Grade 91 steel and creep crack growth behaviour during creep loading will be required for design and safety assessment of such components.

Figure 3.78: **Minimum values used in ASME 617 Code Case to determine which criterion governs the determination of St, maximum allowable general primary membrane stress intensity, for each time/temperature combination**

1% Strain	Tertiary Creep	Rupture

	Stress (MPa)										
Time (h)→	**1**	**3**	**10**	**30**	**100**	**300**	**1000**	**3000**	**10000**	**30000**	**100000**
Temperature (C) ↓	Minimum, All Criteria										
425	245	245	245	245	245	245	245	245	245	245	245
450	245	245	245	245	245	245	245	245	245	245	245
475	242	242	242	242	242	242	242	242	242	242	242
500	240	240	240	240	240	240	240	240	240	240	240
525	238	238	238	238	238	238	238	238	238	238	238
550	235	235	235	235	235	235	235	235	235	233	199
575	234	234	234	234	234	234	234	234	220	190	162
600	233	233	233	233	233	233	233	213	180	155	131
625	232	232	232	232	232	232	204	175	148	126	106
650	231	231	231	231	231	201	169	144	120	101	83
675	231	231	231	231	197	167	140	116	95	80	65
700	231	231	231	198	164	137	112	93	76	63	51
725	231	231	197	165	133	110	89	74	60	49	40
750	231	201	163	134	108	89	72	59	47	39	31
775	202	166	133	109	87	71	57	47	38	31	25
800	167	136	109	88	71	57	46	37	30	24	19
825	138	112	89	72	57	46	37	30	24	19	15
850	114	92	72	58	46	37	29	24	19	15	12
875	94	75	59	47	37	30	23	19	15	12	9.3
900	77	62	48	39	30	24	19	15	12	9.4	7.3
925	64	51	39	31	24	19	15	12	9.3	7.4	5.7
950	53	42	32	25	20	16	12	9.5	7.4	5.8	4.5

Figure 3.79: **Comparison of creep crack growth rates for base metal, weld metal and heat affected zone of Grade 91 steel at 600°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 thermomechanical properties of SiC and SiC-SiC composites and oxidation in C-C composites. The results of this work is being actively incorporated into developing testing standards and design codes for composite materials, and to examine irradiation effects and fabrication methods on ceramic composites for these types of applications.

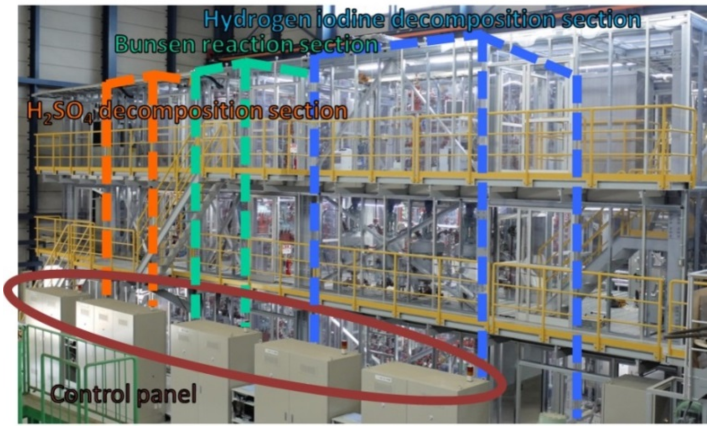
Hydrogen production

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

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

Japan, Korea and China are strongly involved in SI developments. Japan has been working on the verification of integrity and stability of industrial material components for use in the SI process. Japan has integrated the three reaction sections, i.e. HI decomposition section, $H2SO_4$ decomposition section and Bunsen section in a hydrogen production test facility constructed in JAEA Oarai Research and Development Center (Figure 3.9). On 26 October 2016, JAEA succeeded 31 hours of continuous operation of hydrogen production at the rate of 20 L/hour. Currently, maintenance and improvements are underway to achieve extension for continuous operation. In addition, basic design for the HTTR-GT/H2 test plant has been completed, consisting of a helium gas turbine power conversion system rated at about 1 MW electric power output and an IS-process H_2 plant with hydrogen production rate of about 30 Nm^3/h. Furthermore, Japan prepared a conceptual design for a commercial large-scale H_2 plant which showed a thermal efficiency of 50% can be achieved by incorporating innovative technologies such as H_2 permselective membranes, etc.

Figure 3.80: **H_2 production test facility in JAEA**



Korea has performed a 72-hour continuous hydrogen production reproducibility test with 50 NL•H_2/h in their scale SI integration facility. Korea has tested an improved Bunsen section and successfully completed a Bunsen reactor unit test for 24 hours in October 2016. Korea currently focuses on computer simulation using the KAERI-DySCo code to perform start-up dynamic analysis and safe operation condition analysis of the hydriodic acid distillation column and sulphuric acid distillation column for hydrogen productivity of 1Nm-H_2/h scale or more. This simulation information will be used for preliminary design of a pilot scale SI thermochemical process coupled via an intermediate heat exchanger (IHX) to the secondary helium loop.

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 rate of 60NL/h for 86 hours. During the operation of this facility, major key parameters of three sections (Bunsen, SA, and HI) were monitored and measured.

Canada has been active in developing the four individual steps (electrolysis, hydrolysis, thermal decomposition and physical separation) involved in the copper chloride cycle for an integrated ~50 L/h demonstration system. Many advancements in each step have been achieved, rendering the overall process practical and potentially economical for large-scale hydrogen production. Advancements in the electrolysis step (copper chloride-hydrochloric acid electrolysis producing hydrogen) involved identification and development of suitable membrane electrode assemblies, a unique cell design (a double membrane cell) and understanding of the temperature effects on the power requirements. These achievements led to the continuous operation of a single cell for several days (e.g. ~1 600 h) at a desirable power consumption (~0.7 V and 0.1 to 0.4 A per square centimetre). The earlier concerns with copper crossover into the cathode compartment was overcome through the above developments. A spray drier-type reactor is developed to produce the oxygen containing intermediate species and a two chamber reactor designed for oxygen generation. Work is progressing to integrate the different steps.

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

In France, the 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 worlds 1st solid oxide electrolyzer 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 hydrogen. Comparison of operating points of alkaline, proton exchange membrane and HTSE showed that the HTSE can be characterised as having a better efficiency and lower sensitivity to the price of electricity, but higher cost for initial investment. Globally hydrogen produced is cheaper with proton exchange membrane or alkaline electrolysis.

Computational methods validation and benchmarks

Four years after activity was suspended after the 10th provisional PMB meeting in 2010, the CMVB Project Management Board was restarted in Weihai, China, just before the HTR-2014 conference. On this meeting, the WP of the draft project plan (PP) were preliminary identified, and specific member countries were assigned to lead each WP. During the following two years, provisional members focused on these tasks and the detailed content of every work package of the draft PP.

On 11-12 July 2016, the 14th provisional PMB was held in the Institute for Energy and Transport of JRC, Petten, the Netherlands. All five provisional PMB members (China, United States, Euratom, Korea and Japan) attended this meeting. The current status of the CMVB research activities among member countries was presented at the meeting. Each WP and Task in the draft PP, which had been revised up to the 13th CMVB meeting, was reviewed and modified to reflect the discussions made at the meeting. Input was received from all participants. Five WPs were furtherly developed, each with task descriptions, schedules, contributors and leaders:

WP No.	WP Title	Lead
1	Phenomena identification and ranking table (PIRT) methodology	JRC (Euratom)
2	Computational fluid dynamics (CFD)	INET (China)
3	Reactor core physics and nuclear data	DOE (United States)
4	Chemistry and transport	INET (China)
5	Reactor and plant dynamics	INET (China)

After discussion of the content, the US DOE hosted the 15th CMVB provisional PMB meeting in Las Vegas, United States on 4-5 November 2016. Participants from all five provisional member countries attended. This PMB meeting mainly focused on the draft PP. In accordance with discussions about purpose/objectives, deliverables, schedule of each WP made at the meeting as well as these comments, the PP document has been modified. The draft PP containing the members' contributions and budgetary data as well as the Project Arrangement (PA) will be finalised for signature in early 2017 after the new Framework Agreement is signed. The next provisional PMB meeting will be held by JAEA in Oarai, Japan on 1-2 June 2017.

Figure 3.81: **KAERI Hybrid RCCS Test Facility**

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 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 in-core temperatures is under implementation. 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 (Figure 3.83). 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 (High Temperature Test Facility [HTTF], NSTF, Matched Index of Refraction Facility [MIR], etc.) have been constructed. Data from NSTF experiments is available for validation of air-cooled RCCS models while HTTF experiments began 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.

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Kim, W-G et al. (2013), “Time-Dependent crack growth behaviour for a SMAW welment of Gr. 91 steel”, *International Journal of Pressure Vessels and Piping,* Vol. 110, pp. 66-71.

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Chapter 4. Methodology working groups

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. Economics Modeling Working Group

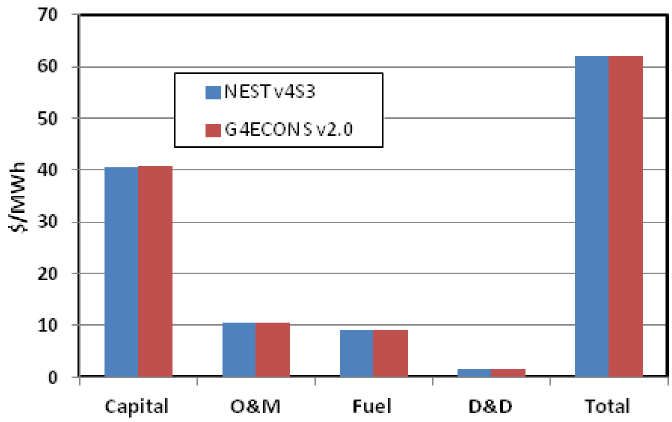
The objective of the Economic Modeling Working Group (EMWG) is to provide a methodology for the assessment of the Generation IV systems against the two economic-related goals stated in the Generation IV Technology Roadmap. EMWG published its cost estimation guidelines in 2007 along with an Excel-based software, G4ECONS v2.0, for economic assessment of Generation IV systems.

In 2016, EMWG published a paper demonstrating top-down approach for cost estimation of the Generation IV systems that are under development, as outlined in the EMWG's cost estimation guidelines; see Moore et al., 2016. The cost of the Canadian supercritical water-cooled reactor (SCWR) concept was estimated using the published costs of advanced boiling water reactor components and systems, using appropriate factors based on the similarities and differences between the two technologies. Cost estimation was done in 2007 United States dollars assuming an Nth-of-a-Kind SCWR. The analysis considers capital costs, operating and maintenance costs, fuel costs, and decommissioning costs in estimating the total capital investment cost and the levelised unit energy cost for the Canadian SCWR concept. Given the high degree of uncertainty in the future cost of the SCWR, sensitivity analyses were performed to identify key factors affecting the estimates. Limitations of the methodology are also discussed in the paper. It is recommended to revisit the economic analyses throughout the development of the Generation IV systems, as details of the component specifications become available for more specific cost estimates.

Benchmarking of EMWG's economic assessment tool G4ECONS v2.0 against IAEA's Nuclear Economics Support Tool (NEST) was completed for fast reactors with closed fuel cycles in collaboration with IAEA. 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 INPRO GAINS Project (IAEA, 2013). The benchmarking results for the break-even fast reactor were presented in the *2015 GIF Annual Report* and showed comparable outputs from the two models. Further benchmarking activities were carried out for a burner type, 1 000 MWe reactor with thermodynamic efficiency of 38%, capacity factor of 85% using metallic 19.5% Pu fuel with actinide recycle. The overnight capital cost was assumed to be USD 4 594/kWe, similar to the current Generation III reactors, with operating life of 40 years. Fuel fabrication and reprocessing costs were based on the data available in the INL Advanced Fuel Cost Basis report (INL, 2009). Economic assessment was performed using a discount rate of 5%, construction period of five years and decommission cost at 30% of the overnight capital cost. This benchmarking exercise revealed six differences between G4ECONS V2.0 and NEST, including, first core cost, discount and interest rates, delays/losses/lags in fuel cycles, reactor core assumptions (homogeneous versus heterogeneous), profitability indicators and decontamination and decommissioning costs. Therefore, careful considerations were given to the inputs for the two models to obtain comparable results; requiring some pre-calculations outside the models. With the consistent assumptions in both G4ECONS and NEST, the levelised unit energy cost was comparable, as shown in Figure 4.1.

The results of benchmarking of economic assessment tools for fast reactors were documented in a note and distributed to all fast reactor steering committee chairs (SFR, GFR, LFR and MSR) for distribution to other members. Benchmarking work done to date on both thermal reactors with once-through fuel cycles and fast reactors with closed fuel cycles is captured in a manuscript, which has been accepted for publication in a refereed journal: Moore et al., 2017.

Figure 4.1: **Levelised unit energy cost for burner fast reactor**



In a separate activity, G4ECONS v2.0 is being benchmarked against IAEA's Hydrogen Economic Evaluation Programme (HEEP) for economics of hydrogen production using Generation IV reactors.

The next version of G4ECONS, v3.0, with improved user interface underwent alpha testing within the EMWG and was improved based on the feedback. G4ECONS v3.0 was released to the EMWG members in October 2016 and is now available to the GIF community.

The Senior Industry Advisory Panel (SIAP) put forward two important recommendations at the PG meeting held in St. Petersburg (29-30 October 2015). The SIAP recommended that GIF: i) identify the attributes of Gen IV systems most attractive for industry (vendor/utility); and ii) investigate market conditions and timelines for commercialisation of Gen IV reactors. In the first phase of the inquiry a survey of key points on market issues will be identified (by approximately April 2017). The second phase of the inquiry is to identify measures to enhance market drivers (by approximately April 2018). The EMWG and the Sustainability Task Force will determine how to evaluate the merits of advanced reactors together with cost of construction and operation, e.g. high efficiency of uranium resource use, reduction of spent fuel volume and toxicity of LWR irradiated fuel, etc. The EMWG will develop an analysis of economic aspects of load following with Gen IV reactors to harmonise with fluctuating power of renewable energies. EMWG invited members of the SIAP to its meetings in 2016 to discuss economic issues related to market deployment of Generation IV systems. EMWG co-chairs also attended the SIAP meeting in October 2016 and participated in the discussions on market issues and integration of Gen IV technologies with renewable sources.

4.2. Proliferation Resistance and Physical Protection Assessment (PR&PP) Working Group

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 the SSCs (GIF/PRPPWG/2011/002), a set of frequently asked questions about the PR&PP methodology and applications (GIF/PRPPWG/2013/002), and the compendium of materials presented at the PR&PP Methodology Workshop held at the University of California, Berkeley in November 2015 (GIF/PRPPWG/2015/003).

In 2016, the PRPPWG increased its efforts to directly engage with the SSCs to a greater degree in the area of incorporating "PRPP-by-design" into the design process for each of the six GIF nuclear energy concepts. This is a follow-on effort to the joint study by the PRPPWG and the SSCs during the latter part of the previous decade. Overall, PRPPWG received a very favourable response to these efforts, and the SSCs identified a number of opportunities for additional information, collaboration or renewed dialogue. As a result, the PRPPWG has planned a workshop with the SSCs for April 2017 to meet face-to-face with SSC members.

The workshop will take place over two days. On the first day, there will be presentation by the SSCs and the PRPPWG to get a better understanding of the SSC needs and to convey the existing methodology. On the second day, there will be presentations by the IAEA and by the GIF industry group SIAP. This will be followed by a discussion of next steps by the SSCs and the PRPPWG to develop a regular and sustained interaction between the groups. This workshop may lead to additional face-to-face interactions with the SSCs, possible updates to the PR&PP Methodology, and most importantly to the increased use of the methodology during the design process for each of the six GIF concepts.

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 2016 in Paris, France and in Seoul, Korea.

The PRPPWG has been active with the RSWG in exploring the interfaces between the scopes of the two groups with an eye towards assuring that the methodologies being advanced by the two groups can be most effectively utilised by the GIF concept designers. The two WGs are jointly preparing a white paper that seeks to harmonise the methodologies of the two groups. The development of this white paper will form the basis of the groups' interactions going forward.

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).

In 2015, the document on frequently asked questions was adapted and formatted to create a tri-fold leaflet which has been distributed in various international symposia, workshops and conferences, including ICONE 23 (Chiba, Japan, 17-21 May 2015) and the American Nuclear Society (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 2015 and in the subsequent events of 2016.

Members of the PRPPWG have been active in the formation of the Nuclear Non-proliferation Policy Division of the ANS since its inception and in September 2016 a paper on the activities of the PRPPWG was presented at the first major conference sponsored by this division held in Santa Fe, New Mexico. The paper presented at a session co-chaired by a PRPPWG member, 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.

The 27th meeting of the group was held 13-14 October in Jeju, Korea. The location and timing of the meeting were chosen in order to co-locate it with a meeting of the Korean Radioactive Waste Management Society and involve Korean researchers that deal with nuclear fuel cycle issues, and therefore PR&PP topics. The technical workshop, held just prior to the PRPPWG meeting on 12 October 2016, involved key members of the Korean team that is developing a new sodium fast reactor as well as a pyroprocessing facility, as well as other researchers. Ideas were exchanged on how PR&PP concepts and methods can be used early in the design process. PRPPWG members provided a number of presentations on the PR&PP Methodology, its application, and a comparison of this methodology with the Facility Safeguardability Analysis (FSA) procedure that had been developed in the United States a few years ago. The FSA is a procedure for introducing safeguards by design for an advance nuclear facility with emphasis on how designers might interact with their state regulator and ultimately the IAEA in this process. No commitment was made for specific actions at the conclusion of the workshop, but the event provided a valuable opportunity for dialogue on incorporating the concepts of proliferation resistance early in design and planning. The workshop presentations are available on the GIF public website.

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 27th meeting, the group discussed opportunities to strengthen its co-operation with other groups, such as the GIF Task Force on Education and Training, 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 2016 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.

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 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 April 2016 at the IAEA Headquarters in Vienna, Austria, where fruitful discussions were conducted on opportunities for future collaboration. A representative of INPRO participated in the

27th 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 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 System Steering Committees (SSCs). The RSWG proposes safety principles, objectives, and attributes based on Gen IV safety goals to guide R&D plans. The RSWG also provides consultative support to SSCs and other Gen IV entities such as the Safety Design Criteria Task Force, 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 Integrated Safety Assessment Methodology (GDI).

The RSWG efforts in 2016 focused on the finalisation of the risk and safety white papers for the Generation IV systems with the approval of the SFR, LFR, and GFR documents by the GIF Expert Group and are now under open access on the GIF website. The VHTR document is under the final review process. In addition, the SCWR and MSR documents are being reviewed by the RSWG members. The risk and safety white papers are a joint work of the RSWG and each System Steering Committee to present high-level information about the safety assessment of their systems from the perspective of the applicability and helpfulness of the ISAM methodology. The MSR pSSC was invited to the RSWG meeting in October 2016, and the MSR pSSC presented safety characteristics and safety system features of the MSR system now under pre-conceptual design stage, and the first application of the ISAM methodology on the MSR system was provided. This important feedback from the users of the ISAM as a design tool for safety architecture improvement will be beneficial for the other systems when utilising the ISAM methodology for improving the safety designs.

Also in 2016, 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. The goal is 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. After the joint workshop in 2015 among RSWG members and the six SSC chairmen and representatives, the safety assessment document are being developed according to the schedule and to discuss SSCs proposals. Three safety assessment documents from SFR, 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 the Fukushima Daiichi nuclear power plant, owned by the Tokyo Electric Power Company (TEPCO), the RSWG reviewed the report: "Review of RSWG Methodology Against the Lessons Learned from Fukushima Accident" 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 Daiichi 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 2016, the RSWG worked in close collaboration with the GIF Safety Design Criteria Task Force contributing to the 2nd phase activity for development of the safety design guidelines (SDG). The RSWG supports the Safety Design Criteria Task Force 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 on the Safety Approach SDG report, and provide overall general recommendations on the safety approach and safety assessment for the Gen IV reactor system and specific technical suggestions on the safety design for the Gen IV SFR system. The RSWG also made the internal review of a draft version of SDC of LFR system. The review is for providing recommendations especially on consistency of LFR system's safety approach as the Gen IV reactor system based on the Basis for Safety Approach and ISAM reports.

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 was invited to the IAEA INPRO Dialogue meeting, and the ISAM methodology utilisation in the safety design process were introduced to the INPRO member states based on the ISAM and GDI reports developed by the RSWG. At the dialogue meeting, emphasis was put on the importance of the continuous improvement of safety design by using the ISAM methodology in order to achieve the high safety goals for Gen IV reactor systems and on the "build-in rather than added-on" concept of such safety features into the reactor system concept. The RSWG presented the ISAM methodology, especially on Qualitative Safety feature Review and Object Provision Tree tools, and its practical application examples at the IAEA INPRO Methodology-Updating Review meeting in November 2016. The RSWG also maintains internal contacts with the other methodology working groups and in particular with the Proliferation Resistance and Physical Protection Working Group (PRPPWG). In May 2016, the PRPPWG made the progress report and discussed the future collaboration plan with the RSWG in relation to the interface between safety, security and safeguards. 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.

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Chapter 5. Task Force reports

5.1. Task Force on Safety Design Criteria

In 2016, the SFR Task Force (TF) summarised the resolutions in response to the international reviews on the Safety Design Criteria (SDC) Phase I Report and issued the Safety Design Guidelines (SDG) on the Safety Approach, then proceeded the development of the Structures, Systems and Components SDG.

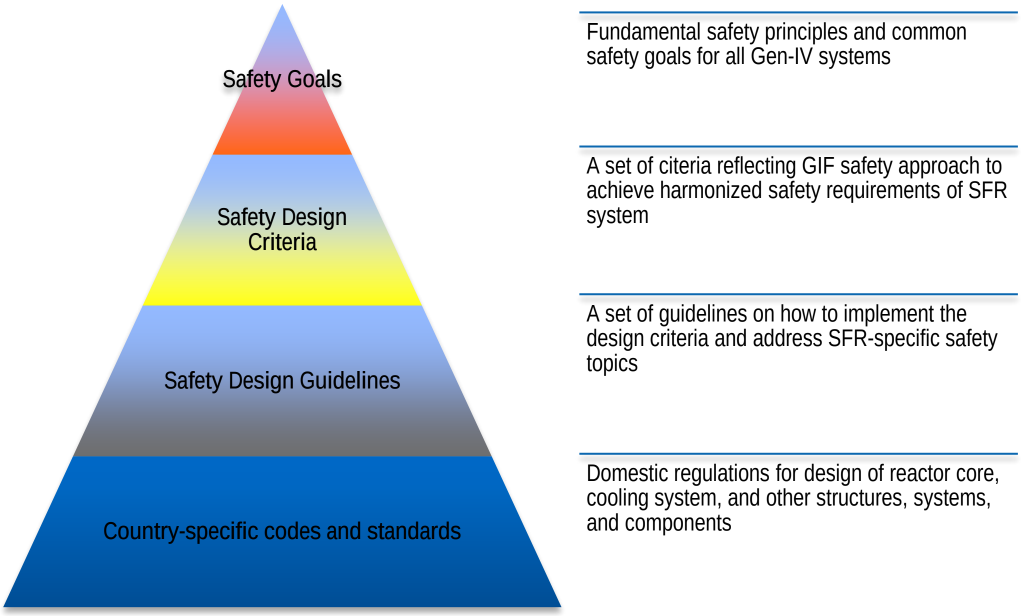
After the approval by the GIF Policy Group in May 2013, the SDC Phase 1 Report prepared by the TF was circulated to international organisations (i.e. IAEA, MDEP, NEA/CNRA) and regulatory bodies of the SFR developing states under the GIF (i.e. China, EC, France, Japan, Korea, Russia and the United States) for external feedback and enhanced interactions with the regulatory bodies. The comments received from the IAEA, the US Nuclear Regulatory Commission, the China National Nuclear Safety Administration (NNSA), and the France IRSN varied from general comments (e.g. safety approach for the Gen IV reactor systems, differences with Gen III systems, and relation between safety and security) to detailed specific recommendations for individual criterion related to technical characteristics of SFRs (e.g. sodium fire/water reactions and their consequences, accident initiators and parameters crucial to transient, design-basis accidents and design extension conditions). The TF held meetings in February, June and October 2016, and conducted a thorough analysis of the feedback received from these external reviews. The TF's response to these comments and recommendations are summarised in a separate TF report and will be available to the international reviewers after the EG and PG discussions and approvals.

The remarkable effort for the development of the SDC Phase 1 Report in a record pace cultivated the incentive and motivation for further technical interpretation and clarification of the SDC. Consequently, the Phase 2 activity of the SDC TF focused on the development of SDG, and the first report, "Guidelines on Safety Approach and Design Conditions of Generation IV SFR systems" (so-called "Safety Approach SDG"), was completed by the end of 2015 and approved by the GIF Policy Group in March 2016. The SDG reports are conceived as a detailed guideline documents one level lower than the SDC in a hierarchy of the safety standards as shown in Figure 5.1. It is intended to support practical application of the SDC to the Gen IV SFR design tracks, and it describes "clarification on technical issues for common understandings" with recommendations to cover wide range of applicable design options.

The first "Safety Approach SDG" guideline report provides safety approaches based on the general safety approach and technical issues listed in the SDC Phase 1 Report. Its primary content includes discussions 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)". It also covers the "sets of design guidelines for these two issues" and "provisions with designs options". To facilitate its external review, the first SDG report was distributed to the participants of the GIF-IAEA SFR Safety Workshop held in November 2016. At the workshop, the technical presentations were made by the TF members followed by the related technical discussions by the participants from the GIF member states, IAEA and the regulatory bodies from US, France and Russia. The GIF TF was also invited to the NEA GSAR (Ad hoc Group on the Safety of Advanced Reactors) meeting in September 2016, and the Safety Approach SDG was introduced for technical discussions in relation to regulatory framework on advanced reactor safety. The external review by the NEA GSAR is anticipated to continue in the coming year.

In order to discuss SDC/SDG with the stakeholders, the sixth joint GIF-IAEA Workshop on “Safety of Sodium-Cooled Fast Reactors” was held on 14-15 November 2016 at the IAEA Headquarters. The main purpose of the workshop was to present and discuss: i) the IAEA’s interim review on “Safety Approach SDG”, ii) the status of the SDC Phase 1 Report external review by national regulators, and further implementation of internal discussions on lesson learnt from the Fukushima Daiichi nuclear power plant accident based on the revised IAEA SSR 2/1; iii) the implementation of SDC and SDG by the designers of innovative SFRs concepts; and iv) the status of “Structures, Systems and component SDG”.

Figure 5.1: **Hierarchy of GIF safety standards, including safety design criteria and safety design guidelines**



5.2. Education and Training Task Force

Introduction

At the GIF meeting held in May 2015 in Chiba, Japan, the Policy Group (PG) directed the Technical Director to develop a scope of activities for the GIF Education and Training Task Force (GIF-ETTF). 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 which was developed and approved in April 2016 by the PG. The GIF-ETTF serves as a platform to enhance open education and training, as well as communication and networking of people and organisations in support of GIF.

Organisation

The GIF-ETTF consists of members nominated by the GIF Policy Group members and by GIF participants on a volunteer basis. Thirteen members from nine countries plus the European Community (EU) listed below participate in monthly conference calls, conferences, summer/winter schools, workshop, and support the Chair Patricia Paviet and Co-Chair Konstantin Mikityuk.

Participant	Organisation	Country
Patricia Paviet (Chair)	US Department of Energy (DOE)	United States
Konstantin Mikityuk (Co-Chair)	Paul Scherrer Institute (PSI),	Switzerland
Pavel Alekseev	Russian Research Center Kurchatov Institute, Russia	Russia
Concetta Fazio	Joint Research Center	European Community
Massimiliano Fratoni	University of California Berkeley	United States
Il Soon Hwang	Seoul National University	Korea
Xiaojing Liu	Shanghai Jiao Tong University	China
Takatsugu Mihara	Japan Atomic Energy Agency (JAEA)	Japan
Nolitha Mpoza	Department of Energy	South Africa
Youngmi Nam	Korea Atomic Energy Research Institute (KAERI)	Korea
Grace Pynn	Canadian Nuclear Laboratories Limited	Canada
Claude Renault	Commissariat à l'Energie Atomique et aux énergies alternatives (CEA)	France
Sun Jun	Tsinghua University	China

The GIF-ETTF will work during the period of 2016-2018. Each member participates in the GIF-ETTF with its own financial and human resources. The GIF-ETTF Chair and Co-Chair have been elected for two years and guide the overall activities to reach the set objectives.

Objectives

The principal objective of the GIF-ETTF is focused on promoting education and training by i) identifying and advertising current training courses, ii) identifying and engaging collaboration with other international education and training organisations, iii) developing webinar series dedicated to Gen IV systems and related cross-cutting topics and advertising these at the national and international level, and iv) creating and maintaining a modern social medium platform (such as LinkedIn www.linkedin.com/groups/8416234) to exchange information and ideas on general Generation IV research and development (R&D) topics as well as related GIF education and training activities. The development of webinars are intended to inform and stimulate not only young scientists' interest, but also managers, key decision makers and the general public about advanced reactors introducing foreseen advantages but also key R&D to be developed, past experience, current research and existing projects.

New available tools such as webinars prompted the GIF Education and Training Task Force to choose to exploit this modern internet technology and reach interest of a broader audience. Therefore, to promote training in the Gen IV system and to ensure a knowledgeable workforce exists, the GIF-ETTF is creating and making available to the public a series of webinars on topics specific to advanced reactor systems and cross-cutting subjects. These webinars are intended to be of interest not only to students currently pursuing formal education in universities but also to those already in the workforce who may need a refresher course or a better understanding of a specific topic, and most importantly to a more general public. We are seeking to develop world class webinars that will also be useful to people like quality assurance officers, data validators, technicians, managers, regulators and others who may benefit from an enhanced understanding of advanced reactor concepts in their work. The GIF-ETTF has established collaborative associations with universities and nuclear organisations actively involved in

Gen IV systems to foster the exchange of scientific and technical information for the development of webinars.

Short (60 to 90 minutes) webinar presentations on specific Gen IV systems and related topics are being developed in co-operation with our university and organisation partners. The webinars are recorded and archived to become a library or collection of information for online access from the Gen IV website (www.gen-4.org). The first series of a total of 13 webinars addresses various topics as shown in Figure 5.2 below.

The webinars consist of lectures and provide an opportunity for the audience to comment or ask questions at the end of each presentation. The system is designed for web conferencing and includes many features such as:

- attendee registration;
- attendee questionnaires about the webinar they followed;
- scheduled reminders for the registered participants and follow-up questionnaires, if desired;
- conferencing capabilities for 200 attendees at one time.

In connection with this activity, flyers are developed to advertise the webinars on the Gen IV website and on LinkedIn as well.

Figure 5.2: **GIF webinars (September 2016 to September 2017) (details on www.gen-4.org)**

September 29, 2016
Atoms for peace - The Next Generation
Dr. John Kelly, Department of Energy, USA

October 19, 2016 Closing the Fuel Cycle **Prof. Myung Seung Yang** Youngsan University, South Korea	February 22, 2017 Gas Cooled Fast Reactor **Dr. Alfredo Vasile** CEA, France	June 12, 2017 Lead Fast Reactor (LFR) **Prof. Craig Smith** US Naval Graduate School, USA
November 22, 2016 Introduction to nuclear reactor design **Dr. Claude Renault** CEA, France	March 28, 2017 Supercritical Water Reactors (SCWR) **Dr. Laurence Leung** CNL, Canada	July 12, 2017 Thorium fuel cycle **Franco Michel-Sendis** NEA/OECD
December 15, 2016 Sodium Cooled Fast Reactors **Dr. Bob Hill** ANL, USA	April 27, 2017 Fluoride-Cooled High-Temperature reactors (FHR) **Prof. Per Peterson** UC Berkeley, USA	August 22, 2017 Nuclear Fuel and Materials **Dr. Steven Hayes** INL, USA
January 25, 2017 Very High Temperature Reactors **Dr. Carl Sink** DOE, USA	May 23, 2017 Molten Salt Reactors (MSR) **Dr. Elsa Merle** CEA, France	September 21, 2017 Energy Conversion **Dr. Richard Stainsby** NNL, UK

Conclusion

The GIF webinars are very successful and demonstrate a strong need for such a resource to inform the general public, but also the scientific community about advances in the Gen IV systems. In addition, because of the passion and grass roots efforts of professionals and educators, the GIF-ETTF goal of creating an archive of online webinars has become a reality. The webinars are accessible for free online in two formats: audio-video recording as well as pdf slides at www.gen-4.org. Having free public access makes it even more attractive to the scientific community.

Chapter 6. Senior Industry Advisory Panel report

The Senior Industry Advisory Panel (SIAP) provides advice to the GIF Policy Group (PG) from the perspective of industry, on issues related to technology development, demonstration, deployment and commercialisation of advanced nuclear energy systems. Historically, the SIAP has been meeting at least once per year to consider systems and/or cross-cutting issues identified by the PG and to provide its recommendations relative to long-term strategic issues, including regulatory, commercial or technical considerations.

2016 has been a year of change for SIAP, after some years of reduced activity. First, SIAP has decided to meet twice per year, during the weeks of the GIF EG/PG meetings: in spring in Paris and in fall in the GIF host country, to ensure a better presence, continuity and more frequent advice to the PG. This is complemented by ad hoc SIAP telephone conferences to better prepare for the meetings, with meeting the agenda and circulation of draft documents. Second, this decision is connected with a longer-term perspective activity of SIAP, based on a three-year programme of work. Third, the PG has called for a reactivation or renewal of the membership of SIAP, asking for a more active participation and also adding the possibility to nominate an "alternate" besides the three full members. Ideally two full country members should be from the supply side (e.g. vendor or utility) and one from the user side (e.g. non-electric heat process industry). This call has been positively answered by the GIF members and the updated list of SIAP members brings high expectations for better SIAP contributions in the future. Nevertheless, SIAP still lacks representatives from the user side.

From an organisational point of view, the spring meetings of SIAP will mainly be dedicated to progress on the three-year work programme, made of two main tracks of activities. First, the SIAP's offers to review and comment on the plans for demonstration of mature GIF systems, from an industry oriented perspective. Second, SIAP will work in support of the GIF Vice-Chair working on Market Issues (currently Kamide-san), providing insights on key systems attributes, necessary to make GIF systems attractive for industry in a market driven low-carbon energy perspective.

The fall meetings, while also allowing to progress on the three-year programme, will continue to dedicate time to answer the yearly "charge" given to SIAP by the EG/PG, which was the standard method SIAP provided advice. SIAP is ready to continue to offer views and advice on any industry relevant issue considered as important by GIF.

Besides the main EG/PG meetings, SIAP reaffirms its interest to be invited to systems meetings, working groups or task forces, as appropriate to better support GIF goals. For example attending such meetings as the Systems Steering Committees, the Task Force on Sustainability, or the Economic Modelling Working Group would more directly inform SIAP discussions and subsequent advice.

The 2016 SIAP Paris meeting was attended by a rather limited number of members, which triggered the subsequent call for more active participation. SIAP discussed the elements for the review of mature systems. Maturity is connected to the level of readiness and timeline for commercial deployment around 2045, with, accordingly, a "pre-FOAK" to be commissioned in the period 2030/2035, a "FOAK" before 2040. These are challenging timeline, but reflect positions already expressed by SIAP previously on the need to get some clear signals on the demonstration phases for some systems, even if beyond the remit of GIF per se.

On that basis, a draft questionnaire was elaborated by the Secretariat, further discussed over telephone conferences, reviewed and finalised during the 2016 Seoul meeting. It was presented to the PG, within the scope of the three-year programme. After the PG comments are integrated into the questionnaire, it will be sent to one system group for a pilot test.

The questionnaire is structured along three chapters:

- description of the characteristics of the proposed pre-FOAK in the perspective of the FOAK: performance, specifications, schedule and costs;
- description of the pre-FOAK implementation plan: covering the technical, industrial, licensing, co-operation, financing, institutional aspects;
- perspective of the transition to and steps towards the FOAK

During the Seoul meeting, SIAP discussed a non-paper prepared by the Secretariat as a first input in support of the Vice-Chair on Market Issues. It was based on the priority attributes for Gen IV systems, as defined at the St Petersburg meeting (2015), and further broadened around the pillars of sustainability. It was quite fortunate that this came at the right time when the EG and PG were considering the future of the Task Force on Sustainability, and allowed SIAP to provide its views on the broad concept of sustainability, resting on an equilibrated balance between i) environment protection; ii) reliability of energy supply; and iii) competitiveness, the sustainability's three pillars.

The document will be further elaborated as a SIAP draft paper, for discussion at the spring meeting in 2017 and later constitute a second output of SIAP, besides the questionnaire on design review, within the three-year programme. The paper intents to describe key challenges and corresponding R&D areas and demonstration phases which would need to be planned and implemented to make Gen IV systems designs attractive for deployment in a market environment, in a foreseeable future. Challenges such as reducing the risk of off-site releases, reducing the long-term footprint of high-level waste, ensuring the integration of nuclear plants in the evolving low-carbon energy mix, and optimising the lifecycle costs, should be translated in concrete R&D and demonstration activities and offered for further discussions.

During the Seoul meeting, SIAP also commented on the 2016 charge.

The first charge was centred on the notion of sustainability and was therefore closely related with the discussions detailed above, in support to the Vice-Chair on Market Issues. SIAP confirmed its view that sustainability needs to be properly defined (there are diverse interpretations and definitions) and that, from its perspective, it should be broad in its coverage, as explained above, resting on the three pillars. When it comes on specifically judging the sustainability of Gen IV versus Gen III, SIAP considers that Gen IV can increase (depending on the features of the proposed systems) the overall sustainability of the global nuclear system, based today on Gen III. It is better to see the notion of sustainability as a challenge for the global system and not as a competition between Gen III and IV.

The second charge was asking to define steps for Gen IV deployment and corresponding R&D objectives. In answer SIAP established a concrete list of R&D objectives for the demonstration phase ("pre-FOAK") and of requirements for the transition from "pre-FOAK" to "FOAK".

The demonstration phase needs to i) provide information on the scaling effects and associated risks; ii) be global enveloping in terms of requirements if to be useful for the supply industry at large and deployment in diverse countries; iii) draw lessons in terms manufacturing, assembling, inspection, maintenance.

Going towards the FOAK requires to have scaling effects integrated, have siting issues solved, have an experienced operator, master the configuration management, have a consolidated supply chain for construction and operation, master the long-term legacy issues of waste management and decommissioning, address the workforce issues, get clarity on the costs/economics based on the return of experience of the pre-FOAK, have a business plan with a market perspective.

In conclusion, 2016 has been a fruitful year for SIAP, with the launch of the three-year programme. The future remains much in the hands of the SIAP members and their dedicated involvement in and between meetings. It also depend on the continued effective interaction with the PG, EG and systems, and their willingness to engage in the necessary demonstration steps, if indeed, some GIF systems want to be commercially deployable before 2050, increasing the global long-term sustainability of nuclear fission energy, as a key contributor to the transition towards a low-carbon economy.

Appendix 1. 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

ADS	Accelerator-driven system
AGR	Advanced gas-cooled reactor (United States)
ALFRED	Advanced lead fast reactor European demonstrator
ASTRID	Advanced sodium technological reactor for industrial demonstration
ATHLET	Analysis of Thermal-hydraulics of Leaks and Transients
ATR	Advanced test reactor (at INL)
AVR	*Arbeitsgemeinschaft Versuchsreaktor*
BWR	Boiling water reactor
CANDLE	Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing reactor
CATHARE	Code for Analysis of Thermal-hydraulics during an Accident of Reactor and safety Evaluation
CEFR	China experimental fast reactor
CFD	Computational fluid dynamics
CGR	Crack growth rate
CLEAR	China Lead-based Reactor
COL	Combined construction and operating licence
CRP	Co-ordinated research project
DHR	Decay heat removal

DNB	Departure from nucleate boiling
DHT	Deteriorated heat transfer
DU	Depleted uranium
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
GSAR	Group on the Safety of Advanced Reactors
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
LBL	Leach-burn-leach
LOCA	Loss-of-coolant accident
LWR	Light water reactor
MA	Minor actinides
MC	Monte Carlo
MELCOR	Methods for estimation of leakages and consequences of release (NRC code developed by Sandia National Laboratories)
MOSART	Molten salt actinide recycler and transmuter
MOU	Memoranda of Understanding
MOX	Mixed oxide fuel
MSFR	Molten salt fast reactor
MYRRHA	Multi-purpose Hybrid Research Reactor for High-tech Applications

NGNP	New generation nuclear plant
NHDD	Nuclear hydrogen development and demonstration
NPP	Nuclear power plant
NSTF	Natural Convection Shutdown Heat Removal Test Facility
ODS	Oxide dispersion-strengthened
PASCAR	Proliferation-resistant, Accident-tolerant, Self-supported, Capsular and Assured Reactor
PBMR	Pebble-bed modular reactor
PDC	Plant dynamics code
PGSFR	Prototype Generation IV Sodium-Cooled Fast Reactor
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
SG	Steam generator
SI	Sulphur Iodine
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
SWATH	Salt at Wall: Thermal Exchanges
TEM	Transmission electron microscopy
THTR	Thorium high-temperature reactor
TMSR	Thorium molten salt reactor
TORIA	Thorium-optimised Radioisotope Incineration Arena
TRISO	Tri-structural isotopic (nuclear fuel)
TRU	Transuranic
UCO	Uranium oxycarbide
ULOF	Unprotected loss of flow
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
ANSTO	Australian Nuclear Science and Technology Organisation
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)
CIAE	China Institute of Atomic Energy
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 (IAEA)
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
ITU	Institute for Transuranium Elements
LEADER	Lead-cooled European Advanced Demonstration Reactor
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre (Euratom)
KAERI	Korea Atomic Energy Research Institute
KEPCO	Korea Electric Power Corporation
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
NIKIET	NA Dollezhal Research and Development Institute of Power Engineering
NPIC	Nuclear Power Institute of China
NRA	Nuclear Regulation Authority
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
NUTRECK	Nuclear Transmutation Energy Research Centre
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 tenth edition of the Generation IV International Forum (GIF) Annual Report highlights the main achievements of the Forum in 2016 under the new Chair of the GIF Policy Group. The Framework Agreement, formally extended for ten years in February 2015, was signed by the remaining countries in 2016. The GIF is set to continue actively engaging in R&D on Generation IV systems with the extension of the four System Arrangements (sodium-cooled fast reactor, gas-cooled fast reactor, supercritical water-cooled reactor and very high temperature reactor) until 2026. Australia became the 14th country to join the GIF after signing the Charter in June 2016 and initiating the process to accede to the Framework Agreement. This annual report also provides a detailed description of progress made in the eleven existing project arrangements and under the Memorandum of Understanding governing R&D exchanges on molten salt reactors and lead-cooled fast reactors. In addition, it outlines the 2016 activities of the methodology working groups and the two dedicated task forces, one on the development of safety-design criteria and the other on education and training.

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